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BATTELLE-NORTHWEST
MONTHLY ACTIVITIES REPORT
FEBRUARY 1965

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BATTELLE-NORTHWEST
MONTHLY ACTIVITIES REPORT
FEBRUARY 1965

Compiled by
Department Managers

March 15, 1964

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PACIFIC NORTHWEST LABORATORY
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RESEARCH AND DEVELOPMENT

1. Reactor and Materials Technology

Production Reactor Support - The mechanical properties of uranium containing submicron dispersions of uranium carbide and uranium phosphides were determined up to 600 C. No marked difference in strength from normal uranium was found.

Metallographic examination confirmed the superior swelling resistance of uranium containing iron and aluminum additions compared to standard "N" composition uranium (Fe and Si additions).

Forty-five test capsules were charged in B-Reactor to test the irradiation behavior of additive-containing, unalloyed and alloyed (2 wt% Zr, Mo, or Cb) uranium.

C-product driver elements tested in KER operated successfully at a maximum specific power of 184.8 kw/ft to a maximum exposure of 2220 Mwd/ton, although fuel swelling up to 3.3 vol% was observed.

Meltdown studies of irradiated target elements indicated that failures resulted from high internal helium and tritium gas pressure.

Two irradiated, Al-1 wt% Li targets clad with anodized and non-anodized aluminum, respectively, exhibited no significant difference in tritium evolution rate when heated in vacuum to 300 C.

Be-Zr brazing wire (0.025 and 0.035-in. in diameter) for N-Reactor fuels was successfully extruded.

Mild steel chain, 90 to 100% "black" to neutrons, was produced by packing the chain in Ferroboration, then holding at 1160 C for 9 hr. The boron was not present as a surface phase subject to abrasion, but was apparently well protected in the mild steel matrix.

The uniform corrosion of Type 6061 and 6063 aluminum alloys in boric acid solutions (up to 500 ppm B) at 50 and 120 C showed a slight

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inhibiting effect of boric acid up to 4000 hr. Aluminum specimens containing a crevice and galvanically coupled to Inconel and stainless steel underwent pitting corrosion in the crevice.

Out-of-reactor tests simulating the N-Reactor graphite coolant system showed that high (40 ppm) oxygen levels cause severe pitting of carbon steel and light pitting of AISI 400 SS.

In-reactor tests of a nitrite-silicate corrosion inhibitor proposed as a substitute for the present sodium dichromate inhibitor (used in once-through cooled reactors) resulted in low corrosion rates for aluminum coupons and aluminum-clad fuel elements but undesirably high rates for carbon steel.

The first set of graphite burnout monitors was removed from N-Reactor. The peak burnout rate measured from startup was 0.2% per 1000 operating days. This is lower than expected for current operation due to low temperatures during the low-power, startup period. A burnout limit of 2% per 1000 operating days was used for the old Hanford reactors.

A Zircaloy-2, overbore tube was removed from C-Reactor after 2 yr use. Postirradiation examination indicated no radiation effects on structure or grain size. No significant corrosion was observed.

Analyses to evaluate possible pressure transients in K-Reactor downcomers during surges in coolant temperatures showed that accurate predictions of the pressure buildup during two-phase flow conditions require further investigation.

A series of tests were performed to determine hydraulic characteristics of various modifications contemplated for existing reactors. These included (1) a determination of the number of solid aluminum dummy pieces required to throttle the tube flow in K-Reactors, (2) the hydraulic characteristics of new overbore fuel elements, (3) the pressure drop across a charge of natural uranium mixed with bismuth and flux monitor pieces, and (4) the characteristics of a K-Reactor nozzle mounted on a C-Reactor, Zircaloy-2 tube.

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Reactor Materials Studies - Two stress rupture tests, two burst tests, and one crack propagation test were performed on PRTR pressure tubes. The crack propagation specimen had a 1-in. long crack which propagated at room temperature at a stress of 48,800 psi as compared to a predicted value of 49,200 psi.

Several precipitation and solution hardenable nickel-base alloys, irradiated at an average temperature of 740 C to an estimated exposure of 2×10^{20} nvt, were tested in tension at both room temperature and 650 C. Results from the tests conducted at 650 C indicated a marked reduction in ductility as compared to the room temperature tests.

An in-reactor test on annealed AISI 304 SS at 650 C, 10,000 psi showed a creep rate of 6.8×10^{-6} /hr after 300 hr at test conditions.

Tensile tests were conducted on BONUS transition-joint specimens after exposure in an out-of-reactor loop for approximately 725 hr. Tests at 550 F and at room temperature on specimens formed by both coextrusion and explosive welding processes revealed that little change occurred in the mechanical properties due to this exposure. The integrity of both types of joints appears to be very good, since all of the specimens failed primarily in the Zircaloy-2 region.

Preliminary results of tensile tests on irradiated stainless steels tested at temperatures between 400 and 750 C indicated a continual decrease in ductility as a function of test temperature up to 600 C. Above this temperature the ductility gradually improved.

A dynamic solution relating crack velocity to crack length in the double cantilever beam specimen was obtained subject to the effects of energy absorption rate.

A numerical method for analyzing thermal stresses in reactor fuel element end caps and transition joints was programmed for the computer, and trial solutions were begun.

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A test of the present insulating foil configuration in the test section of the model gas loop showed that thermal conductivity is dependent on the helium flow rate. A previous section of foil insulation did not exhibit this property.

A purity level of less than 1 ppm of water for gas temperatures up to 1700 F was achieved during the model gas loop test run.

Detail drawings of the second generation helium gas heater for the model loop were completed.

Initial microbalance studies of niobium exposed to oxygen at low pressures indicated a very low temperature dependence for the oxidation process, suggesting rate control by gas phase diffusion.

A Cb-1 wt% Zr alloy showed a much greater resistance to oxide spallation than cobalt-base or nickel-base alloys when exposed to flowing, Grade A helium at 2050 F.

Selected specimens of Coors commercial alumina products were readily vacuum outgassed at temperatures as low as 400 C. Weight was regained within about 30 min upon exposure to laboratory air.

A quartz helium diffusion cell was tested as a source of ultrapure helium. The helium product was found to contain hydrogen, apparently arising from internal contamination.

Rhenium foils, irradiated to 7×10^{18} nvt, were examined by transmission electron microscopy. Defect structures attributable to neutron irradiation were not observed.

A correlation of the drop-in load occurring in polycrystalline molybdenum upon initial yielding was established with neutron exposure and post-irradiation annealing.

An investigation of apparent yield points observed during strain rate changes in tensile tests of iron showed these were not caused by erratic crosshead motion during the rate changes.

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Optical and electron metallography on longitudinal sections of uranium specimens from a capsule irradiated at 450 C and 1000 psi to 0.13 at. % BU, corroborated previous conclusions that pressure almost completely inhibits the formation of tears even in high-purity uranium.

Data indicate that the beta-to-alpha transformation in pure plutonium is a shear (diffusionless) transformation.

X-ray diffraction studies on crystallographic relationships between the monoclinic structures of plutonium showed that a 90 degree shift in the [010] direction frequently occurs in specimens subjected to a beta-to-alpha transformation.

The source intensity of the Co⁶⁰ irradiation facility in the 3730 Building was increased from 86,000 to 185,000 curies. Dose rates of 1.65×10^7 r/hr are now possible.

A high-temperature (775 to 1150 C) graphite irradiation test was reinserted in the ETR and is operating satisfactorily.

Reactor Engineering - A status report on the PRTR, high-power-density core transition was prepared.

A proposed fuel reuse experiment between the PRTR and a fast reactor was completed.

The inadequate decontamination of a contaminated out-of-reactor loop (IRP) using the APSUL procedure recommended for PRTR (and N-Reactor) apparently resulted from reagents which had deteriorated with age.

A crevice corrosion test of a heated, spiral-wound assembly simulating a PRTR, high-power-density fuel pin showed that excessive clad corrosion could occur at the circumferential bands if localized boiling occurs.

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In the initial out-of-reactor pilot plant test of the PRTR boron shim, using 25-50 ppm B at stagnant flow conditions, the aluminum, shim well pitted excessively, presumably because of poor quality water. In a second test with 10 ppm B (Phase 1 condition) and expected PRTR water quality, but with stagnant flow, there was no pitting of the wall of the shim well. Boron deposition on aluminum surfaces was very low ($\sim 4 \mu\text{g}/\text{cm}^2$), representing about 1 mk of reactivity.

A report entitled "The Critical Discharge of Steam-Water Mixtures from Pipe Elbows and Tees" was prepared, presenting experimental results obtained for a piping system adjacent to a tee or elbow when opened to the atmosphere.

Experimental results obtained to date for boiling burnout at positions upstream from the exit of electrically-heated flow channels were surveyed. Based on this survey, further investigations of length effects at both high and low pressures are justified.

Reactor Fuels - Th-U fuel elements continued to operate satisfactorily in the ETR loop and reached an integrated exposure of 10,800 Mwd/ton with minimum swelling.

Th-U alloy was successfully clad with Nb-1 wt% Zr alloy by coextrusion.

Environmental tests on HTLTR materials showed no evidence of reaction between TD nickel and UO_2 or an Al_2O_3 - Gd_2O_3 mixture.

Four UO_2 -2 wt% PuO_2 high-power-density elements being irradiated in the PRTR reached a burnup of 2010 Mwd/ton_U. Four additional 2 wt% PuO_2 clusters, two with graded enrichment, were fabricated during the month.

A UO_2 fuel capsule, containing tungsten microspheres in coaxial holes, was irradiated in a horizontal position of the KW "Snout" facility to test the hypothesis that once-molten UO_2 can solidify during continued irradiation because of improved thermal conductivity.

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A report on Battelle-Northwest fuel fabrication experience with high-exposure plutonium was prepared.

An internal accounting system was set up to accumulate labor cost for preparing plutonium-bearing fuel elements. A flow sheet was developed for a 1 ton per day plant to produce mixed oxide fuel elements by both powder and pellet techniques. This hypothetical plant will be used for a basis for further fuel cost studies.

Intensive studies are under way to determine the nature and magnitude of stresses developed in fuel cladding during vibrational fuel loading. Preliminary results indicated that pneumatic vibration may induce less clad fatigue than electrodynamic shaking.

An instrument was developed to determine pour and tap densities of crushed particles in conjunction with a study of oxide particle size and shape effects on vibrational packing efficiency.

Specifications defining the chemical and physical properties for UO_2 powder used in the pneumatic impaction process were written.

A vibrationally compacted tube-in-tube UO_2 element that failed in the MTR had a small defect in the inner annulus of the inner tube.

Postirradiation examination of a failed "hot-wall" element indicated burnout due to eccentric loading of the powder around the pellets.

Attempts to rejuvenate a fuel element with plutonium alloy wire were unsuccessful due to mechanical problems.

Metallographic examination of a PRTR test element with replaceable rods showed no fretting or chemical corrosion after 16 days of operation in the MTR.

The "Salt Cycle" recycle fuel element fabricated for PRTR irradiation was found to contain organic contaminant. Replacement rods are being fabricated. An additional capsule containing "Salt Cycle" mixed oxide was prepared and is awaiting irradiation in the MTR.

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The first extrusions of a plutonium-bearing cermet were successfully made. Stainless steel-20 vol% PuO_2 cermet billets were coextruded with stainless steel cladding into 30-in. long, 1/4-in. diam pins.

High-density and low-density PuO_2 -stainless steel cermets were discharged from the MTR after successfully operating to 2 wt% BU with a calculated centerline temperature of 1300 C. Laboratory tests showed considerably less swelling in low-density pellets than in high-density pellets after 400 to 800 hr at 900 C.

High-fired plutonium oxide, obtained by burning plutonium metal at high temperatures, gave PuO_2 particle sized desirable for cermets. In a further study of source material, a joint program was initiated with Mound Laboratory to produce PuO_2 fused microspheres.

Short EBWR test rods being irradiated in the MTR reached an exposure of 4.8×10^{20} fissions/cm³ (17,000 Mwd/ton of fuel). A 21-rod bundle of full-size EBWR rods is being irradiated in the PRTR rupture loop, and twelve 3-rod clusters of full-size EBWR rods are being irradiated to long exposure in a Hanford reactor.

Mixed oxide powder was produced for fabrication of about half the Saxton reactor rods. The remainder of the oxide was heat treated to reduce gas content. Rod loading awaits approval of weld samples by Westinghouse.

Three kg of PuO_2 were impacted to 97% TD as part of a cooperative program with Argonne National Laboratory.

The electrical resistivity of a single crystal specimen of UO_2 was measured in argon to 1600 C by the four-probe method.

The PuO_2 in a UO_2 -20 wt% PuO_2 fuel was observed to be distributed in both small particle agglomerates and as a continuous phase surrounding the UO_2 particles.

A new hot stage specimen holder was constructed for reflection electron microscopy. A thoria sample holder for the high temperature

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dilatometer was calibrated, and the system is now being installed in a glove box. Calibration of the hooded tensimetric apparatus with CeO_2 was completed, and the equipment appears to produce excellent results.

2. Physics and Instruments

Analysis of exponential and PCTR experiments in support of NPR coproduct studies continued. Promising attempts to reduce discrepancies between theory and experiment include the use of transport analysis and more careful treatment of thermal flux depression corrections.

Use of the graphite scattering kernel rather than the gas scattering kernel has significantly improved the results of analytical studies of the low exposure PuO_2 lattice experiments. Calculations have been started to assist in determining the high exposure PuO_2 concentration in $\text{PuO}_2\text{-UO}_2$ fuels for the next series of lattice studies.

Simulation of N-Reactor Phase II (conversion) has begun. The initial model, consisting of lumped steam and condensate headers, will permit study of transients with turbine drop off or scram conditions.

Basic criticality studies were continued with PuO_2 -plastic mixtures in the undermoderated range (H/Pu^{239} atomic ratio ~ 16 , 8 wt% Pu^{240}). Data were obtained for correcting critical masses to "clean" systems. Core dimensions of the reflected parallelopipeds were varied, yielding critical masses in the range from 17 to 37 kg for determining the effect of Pu^{240} on critical dimension (shape of core) and mass.

In connection with the analyses of pulsed neutron source experiments and measurements of reactor noise, a Monte Carlo technique has been devised for calculating neutron decay constants. The computed and measured decay constants were within the calculational and experimental uncertainties for a plutonium critical assembly with an H:Pu ratio of 15.

Further measurements were made of the inelastic neutron scattering from zirconium hydride using the new rotating-crystal time-of-flight

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spectrometer. These measurements yielded the first observation of the excitation of the third oscillator level. Preliminary analysis of the results shows good agreement with the expected scattering from a simple harmonic oscillator.

An improved error analysis was developed for the analysis of fast-neutron total cross section data.

A summary of D_2O lattice experiments is being prepared for inclusion in the D_2O lattice data compilation of the European-American Committee on Reactor Physics.

In cooperation with Washington State University the momentum distribution of the beta-ray spectrum of Zr^{97} has been measured using the Double-Focusing Beta-Ray Spectrometer.

Code development work continued with improvements in ease of operation of the PHYSICS CHAIN, addition of a restart option to the two-dimensional diffusion code TWENTY GRAND, and updating of Th^{232} and protactinium in the Hanford Basic Library.

The effects of various materials in the EBWR core have been evaluated by EBWR critical experiments. Three-group perturbation theory predicts values which are in good agreement with experiment for rods composed of reactor poisons that have thus far been analyzed. Multithermal group rather than single-thermal group HFN analysis of the EBWR-fueled approach-to-critical lattice changes the effective multiplication from 1.004 to 1.002.

The 22 fuel cans for the first Pu-Al polyethylene Phoenix fuel PCTR experiments have been loaded with fuel, decontaminated, and inserted into the PCTR. Measurements will be made as soon as the axial thermal flux has been satisfactorily flattened.

Studies continued on the PuO_2 grain size effects in Pu-Be systems. Good agreement with published results was obtained using the ZUT resonance absorption code. Several cases are being set up for analysis with RBU.

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Preparations have been completed for the second run on the HTLTR mockup assembly. The run will include further tests of heater elements and thermocouples, and initial high temperature testing of the prototype control and safety rods. In HTLTR materials tests, samples of UO_2 -graphite matrix material were tested for 100 hr at 1100 C in nitrogen. There was no visible deterioration.

The analysis of the pulsed source experiments on the PCTR has been completed. It appears that the $k\beta/t$ method of determining shutdown will be quite useful in the range in which the delayed neutron contribution may be separated from the background with fair statistical certainty. This method is characteristic of the entire, quite complicated, reactor arrangement.

Reactor multiplication of the HTLTR has been calculated for several conditions of driver fuel enrichment and temperature. The gadolinium requirements to compensate temperature defects are being determined.

Some initial computer calculations using the THERMOS-HGR codes to calibrate the revised MELEAGER code for UO_2 - PuO_2 in light water-moderated lattices with varying degrees of moderation have been completed. While the revised version of MELEAGER yields a relatively much improved survey code, it appears that by sharpening the definitions of p and g employed, a better calibration will be possible.

The VESTA code (fuel utilization code) has been revised to include improved plutonium recycle techniques in thermal reactors and to improve the nature and form of the output including the annual cascade separative work requirements. Results have now been obtained for a relatively wide spectrum of hypothetical thermal and fast reactors in various combinations.

The MONTE CARLO code, RBU, is being employed to initially evaluate the characteristics of placing plutonium throughout UO_2 fuel rods, in the center of the UO_2 rods, or on the surface as an annulus. The lowest value of alpha for Pu^{239} is achieved with plutonium in the center of the rod, but other factors such as flux depression may override this apparent advantage.

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Additional run time appears necessary to firm up the MONTE CARLO statistics which may adjust the answers.

Analysis is under way as to the degree to which breeding could be achieved with plutonium fuels in demonstrated fuel configurations. The N-Reactor fuel in D₂O and Na graphite systems appears capable of low cost operation (approximately 0.5 mill/kWh_e) with conversion ratios of 0.95 to 1.0+ using equilibrium plutonium compositions enriched with tails uranium.

A stainless steel foil thermal transducer has been fabricated which produces a sinusoidal thermal wave in a test sample. This equipment is being used to collect data for evaluation of the sinusoidal thermal test technique.

Use of an electrostatic transducer has been practically demonstrated with measurement of elastic properties of plutonium samples.

A solid state electrometer was developed, using insulated gate field-effect transistors, for use in measuring the charge accumulated on falling raindrops.

A satisfactory gamma background subtraction method was devised to enhance the measurement sensitivity of the algae radionuclide uptake monitor for use in biological studies.

Additional analog simulation transient studies were made to update the biological radioisotope system analog model for Te¹³² and I¹³². These parent and daughter radioisotopes were injected into the bloodstream in the standard-man model yielding curves that seemed to be typical.

A recent whole body count of the Eskimos at Anaktuvuk Pass, Alaska, showed the average Cs¹³⁷ body burden to be 10% lower than it was in November, but 40% higher than it was in January, 1964.

The metabolism and radiation dose of technetium isotopes are being studied. In particular, this study will be helpful in determining the internal

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radiation dose from Tc^{99} which will be separated in large quantities from radioactive wastes at Hanford.

A copper heat baffle incorporated into the small radiation calorimeter gave a large improvement in its performance by reducing the temperature variation down to only 5×10^{-6} C.

A scavenging test (REDOX-4) using the REDOX stack off-gases was conducted in the 0.42 mm/hr snow of January 10, 1965. The washout coefficient measured for inorganic gaseous iodine was $4.2 \times 10^{-5} \text{ sec}^{-1}$, which is nearly the same as measured in the 1.0 mm/hr snows of December 14, 1965 (PUREX-2 and PUREX-3). The predominant crystals in these two snowfalls were columns and spatial dendrites, respectively. The lower terminal velocity of the columns (and correspondingly longer residence in the plume) probably provides a higher washout per depth of snowfall and is likely the reason the coefficient in REDOX-4 was not 0.4 times that in the PUREX tests.

3. Chemistry

Two additional irradiated N-Reactor coproduct target elements were subjected to rapid thermal transients to simulate the effect of overheating followed by delayed emergency cooling. In both tests the elements failed and the core material flowed away from the element at a temperature of 800 C.

Cross mixing of about 6% in "half-reactor" process water experiments was measured using Br^{82} as tracer.

Only minor changes, most of them showing decreasing concentrations, were observed in radioisotope analytical results for the ground-water monitoring program during the current month.

Two additional plutonium rods were ignited at about 600 C in air flowing at 3 cm/sec. The size distributions of the large particles were essentially the same as obtained from earlier ignitions.

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In codissolution studies on aluminum-clad thoria, increasing the concentration of aluminum ion has been shown to markedly decrease thoria dissolution rates. However, a threefold increase in fluoride ion concentration can overcome a sixfold increase in aluminum concentration.

Di-n-octyl amine and di(sec-butyl) phenyl phosphonate were found to be poorer extractants than dibutyl carbitol for Po-Bi separation.

Uranium has been shown as an adequate (mass transfer standpoint) stand-in for Po^{210} in solvent extraction studies of the recovery of Po^{210} from bismuth. Pilot plant studies of Po^{210} recovery by solvent extraction (dibutyl carbitol) continues to be promising with $> 10^4$ separation factors between bismuth and polonium.

An alternate process for separating rare earth concentrates from manganese was developed. It uses sulfate precipitation followed by hydroxide metathesis and dissolution with HNO_3 -HEDTA.

Denitration of bismuth raffinates from solvent extraction processing was successfully demonstrated in the pilot plant spray calciner.

Sodium persulfate was shown to be a more desirable cerium oxidant than the potassium salt. Peroxysulfate was also tested for solvent cleanup application.

The feasibility of extracting Co^{60} with D2EHPA was demonstrated.

Recovery of $> 99\%$ of neptunium and plutonium from Purex waste sludge with D2EHPA has been demonstrated in laboratory studies.

Adaptation of a continuous persulfate oxidation process for separation of cerium from rare earths has been demonstrated with excellent results in pilot plant pulse columns.

The ability to transmit process control data between the GE-412 process computer and remote processes via existing telephone lines was investigated.

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The need for good process control in the Salt Cycle coelectrodeposition has been emphasized by difficulty in producing UO_2 -20 wt% PuO_2 on a scaled-up basis. Two approaches—one electrochemical, the other photometric—give promise of a method for determining cell and deposit composition.

The High Level Radiochemistry Facility A-Cell cleanup following the glass demonstration is proceeding satisfactorily prior to installation of Pm^{147} purification and process equipment.

Sulfate-free Purex wastes were studied in glass-forming experiments. Behavior was similar to sulfate-containing wastes.

Studies relating spray calciner-melter operability with phosphoric acid concentration in simulated 1WW feed were completed. Separate addition of solid fluxing agents to the melter was found feasible using 0.5-in. borate-containing glass balls.

Use of the GE-412 computer for data logging and data reduction from 25 input signals during three waste calcination runs was very successfully demonstrated.

A pilot scale run was completed to demonstrate the use of a horizontal plate filter with a steam stripper and two bed anion-cation ion exchange system for decontaminating alkaline condensate. All radioisotopes were reduced below their respective MPC_w values except Ru^{106} which was 1.8 MPC_w .

Installation of the Containment Systems Experiment vessel continued on a delayed schedule, and installation of laboratory equipment was started. Two runs were made in the Aerosol Development Facility in which stainless steel clad UO_2 specimens irradiated at 2×10^{17} nvt were melted as a fission product aerosol source. No firm conclusions can be made until analytical results are available, but a very tentative observation is that iodine from the "real" fission product release sources did not behave differently than that

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from earlier synthetic source releases. Two "cold" Zircaloy-clad UO_2 specimens were also melted.

In the study of disposal of reactor off-gas to ground, an analysis of a gas flow problem is under way for homogeneous soil in which the horizontal permeability is greater than the vertical permeability.

Experiment designs were developed for heating tests of specimens of calcined and vitrified wastes.

The program "Steady State Flow Through Soils" was evaluated to determine the inherent error due to the approximation of continuous derivatives with discrete segments used in the solution of finite difference equations. Also, the error magnitude related to averaging permeabilities about the central node in solution of flow through heterogeneous soils was determined.

A submergible television camera has been found useful for viewing the Columbia River bottom.

A tracer test was run in three piezometers which are sand packed in a well.

Activation analysis followed by multidimensional gamma spectrometry has been used to characterize samples from two basalt flows.

Chemically and radiation-produced free radicals from p-nitrobenzoate were found to be distinctly different in ESR studies.

High resolution gamma ray spectrometry is being evaluated for measurement of U^{235} and Pu^{239} fission product yields.

Activity from the USSR underground test, which vented, has been detected in air samples.

Sodium hydroxide was found to be an acceptable substitute for ammonium hydroxide in the rare earth chromatographic separation process, and the elutriant study for this process has been completed.

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Analytical studies (paper) of reentry behavior have been continued to guide encapsulation development.

Outfitting A-Cell for the promethium purification program has continued, as have compatibility studies using various metals and Nd_2O_3 as a simulant for Pm_2O_3 .

4. Biology

A 6% incidence of fish carrying columnaris organisms was observed in fish samples at the Priest Rapids dam fish ladder. The incidence last year was 8%. None of the fish showed clinical symptoms of the disease.

Preliminary results indicate that eggs from local chinook salmon stock require about 1000 temperature units (sum of daily water temperature above 32 F from fertilization to hatch) to hatch compared with 900 for Puget Sound salmon eggs. The significance of this apparent difference is not yet known.

Two ewes being fed Cs^{137} daily (1 $\mu\text{Ci}/100$ g feed) have received 800 to 765 μCi over a period of about 70 days. Body burdens of these animals reached approximately nine times the average daily Cs^{137} intake 16 days after feeding began and have remained relatively constant.

Long-chain fatty acid synthesis was demonstrated in the mitochondrial fraction of dog lung tissue homogenates, similar to that previously reported for rat and rabbit lung. It was also found that rat lung mitochondria are capable of esterifying fatty acids.

Total body exposure of mice to 500 R is followed by a depression of hemolysin plaques/ 10^6 spleen cells, hemolysin plaques/spleen, spleen index, and serum hemolysin titers. This decrease in the primary antibody response occurs regardless of variations in the time between antigenic stimulation and irradiation. Timing is an important factor at lower radiation doses. Exposures to 250 R, at 1 day before or at 3 days after antigenic stimulation caused by significant decrease in the primary antibody response,

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but exposure 1 day after stimulation produced only a decrease in serum hemolysin titers.

The average Cs^{137} body burden in a control group of 25 Anaktuvuk Pass Eskimos was 640 nCi on January 21-22. This value is about 15% lower than those obtained in November, one-half those of last July, and 50% greater than those obtained 1 yr ago.

5. Radiation Protection

Analysis of dietary questionnaires and results of whole body counter examinations for a 9-month period indicated a significant correlation between milk consumption and Cs^{137} body burden. It was found that an average of 26% of the Cs^{137} body burden was obtained from milk. The effective half-life of Cs^{137} in the body calculated from these data is 67.3 ± 22.3 days, which agrees well with the ICRP value of 70 days.

Design of a new thermoluminescent reader was completed. The design provides for improved stability within the heating and light integrated circuits and acceptance of larger quantities of thermoluminescent materials with resultant increased sensitivity. Ample space and design considerations will permit installation of an optical system so that specific light wave lengths may be examined independent of the total luminescent glow. Specific wave length study may provide a lower background for the system by eliminating nonradiation caused transitions.

An improved and wider time range system for pulsing the electron Van de Graaff for special dosimetry physics and chemistry studies is being designed and installed. The addition of this system to the Van de Graaff accelerator when coupled with the capabilities of the flash x-ray equipment will provide a precisely controlled irradiation capability with exposure durations ranging from 0.1 μsec to many hours.

A joint proposal by the Radiation Protection Department, Physics Laboratory, and Applied Mathematics was prepared and submitted to the

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U. S. Army Electronics Command, Fort Monmouth, New Jersey. The objective of the proposed research is to develop an improved land fallout prediction system based on the dynamics of the fallout environment from time of burst to fallout deposition.

TECHNICAL AND OTHER SERVICES

The average concentration of P^{32} in the flesh of whitefish caught at Hanford was 46 pCi/g. This was less than half that found during January 1964, but quite similar to 1963 experience. The rate of transport of P^{32} in the river as measured at Richland dropped from an average of 40 Ci/d during December to about 25 Ci/d during January. This apparent decrease was not indicated by bulk release rates measured at the outfalls by IPD-GE which showed no change between December and January. The value of 25 Ci/d, if valid, would represent a new low for this time of the year.

The U. S. Testing Company began analyzing environmental samples on a limited basis on February 23. Samples which are being delivered to them are: air sample scrubbers and filters, water samples requiring total beta only, one large volume water sample per week for detailed isotopic analysis, beef thyroids, milk, and oysters. This limited delivery will be continued for several more weeks until U. S. Testing has completed calibration. The Radiological Analysis group will continue to analyze other off-site samples so that our schedule may continue at the reduced level established in December.

New studies are underway, at the request of RLOO-AEC, of the effect of various postulated production reactor accidents on the Wahuake Slope, assuming that the entire area was developed for farming. The economic parameters have been determined, analyses based upon them have been made, and conclusions are being formulated.

A previously used plan for sampling steam generator tube bundles during production outages was reviewed. This plan is being updated and

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expanded to provide an overall sampling procedure commensurate with the current goals.

Technical assistance in the installation, adjustment, and conduction of acceptance tests for ground detection devices on the 105 Building reactor coolant pressure monitoring systems has been completed at B and C areas.

A solution was obtained and an EDPM program written to evaluate the transient heat transfer and the maximal temperatures experienced in a source laden slab which radiates heat into two media at different temperatures.

A table of particle sizes and proportionate mixes for a 0.505-in. ID cylindrical fuel element was computed which lists three almost equally desirable alternatives from a density standpoint. Studies were also made on modes of packing which would have the property of uniformly distributed voids.

The attenuation properties and the direction of motion for a spectral element of a plane wave pulse propagating in a semi-infinite medium have been evaluated in terms of material parameters and the geometric orientation of the plane wave. The ratios of incident to rebounding potential amplitudes of spectral elements have been evaluated by solving the appropriate stress-displacement equations for liquid-liquid, liquid-solid, solid-liquid, and solid-solid interfaces.

An analysis of plutonium burn-up (PRTR) data was continued. Error evaluations from prior analyses were employed in the function used to compute the cross sectional ratio, in an attempt to estimate the error propagated to the ratio.

An experimental design was proposed for obtaining estimates of the variability and chemical reactivity of each of ten types of graphite.

A flow chart of the PRTR fuel fabrication system was completed. A design for an experiment relative to the Nupac and Vipac subsystems was proposed.

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Work continued on the statistical calculations subroutine for non-linear search program to be used to obtain least-squares estimates and joint confidence regions for the diffusion constants of an equilibrium model describing the extraction of sulfuric acid by Toa-Benzene.

A statistical analysis was begun to investigate the effect of diet and time on the Cs^{137} body burden of villagers in Northern Alaska.

SUPPORTING FUNCTIONS

Plutonium Recycle Test Reactor output for February was 1719 Mwd for an experimental time efficiency of 91% and a critical efficiency of 90%. There were five operating periods during the month, two of which were terminated manually, two were terminated by scrams, and one continued through month-end. A summary of the fuel irradiation program as of February 28, 1965, follows:

	<u>Al-Pu</u>		<u>UO₂</u>		<u>PuO₂-UO₂</u>		<u>Other</u>		<u>Program Totals</u>	
	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>	<u>No.</u>	<u>Mwd</u>
In-Core	0		6	1864.8	74	15894.2			80	17759.0
Maximum				403.0		451.4				
Average				310.8		214.8				
In Basin	7	572.5	27	3076.1	61	9248.4			95	12897.0
Buried							1	7.3	1	7.3
Chemical										
Processing	<u>68</u>	<u>5465.8</u>	<u>35</u>	<u>1965.8</u>	<u> </u>	<u> </u>	<u> </u>	<u> </u>	<u>103</u>	<u>7431.6</u>
Program										
Totals	<u>75</u>	<u>6038.3</u>	<u>68</u>	<u>6906.7</u>	<u>135</u>	<u>25142.6</u>	<u>1</u>	<u>7.3</u>	<u>279</u>	<u>38094.9</u>

(Note: $\text{Mwd/Element} \times 20 \approx \text{Mwd/ton}_U$ for UO_2 and $\text{UO}_2\text{-PuO}_2$)

Heavy water loss and indicated helium loss for the month were 455 lb and 109,509 scf, respectively.

The main maintenance jobs during the month's outages were primary system leak repairs and diesel generator wiring repairs.

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The irradiation of the EBWR cluster in the Fuel Element Rupture Testing Facility (FERTF) was completed with the reactor shutdown of February 24. The test element accumulated about 552 hr operating time and about 10.6 Mwd in the loop during FERTF Test No. 3. The power level of the fuel element was found to be about 480 kw.

Chemical processing of spent PRTR fuels has been delayed because CPD has postponed the next chemical processing campaign until July 1965.

Total productive time in the Technical Shops Section for the period was 27,719 hr. Distribution of time was as follows:

	<u>Man hr</u>	<u>% of Total</u>
N-Reactor Department	2 844	10.3
Irradiation Processing Department	5 227	18.9
Chemical Processing Department	343	1.2
Battelle-Northwest Laboratory	19 305	69.6

Total productive time in the Laboratory Maintenance Section for February was 22,032 hr of 23,228 potentially available. Distribution of time (in hours) was as follows:

A. Shop Work		1 896
B. Maintenance		7 471
1. Preventive Maintenance	1 754	
2. Emergency or Unscheduled Maintenance	1 518	
3. Normal Scheduled Maintenance	4 199	
C. R&D Assistance		12 665

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REACTOR AND MATERIALS TECHNOLOGY DEPARTMENT
MONTHLY REPORT - FEBRUARY 1965

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - O2 PROGRAM

1. Metallic Fuel Development

Alternate Uranium Composition. The development of improved metallic uranium fuel involves in- and ex-reactor studies of a number of additive and alloy compositions.

Four uranium compositions, fabricated to N-single tube geometry, were obtained for heat treatment and ex-reactor property evaluation. The compositions are: (1) standard 150 Fe-100 Si, (2) 350 Fe-800 Al, (3) 250 Fe-250 Si-250 Al, and (4) 250 Fe-350 Si. Fuel element lengths were beta and gamma heat treated; both treatments being the salt-salt process. Samples are being processed for microstructure examination, hot hardness and mechanical property testing, and recrystallization behavior. Specimens for recrystallization study were cold worked 22% by swaging and are being annealed. Hardness tests at 300 C were completed. Alloys (2) and (4) are hardest and essentially the same, while the standard composition (1) is the softest. Gamma treated material in each case is slightly harder.

Three Zr-2 clad rod coextrusions were made with uranium shot containing (1) 60 ppm carbon, (2) 550 ppm carbon, and (3) 500 ppm carbon and 1130 ppm phosphorus. Sections of this material were beta heat treated and the microstructure examined to characterize the type of dispersion of UC or UP present. No agglomeration of the compounds was noted by optical microscopy. Tensile properties were determined to 600 C in both the as-extruded and beta heat treated conditions. No marked difference in strength was found due to either the UC or UC+ UP dispersion (approximately 1.5 and 3.5 vol% dispersed phase, respectively). The tensile properties of these materials were comparable to normal uranium compositions in the cast, wrought, and beta heat treated condition except that the ductility at 500-600 C was reduced.

Studies are also in progress to evaluate uranium base alloys in terms of composition, fuel fabrication history, corrosion behavior, and irradiation performance. Alloys being studied include uranium with 2 wt% Zr, Mo, or Cb and up to 800 ppm Al or Si as well as uranium with Fe, Si, Al, P, or C additions of up to 800 ppm. Assembly of 40 irradiation test capsules, containing three fuel samples each, and five temperature monitoring capsules with one fuel sample each has been completed. The finished assemblies were shipped to B-Reactor and were charged on March 1. These fuel samples will be irradiated to approximately 3000 Mwd/ton at a volume average uranium temperature of 550 C.

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Comparative Swelling of Uranium Alloy Fuels. Samples of irradiated fuels from KSE-5 fuel elements with 0.31 at% B.U. (2700 Mwd/ton) have been postirradiation annealed and studied to obtain further information on the influence of alloy additives on swelling behavior. One sample contained 150 ppm Fe and 100 ppm Si (N-Reactor fuel), and another contained 400 ppm Fe and 800 ppm Al. Volume changes during annealing were as follows:

Fuel	Percent Volume Increase			
	As Irrad.	100 Hrs @ 625 C	100 Hrs @ 700 C	Total
With Fe & Si	2.2	1.1	1.7	4.9
With Fe & Al	1.2	0.7	-0.3	1.6

Postirradiation metallography showed both grain boundary tearing and "aligned porosity" in the Fe and Si fuel, while the Fe and Al fuel showed only slight grain boundary tearing and no aligned porosity. After the 625 C anneal, a coalescence of both types of porosity was evident in the Fe-Si additive fuel, but little significant change occurred in the Fe and Al fuel. After the second anneal (at 700 C), spherical porosity was observed in both the grains and grain boundaries of the Fe-Si fuel. This porosity is believed to result from coalescence of fission gases. In the Fe and Al fuel there was a small amount of porosity developed at grain boundaries. Both the metallographic observations and the density data indicate that fission gas mobility is inhibited to a greater extent in the Fe-Al fuel than in the Fe-Si additive fuel.

Evaluation of KER Irradiated Co-Product Drivers. A review of KER-3 operating data indicates that from the standpoint of stable prototypic operating conditions the co-producer test was one of the most successful. The loop outlet temperature was maintained nearly constant at 270 C with the ΔT maintained between 59 and 63 C. An average specific power of 105.8 kw/ft of uranium was maintained within less than 7%. Based upon "weasel" data for the charge, the maximum fuel element exposure was 2220 Mwd/ton, and the maximum specific power was 184.8 kw/ft of uranium. In the four central elements of the test, in which fuel swelling up to 3.3 vol% was observed, maximum fuel operating temperatures ranged from 500 C to 555 C.

Length measurements made on two of the co-producer element indicate that no significant length changes occurred during irradiation.

Target Element Development. A special study was started during February to determine the effects of a high temperature reactor excursion on Zr-2 clad Al-Li alloy target elements.

Examination of the three irradiated targets which were subjected to heating tests is under way. All three elements failed during heating, and the molten target alloy was released into the surrounding annulus. Although the examination is not complete, all evidence indicates that the elements failed due to the high internal pressures generated by the helium and tritium released from the target alloy during heating and melting.

A 1/4 to 3/8-inch-long split in the cladding of one element has been examined, and high ductility in the Zr-2 cladding at the point of failure is evident. It is estimated that at least one-half of the molten core was discharged through the split at the time of failure. Metallographic examination of sections taken at the cladding split and near the center of the element shows a compound layer of Zr and Al on the inside surface of the Zr-2 cladding. However, from 23 to 30 mils of the original 30-mil-thick cladding remained unreacted.

Zr-2 tubes, 0.565-inch diameter by 0.020-inch wall thickness, were pressure tested at 800 C and 930 C to determine the rupture strength of the Zr-2 target element cladding at high temperature. After reaching the test temperatures on two specimens, the pressure was increased until failure occurred. In three tests, the pressure was applied rapidly. All tubes showed extensive elongation of the Zr-2 at the point of failure, and all failures occurred in a 2-inch length between the end caps and supports. Failure occurred at 700 to 1000 psi for tests at 800 C and near 125 psi at 930 C. The low rupture pressures observed in these tests are well below the pressures which could be generated within the irradiated target elements during heating.

Tritium Evolution from Aluminum-Clad Target Elements. A pair of irradiated, aluminum-clad target elements -- one anodized and one nonanodized -- were heated in a vacuum furnace at 300 and 350 C to (a) determine tritium species adsorbed on the target surface, (b) measure the rate of tritium evolution through the cladding at 350 C, and (c) see if the anodized coating retarded tritium evolution. The target elements, irradiated in KEB loops, contained an aluminum-0.8 wt% enriched lithium (40 wt% Li^6) alloy. The alloy contained 15 volumes gas per volume metal. Irradiation time was 61.2 days (26.2 days at 40 C and 35 days at 280 C). Total tritium in the elements (corrected for tritium decay) was 145 cc, STP. No tritiated water appeared in either outgassing run (1 hr at 300 C). A trace of HT (0.04 mole%) appeared in the 300 C run on the anodized element, but was barely above the detection threshold (0.02%). Tritium evolution at 350 C appeared predominantly as HT. Differences in average tritium evolution rate from the two claddings were not significant.

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Protective Coatings for Zirconium. Flame sprayed coatings on Zircaloy-2 are being studied for their possible effect in protecting the alloy from attack by high temperature molten aluminum. Nearly all available materials which can be flame sprayed have been used in the first rapid survey of coatings. Some materials, though unlikely to be protective in themselves, may form intermetallic compounds with either the zirconium or the aluminum which may offer some degree of protection. The coatings thus far applied have been: Al_2O_3 , ZrO_2 , tungsten carbide, an aluminum-iron bronze, molybdenum, tantalum, aluminum, iron, copper, and nickel. SiO_2 and niobium will be applied if forms can be found suitable for flame spraying.

Refractory Target Materials. Small quantities of pure Li_2SiO_3 were prepared by reacting stoichiometric ratios of Li_2CO_3 and SiO_2 powders at 1100 and 1250 C. Respective densities were 83 and 98% theoretical.

Pressed and sintered pellets of LiAlO_2 (75% theoretical density) were crudely crushed and loaded into cylindrical tubes by vibrational compaction to 62.5% theoretical density.

Be-Zr Braze Wire for N-Fuels. Zr - 5 wt% Be brazing wire was successfully extruded to 0.025 and 0.035-inch diameter. The fabrication involved successive extrusions of Be-Zr alloy rods clad with stainless steel and vacuum sealed in mild steel billets. Final extrusions at 800 C resulted in 49 wires/rod of uniform diameter. The extrusion constants were 22 tons/in² for ingot breakdown and 28-32 tons/in² for reducing the composite billets.

N-Reactor Fuel Support Development. The mechanical test series on co-producer and tube-tube fuel inner supports continued. Fatigue data are being developed for 1/4-, 3/16-, and 1/8-inch-wide inner supports. Target fuel support assembly tests specimens are being fabricated.

2. Corrosion and Water Quality Studies

Boridizing Steel Chain. It has been suggested that mild steel chain, suitably treated to make it "black" to neutrons, be dropped into existing reactors for reactivity control during down-time.

In an effort to produce mild-steel chain black to neutrons, chain segments were pack-boridized both in commercial Ferroboron and B_4C powder. The best results were obtained with Ferroboron, using a nine-hour exposure at 1160 C. Chain so treated exhibited 90 to 100% of the neutron blackness of 1% boron-steel balls in reactor tests. The chain exhibited no high-boron phase on the surface of the steel which could be abraded off and lost in the reactor.

Improved Inhibitors for Reactor Coolant - IPD. Sodium nitrite-silicate mixtures are being evaluated as an alternative corrosion inhibitor to sodium dichromate in once-through cooled reactors. This

inhibitor system is more economical and contains no chromium which can produce Cr^{51} activity and Cr^{VI} compounds in the reactor effluent.

An in-reactor test in the 1706 KE single-pass facilities using 1 ppm NaNO_2 and 1 ppm Na_2SiO_3 at typical temperatures and flow rates was completed. After five months of testing, the coupons and fuel elements showed corrosion rates for aluminum similar to those in water inhibited by dichromate, while corrosion rates for carbon steel were higher. The aluminum coupons corroded in both types of water at a rate of $\sim 1 \times 10^{-3}$ inches/year; the corrosion rates of the aluminum jackets on the fuel elements were unmeasurable, indicating a good protective oxide film. The corrosion rate of carbon steel in the nitrite-silicate water was $\sim 3.0 \times 10^{-3}$ inches/year as compared to 0.4×10^{-3} inches/year in the dichromate-inhibited coolant.

Effect of O_2 in N-Reactor Graphite Cooling System. A one-month ex-reactor test to determine the corrosion effects of high oxygen concentrations (40 ppm) in water at N-Reactor graphite cooling system conditions has been completed. Severe pitting was found on the carbon steel and light pitting on some of the 400 series stainless steels. There was no stress-corrosion cracking of any of the alloys, including 17-4PH in the H900 condition stressed to yield. This alloy was found to be very susceptible to cracking in similar-quality coolant at 550 F, however.

Decontamination of N-Reactor Steam Generator Tubing. The N-Reactor primary system steam generators are scheduled to be decontaminated in April 1965, using the proposed two-step APSUL procedure (alkaline permanganate followed by inhibited sulfamic acid). Tests were performed during the month to evaluate potentially less-corrosive (HV) or simpler (one-step) procedures.

The HV procedure utilizes a solution of ammoniated hydrazine and versene (EDTA) solution. In the laboratory, a sample of the steam generator tubing was immersed in the HV solution for 22 hours at 95 C. The decontamination factor was so low (~ 2.0) that this reagent was eliminated from further consideration.

A one-step proprietary reagent (Wyandotte 5231-1) was evaluated using steam generator tubing. Decontamination factors were very good (~ 500), but the solution formed a precipitate after about 20 hours at the use temperature (100 C). This precipitate may limit its value for the steam generator decontamination.

Corrosion of Stressed Zircaloy-2 in Ammoniated Coolant. Examination of a failed Zr-2 sample holder irradiated in the MTR (KAPL 3029) showed both excessive hydriding near the fracture areas and localized subsurface oxidation. The cause of the hydriding has not been defined, but important parameters appear to be exposure temperature (< 475 F) and time, a possible effect of NH_4OH , surface damage from handling,

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and environmental factors such as neutron flux or stray currents present in the loop.

An in-reactor test was performed in the K-1 Loop in K-Reactor to determine if coolant conditions employed at N-Reactor would similarly attack cold-worked Zr-2. Samples with various combinations of peening, shearing, and drilling and in both etched and unetched conditions were exposed in- and out-of-flux for two months in pH 10 ammoniated coolant at temperatures below 475 F. Weight measurements and metallographic examination of the specimens failed to reveal any abnormal oxidation or hydriding.

Corrosion of Copper Alloys by Ammoniated Water. The ammoniated coolant adopted as standard for the N-Reactor primary system appears satisfactory from a general corrosion standpoint, but escaping ammoniated water or vapors may accelerate corrosion of copper alloys located internally or externally to the system. A series of tests is being run to evaluate conditions which might lead to corrosion in N-Reactor components.

Two valves were exposed in a loop, operating with pH 10 ammoniated water at 530 F and 1100 psi, for five and eight months, respectively. The packings were loosened to allow leaking ammoniated steam to impinge on the ASTM B132 and B138 manganese bronze bushings and flanges. Fluorescent penetrant and microscopic examination following the test did not reveal any ammonia stress-corrosion cracking. The Teflon valve packing was in excellent condition. The silver plating on the retaining rings was completely removed at all locations in contact with the ammoniated water.

Stressed, corrosion samples of seven additional brass alloys were placed in N-Reactor piping chambers to monitor ammonia stress-cracking corrosion. Although several ASTM B147 samples cracked during earlier tests, the nonrecurrence of this cracking under the present test indicated improved control of ammoniated water leakage from the primary system.

3. Gas-Atmosphere Studies

N-Reactor Graphite Burnout Monitoring. The first set of burnout monitors at N-Reactor has been examined. This set consisted of alternating samples of TSX and TSGBF graphite placed end-to-end down the entire length of the reactor core.

The monitoring facility, Process Tube Channel 1648, was uncooled and unvented. The only gas reaching the burnout monitors diffused through the process tube block from the 2 by 3-inch gas passages. Burnout of the moderator by O₂ oxidation would not be detected because the O₂ would be depleted before it could diffuse through the process tube

block. After the first set of monitors was discharged, this condition was corrected by drilling several 1-inch diameter holes at each end of the monitoring facility; these holes were located in the front and rear gas plenum chambers and will permit a flow of reactor gas through the monitoring channel in future irradiations.

A profile of the burnout rates from the first set of monitors showed a dome-shaped curve with a peak of approximately 0.2% per 1000 operating days located three feet upstream from the reactor-core centerline. It was found that the TSGBF material was approximately 1.5 times more reactive than TSX. The reactor was not operating at full power during the entire monitoring period from July 9, 1964 to January 10, 1965, and the temperature of the monitors was in the range from 500 to 625 C.

4. Process Tube Development

Postirradiation Examination of a C-Overbore Tube. A Zircaloy-2, C-overbore process tube was removed from C-Reactor after two years use. General Electric Company, IPD, requested that this tube be examined for irradiation effects. Two small pieces from the rear-face end of the tube and one three-foot piece from the center of the tube were sent to Radiometallurgy for this purpose. Borescope examinations indicated only superficial scratches on the interior of the tube. No effects of reactor exposure on structure or grain size were observed. Some small areas of localized hydriding were seen on the interior surface of the tube, but this could not be attributed to irradiation. Ring tensile tests indicated an increase of approximately 14% in ultimate strength due to irradiation. A BNWC report on the results of the postirradiation examination is being prepared.

Fracture Studies. Acoustical methods of detecting crack initiation and propagation are being investigated as a possible tool in controlling the fatigue growth of cracks in test specimens of pressure tubing. Initial work in selecting proper transducers and developing electronic circuitry is being conducted on simplified flat plate specimens to conserve the more scarce and expensive tube sections. The two most promising transducers to date appear to be a solid-state strain gage and an accelerometer. One of each of these was applied to a Zircaloy-2 flat plate specimen and an incrementally growing crack induced by stressing the specimen in a tensile machine. The strain gage consistently produced a stronger acoustical signal than did the accelerometer. The signals produced were quite clean cut sine waves with a frequency of approximately 10,000 cycles per second. It is not yet certain whether this frequency is a typical signature of the fracture of this particular material or whether it is strongly influenced by the geometry of the specimen.

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5. Thermal Hydraulic Studies

Predictions of Pressures in a K-Reactor Downcomer. An analysis was started to determine the pressures in the K-Reactor downcomer during a temperature increase of the reactor coolant. Such an event could occur during an unscheduled power surge and possibly cause damage due to overpressurization of the downcomer with boiling of the bulk coolant.

Four cases were considered to determine the rate of water level rise and the time at which boiling occurs. These are as follows:

Case	Temperature Rise	Initial Conditions		
		Near Riser °C	Far Riser °C	Flow gpm
1	0.5 C/sec for 20 sec	97	93	210,000
2	"	93	97	"
3	2 C/sec for 3 sec	97	93	"
4	"	93	97	"

For cases 1 and 3 (near side riser hot), a large amount of overflow occurs before surface boiling occurs, especially for the fast transient. This would cause a possible siphon effect with pressure oscillations of unknown magnitude. Even if the overflow did not cause a siphon effect, the hot flow from the near side stream would begin to flash and overflow part of its stream as flashing, two-phase water. This part of the transient is believed to have the largest unknown effect.

For cases 2 and 4 (far side riser hot), no boiling occurs in the approach section until saturated water from the far side stream arrives. Boiling also approximately coincides with the beginning of overflow. A significant siphon effect would not be expected for these cases. It is difficult to determine maximum pressure, since a two-phase mixture would be discharged through the overflow and possibly part of the downcomer elbow.

It is planned to continue the analysis to improve pressure estimates during boiling and to test further the assumptions used.

Hydraulic Tests for the Present Production Reactors. A series of tests were performed to determine the hydraulic characteristics of various modifications contemplated for the existing reactors.

Pressure drop tests were performed to determine the number of solid aluminum dummy pieces required to throttle tube flow for arch-rail fuel charges to that for a bridge-rail charge. Data were also

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gathered to determine the effects on Panellit pressure and tube flow when the solid elements were placed in various positions in the downstream dummy pattern. The results of the tests were reported in BNCC-16, "Pressure Drop Characteristics of Solid Aluminum Dummy Patterns in K-Reactor, Zircaloy-2 Process Tube."

The hydraulic characteristics of new overbore fuel elements in the C-overbore process tube were determined. These elements, identified as CVIAN I&E's, are hot die sized to 1.975-inch OD and 0.350-inch ID--a 10-mil increase in annular and hole dimensions over the standard CVIN elements. Results of the tests are given in BNCC-8, "Hydraulic Characteristics of CVIAN I&E Fuel in the C-Overbore Process Tube."

Flow tests were conducted in the laboratory to find the pressure differential and flow associated with two "mixed" charges of natural uranium, bismuth, and flux monitors in the C-overbore tube. The results are reported in BNCC-13.

6. Graphite Studies

K-Reactor Hot-Test-Hole Temperatures. Five thermocouple boats of two designs were placed in the hot test holes of the K Reactors. One design provides information on the temperature gradient across a typical boat loading and across a reactor-size bar. Four boats of the second design give temperatures at the center of a boat loading for the respective boat positions in the reactor.

Complete analysis of the thermocouple data has not been made; however, some generalizations may be made at this time.

- (1) All graphite samples are presently being irradiated in the temperature range 500 C to 700 C.
- (2) Each boat position of each test hole has a range of ± 50 C unless the reactor loading is changed appreciably.
- (3) All samples within a boat lie in a range of ± 7 C.
- (4) The temperature variation in a reactor-size bar will be in a range of ± 10 C.

N-Reactor Flux Monitoring. Samples for dimensional monitoring experiments in Rod Channel 74 at N-Reactor have been fabricated, assembled, and will be charged in the next two or three months.

Two, 24-inch long flux boats were charged into the rod channel. The flux-monitors consist of Fe, Ni, and Co-Al wires at nine positions in each boat. They will remain in the reactor for two or three months.

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N-Reactor Irradiations. Irradiations of samples of N-Reactor graphite in the GETR continue to progress satisfactorily. Two, fifth-generation capsules, H-4-5 and H-5-5, have successfully completed four and two cycles of operation, respectively. The maximum neutron exposure on the long-term samples is estimated to be approximately 7.5×10^{21} nvt, $E > 0.18$ Mev. Construction of the final, fifth-generation capsule, H-6-5, is completed, and the capsule will be installed in the GETR in early March.

During the GETR shutdown the first week in February, a test run was made of a revised core loading called the "checkerboard" arrangement. The test, consisting of 8-hour operation at 30 MW, depressed the H-4-5 and H-5-5 sample temperatures to 125 C below operating conditions. Consequently, the core arrangement was returned to the regular pattern for the startup of the present cycle. A fuel element was installed adjacent to the capsules, since the physics calculations showed a decrease in flux with the pressurized water loop removed. Upon startup, however, the H-4-5 and H-5-5 sample temperatures were on the maximum operating limits, restricting reactor power levels to less than 30 MW. Consequently, the reactor was shut down after 35 hours of operation, and the fuel element replaced with an isotope capsule and filler piece. Since subsequent startup, both sample temperatures and capsule operation are satisfactory.

B. WEAPONS - O3 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR High Power Density Core Elements. The maximum burnup of the four UO_2 -2 wt% PuO_2 high-power-density elements in PRTR is 2010 Mwd/ton_U. This burnup of the vibrationally-compacted elements was accumulated at a tube power of 1350 Kw (equivalent to a rod power of 19 Kw/ft), which produced maximum fuel temperatures of approximately 2600 C.

Flux peaking in the ends of these elements caused high fuel temperatures near the end caps. Localized fuel melting occurred near the bottom ends of rods of the first high-power-density elements (FE-6000) irradiated in PRTR at a tube power of 1350 Kw. High-power-density elements are now loaded with graded enrichment in the ends of the rods to avoid high fuel temperatures adjacent to the end caps. The top end is loaded with 1-cm of natural UO_2 , followed by 1-cm of UO_2 -1 wt% PuO_2 , followed by the UO_2 -2 wt% PuO_2 fuel material. The bottom end, which has more severe flux peaking, is loaded with 1-cm of natural UO_2 , followed by 3-cm of UO_2 -1 wt% PuO_2 , followed by the UO_2 -2 wt% PuO_2 fuel material. This loading scheme should cause maximum fuel temperatures to be generated in the midplane of the elements. Four short clusters containing graded enrichment were fabricated during the month.

Fuel Cycle Analysis. A flow sheet was developed for a one-ton per day plant which produces mixed-oxide fuel elements. When the plant design is complete, this information will be incorporated into the Fuel Element Fabrication Code to calculate unit fabrication costs. These costs will be used as a basis for the fuel cycle analysis of several different plutonium enrichment schemes.

Fuel Element Cost Studies. An internal accounting system was established to accumulate labor costs for preparing plutonium fuel elements. The system will be used to generate primary unit data for fuel fabrication cost analyses, incremental cost analyses, forecasting, estimating, planning, and scheduling.

Specification for Nuclear Uranium Dioxide Powder, Pneumatic Impaction Grade. Specifications were published (BNWC-17) to define the chemical and physical requirements of UO_2 powder for densification by pneumatic impaction. The high density material is intended for use in packed-particle nuclear fuel elements made by vibrational compaction, swaging, or similar processes.

Nondestructive Testing. Infrared testing of swaged and vibrationally compacted fuel elements is being evaluated for quality control and prediction of in-reactor performance. The technique, being developed

by the Physical Measurements Unit, involves analysis of the infrared radiation emitted by fuel elements when heated by various methods. The effects of variations in density, thermal conductivity, and continuity (i.e., cracks in cladding or fuel) on thermal diffusivity are indicated by a pulsed heat wave.

Vibrational Compaction Facilities. A small pneumatic vibrator was modified for possible use in compacting small fuel elements having simple configurations (e.g., rods and tubes). Preliminary strain-gage measurements showed lower stresses in EBWR cladding tubes, both empty and filled, than those induced by either the variable impact or the electrodynamic vibrators.

Determination of Bulk Densities of Crushed Impacted Fuel Materials. An instrument was developed for reproducibly determining pour and tap densities of irregularly-shaped particles usually produced by crushing pneumatically-impacted mixed oxides. The instrument, designed and built locally, provides fuel particle data useful for studying vibrational compaction behavior of this type of fuel material.

Test Element with Replaceable Rods - GEH-4-107. Metallographic examination of both a connecting end bolt and end cap of the replaceable-rod test element showed no fretting or chemical corrosion after 15.8 days of operation in the MTR. Photomicrographs of transverse cross-sections showed Zr-hydride concentration to be less than 100 ppm. One of the four rods of the element was remotely replaced, demonstrating the feasibility of this end cap design. A test cluster will be irradiated in PRTR.

Wire Enrichment. Fuel elements containing sintered UO_2 pellets enriched with 85% plutonium-zirconium wire are being prepared for irradiation in the PRTR. The wire and enough cored, UO_2 pellets to assemble the first full-length PRTR fuel rod were fabricated.

Fuel Element Rejuvenation. Mechanical problems caused an unsuccessful attempt to rejuvenate a fuel element with Pu alloy wire. The element, irradiated for one cycle in MTR, was returned to Radiomet for rejuvenation. A carbide bit broke while drilling a pellet in one rod, and galling occurred while parting the end cap in a second rod. A second element will be returned to Hanford for rejuvenation next month.

Heterogeneously Enriched Fuel Element. Failure of the heterogeneously enriched fuel element (GEH-4-108) was due to eccentric loading of the UO_2 -8.4 wt% PuO_2 in the annulus between the UO_2 pellets and the cladding. Examination of this rod indicated that failure after an exposure of 6×10^{18} fissions/cc was of the burnout type. The only fuel washout experienced was a small amount of loose powder exposed by the defect.

PRTR Salt Cycle Fuel Elements. Carbon tetrachloride extraction of "salt cycle" fuel disclosed an organic contaminant identified as Unichrome, a polyvinyl chloride latex used as a corrosion-inhibiting coating. Decomposition of this material during irradiation could cause internal hydriding of the Zircaloy-2 cladding. Equipment causing the contamination was discarded, and fuel rods scheduled for PRTR irradiation were recharged with fresh, salt-cycle fuel.

A single clad capsule containing salt cycle UO_2 -2 wt% PuO_2 was scheduled for irradiation in the MTR core. Maximum power generation of this capsule will be 2.2 Kw/inch and will operate with a peak surface heat flux of 192 watts/cm². End spacers were attached to a bottom section of a PRTR salt cycle rod to insure positioning in a standard S-basket.

PRTR High-Power-Density Core

Design Analysis Studies. Design data developed for the PRTR "chemical shim" soluble poison shim control were updated in accordance with recent reactor physics calculations and were summarized for criteria consideration.

For the proposed eight-month "batch cycle" operation of the PRTR, the desired linear decrease in boron-10 concentration over the eight-month period will require boron addition during the first six months and removal during the last two. Because of the extremely low flow rates required for continuous boron concentration adjustment (maximum rate of 0.09 gal/hr addition, 0.2 gal/hr removal), stepwise rather than continuous adjustment was recommended.

Use of the chemical shim system for scram recovery will require boron removal rates equivalent to about 10 mk/hr reactivity increase. Flow rates in the removal system will vary with boron concentration from about 2-18 gpm. To compensate for subsequent xenon burnout during startup of the reactor, boron injection flow rates up to 50 gpm may be required, actual rates varying with the time the reactor has been subcritical. These values are in substantial agreement with previous calculations.

A review of potential boron migration from the moderator to other reactor systems indicated a low likelihood of such migration except possibly for the reflector system. It was recommended that existing interties between the moderator and reflector systems be eliminated or modified to provide more positive isolation; that the fluid collected in tanks TK-A-1, -3, -4, and -6 should be routinely monitored for boron; that all borated D_2O be stored in unmistakably marked containers; and that investigation be made of the feasibility of slightly pressurizing the reactor reflector vessel to reduce possible in-reactor leakage from the moderator to the reflector.

Thermal Hydraulics Studies. Consideration was given to the potential hazards caused by a jacket failure of a PRTR fuel element operating at a power density where the core would be molten. Hazards arise from large volumes of steam formed should hot (approx. 6000 F), molten $\text{PuO}_2\text{-UO}_2$ ceramic contact the coolant. Local pressures from rapid steam formation might rupture the pressure tube or molten ceramic might contact the pressure tube wall and cause failure.

It was concluded that more information is needed to evaluate the problem. Specific information is needed on the nature of molten ceramic release into the coolant, the heat transfer coefficient between the ceramic and the coolant, the extent and heat of chemical reaction between the ceramic and water, and the heat of fusion of the ceramic fuel.

Corrosion and Water Quality Studies

PRTR Boron Shim Pilot Plant Corrosion Tests. Two tests were conducted in the Boron Shim Pilot Plant. The first test was conducted at 25-50 ppm boron at 140 F for two weeks. The test section coolant was in a static condition (no-flow) for almost the entire test period. Examination of the aluminum sleeve force-fitted against the inconel shim rod control section revealed severe pitting (up to 10 mils deep) on the surface contacting the shim rod. Although most pitting occurred at areas where the sleeve was scratched during the charging operation, some pitting was observed on other areas of the tube. Coupons from the loop indicated no excessive attack.

The second test was conducted under conditions similar to those described above except that a 10 ppm boron solution was used and the shim rod was mounted in a 6061 aluminum well similar to PRTR conditions. The well was bent slightly to insure contact with the shim rod. After 10 days of operation there was no apparent pitting; however, there were several areas where pitting attack had started on the aluminum webbing on the shim rod. Coupons from the loop were similar to those for the first test. The test is being continued under the same conditions for another two-week period.

Studies to determine equilibrium boron deposition values on 6063 aluminum and 304 stainless steel specimens exposed to various boric acid concentrations at 75 F and 140 F under dynamic conditions were completed. The maximum value obtained was 4×10^{-6} gm boron/cm² for 6063 aluminum exposed at 140 F to a 98 ppm boron solution. This value is not significant in terms of a permanent reactivity poisoning effect (~ 1 mk) at PRTR. Maximum deposition on specimens exposed under anticipated Phase I test conditions (140 F and 10 ppm B) was only 2×10^{-6} gm boron/cm². There was no appreciable effect of temperature, surface conditions, or specimen materials during these tests. Equilibrium deposits formed within two days during these tests.

Desorption tests were conducted in deionized water to measure release of boron from specimens exposed in the tests described above. Results indicated that $<1 \times 10^{-6}$ gm boron/cm² was released after four days of exposure to 140 F coolant. It was concluded that the boron deposits essentially irreversibly under these conditions.

Crevice Corrosion Tests on Simulated PRTR High-Power-Density Fuel Elements. An initial test to study potential crevice corrosion on PRTR fuel elements using Zircaloy-2-clad, cartridge heating units was terminated by cartridge failure after 70 days. The test was conducted in pH 10, lithiated water at 293 C and 1350 psig. Microscopic examination of the heating unit surface showed gray-white oxide deposits up to 0.005 inch thick in crevices associated with the "cluster" band. Surface heat flux of the test assembly was 550,000 BTU/hr-ft², which was high enough to initiate local (surface) boiling. Oxide formation is attributed to lithium concentration in the crevices during boiling. This initial test indicates that PRTR High-Power-Density fuel elements should be designed to avoid the concentration of additives by local boiling. Tests are continuing.

PRTR Decontamination Studies. The IRP Loop was decontaminated using the APSUL procedure recommended for decontaminating the PRTR and used successfully to decontaminate an ETR Loop. However, difficulties were encountered during the decontamination of the IRP Loop. Radio-active material, precipitated during the procedure, entrapped the activity. Since this precipitate collected at the filter used to simulate deadlegs and low flow areas, the decontamination factors were very low (<2).

The principal reason for the precipitation and low efficiency appears to be the decomposition on storage of one of the reagents, the proprietary inhibited sulfamic acid. Subsequent to the IRP decontamination, the reagent was tested in the laboratory and was found to be only partially soluble under the conditions used. Apparently some degradation had occurred during the storage (2 years) of this material. The IRP decontamination indicates that it is necessary to determine the effectiveness of decontamination chemicals immediately prior to use.

Corrosion of Aluminum in Boric Acid. The study of aluminum corrosion in dilute boric acid solutions has been required by the proposed use of these solutions in the PRTR moderator system.

The corrosion of type 6061 and 6063 aluminum alloys by dilute boric acid solutions in high-purity water and deuterium oxide at 50 C and 120 C was studied in small static systems. The results indicated a mild inhibiting effect of boric acid (up to 500 ppm B) in the early stages of corrosion. This inhibition disappeared after 800 hours for solutions up to 100 ppm boron in H₂O but was still apparent for 166 ppm B

in D₂O. Weight gains in the 250 ppm B solution in H₂O approached the pure H₂O curve after 4000 hours at 50 C.

Specimens of 6061 aluminum alloy were galvanically coupled to Inconel, Haynes 25, 304 stainless steel, Zircaloy-2 and 6061 aluminum by bolting together 1 x 2-inch coupons at one end. These couples were tested in various boric acid solutions at 50 C. The outer surfaces of the aluminum coupons developed uniform corrosion films of expected appearance. The surface of the aluminum coupons forming the crevice developed nonuniform films in the tighter portions of the crevice, and some of the aluminum specimens pitted substantially. Specimens exposed to a 100 ppm B static solution (air not excluded) coupled to Inconel and stainless steel pitted most rapidly. No pitting was observed on similar specimens exposed to a 100 ppm B refreshed solution (argon blanketed) in 2000 hours.

PRTR Pressure Tubes

Mechanical Testing. Two stress rupture tests, two burst tests, and one crack propagation test were completed. The results indicated an initial increase in stress rupture life with short exposure followed by some decrease in stress rupture life with prolonged exposure to PRTR conditions. The relationship of bursting strength to fast neutron exposure for cold worked material tested at 550 F and for annealed material tested at room temperature was extended by the two burst tests. The crack propagation stress in a PRTR tube was 48,800 psi which, because of the one-inch long crack in this specimen, lies very close to the curve expressed by the equation:

$$\text{Stress (crack length)}^{-0.625} = 49,200.$$

This is the twelfth test confirming the above relation that has been obtained using irradiated and unirradiated specimens.

Pressure Tube Monitoring. During the tube inspection run in January 1965, the lower end of the inspection probe was overheated, resulting in some charring of the illuminating lamp assembly and smoke blackening of the objective and relay lenses in the objective section of the borescope. These lenses were removed and cleaned and the lamp assembly repaired and the equipment restored to operating condition. Little or no radiation darkening of the lens elements (Corning 7940 purified fused silica and dense flint glasses) has occurred from service over the past nine months. For this period the accumulated gamma radiation dosage exceeds about 3×10^8 R.

Zr-Nb Pressure Tubing. About 75 feet of Zr-2 Nb alloy tubing was received and inspected on site by means of ultrasonic, Zyglo, and white light borescoping. This material was produced with either 30% or 60% cold work. About 40% of the tubing containing 30% cold work

exhibited inside surface cracking. The vendor will attempt to provide an additional 20 feet of tubing in this cold-worked temper.

Stress Analysis. A numerical method for analyzing thermal stresses in cylinders formed by a grid of triangular and rectangular rings has been programmed, and a rough draft report describing the computer program has been written. Latest additions to the program include a new method for representing a plane strain end condition and the representation of the rectangular rings in terms of four triangular rings where the temperatures and material properties of each triangle ring are specified. The representation of the rectangular rings in this form allows for a finer definition of material and temperature conditions without increasing the work of numerical evaluation. The program for the rectangular rings was checked for cases of uniform internal pressure, twisting, and uniform axial tension.

All the modifications to the stiffness coefficient due to boundaries other than free surfaces are done in one section of the program. Thus, boundary conditions such as plane strain, net axial forces, or radial surface forces can be made by simply changing this section of the program. The program will be applicable for fuel element end caps or transition joints between tubes.

Reactor Components Development

Ex-Reactor Studies. To confirm fretting corrosion results reported last month, the test using a UO_2 fuel element with 1/4-inch-wide by 1/2-inch-long centering feet was repeated. Examination of the liner of the vertical test section of TF-7 after exposure to externally excited vibration confirmed that fretting corrosion associated with a 500-hour test at 530 F is negligible. Penetration of the three abraded areas was less than 0.0005-inch which, on the basis of past experience, is considered insignificant depth. These results establish that fretting attack might be reduced to a negligible level by minimizing radial pad clearance and increasing pad area.

PRTR Tests. The sub-test of PRTR Test 101 dealing with the time history of the zirconium concentration in the coolant was performed on February 15, 1965, and the first preliminary data were received. Several samples will be re-analyzed to verify and to establish the concentration where reported to be greater than 10 ppb.

Vibration data were not acquired during this test; however, a 21% increase in vibratory motion was probable from an increase in steam pressure from 335 to 365 psig, based on past studies. The vibration monitoring system on selected inlet jumpers was inoperative during the month because of amplifier malfunction. The above test will be repeated after the vibration monitoring system is restored to service.

PRTR Mechanical Shim Rods. In a laboratory investigation of a Mark-II, first-generation shim rod, rod sticking was initiated by thorough degreasing and subsequent operation (reported previously). The sticking was traced to jamming in the ball nut assembly apparently caused by the presence of gray, flaky, foreign material. Spectrographic analysis and metallographic examination of the Haynes alloy 25 balls and ball vanes substantiated that this foreign material was corrosion product from the 6063 aluminum lead screw. Metallographic examinations at 1000X indicated the balls, ball vanes, and guide vanes to be free of any foreign or corrosive material or signs of corrosive attack.

Design Analysis

Fuel Re-use. All the economics cases were re-run with the revised QUICK code. Losses are correctly handled for both U-235 enriched and re-use cases. Sufficiently high "fuel service charges" (up to \$300/kg) were applied to the re-use cases to preclude these being limitations on the "indifference value" analyses. Although some extrapolations are still required, it now appears quite certain that the indifference value (of the fuel service charge) is insensitive to the fuel use charge.

A proposed outline of an experimental program for initial in-reactor demonstration of the fuel re-use concept (consecutive irradiation of fuel element in different reactor regimes) was prepared. The experiment is intended to yield data on enrichment in depleted UO_2 fuel elements from irradiation in fast reactor radial blanket. Since this irradiation will cause nonuniform plutonium buildup and distribution, information on the effect of succeeding irradiation of such non-uniformly enriched fuel element in a thermal reactor will be provided. These data will enable the estimating of the amount of fuel element segmenting or rod rearrangement desired in the cycle.

Plutonium Utilization. An investigation of plutonium burning in large thermal power reactors was initiated. The study is designed to reveal limitations related to the substitution of plutonium enrichment for uranium in thermal power reactors. The Yankee pressurized water reactor, Core I, was selected for a first case, and a burnup analysis was set up for the ZODIAK 2 multigroup, multiregion code.

PRTR Advanced Testing Study. Conversion of the PRTR to a two-region core is one of several methods of providing a versatile, multipurpose facility. Preliminary study of this concept was directed toward defining the central region core dimensions and operation conditions. Core power levels in the 40 Mw to 60 Mw range are under consideration. Using current PRTR, 19-rod cluster elements as driver fuel, the number of clusters, average tube power, and fuel rod surface temperatures were calculated as a function of core diameter with fuel-to-flow ratio as a parameter. Required coolant flow rates were determined for coolant outlet temperatures near 150 F.

2. Plutonium Ceramics Research

UO₂-PuO₂ Mixed Fuel Microscopy. Examination of a specimen of impacted UO₂-20 wt% PuO₂ fuel was conducted by replica electron microscopy. The fuel, previously examined by electron microprobe, consisted of PWR grade UO₂ plus PuO₂ (prepared from the oxalate), ball-milled 64 hours and impacted.

Microscopy of replicas of the polished surface revealed comparatively rough areas ($\sqrt{4}$ microns diameter) distributed randomly over the specimen. A similarly rough-appearing, continuous phase interconnected these areas, surrounding most of the particles ($\sqrt{3}$ microns) of the remaining smoother phase.

The appearance, distribution, and proportions of the smooth and rough phases suggest that the PuO₂ is present as $\sqrt{1/4}$ micron-diameter particles that form both the 4-micron agglomerates and the continuous phase surrounding each UO₂ particle. This interpretation is consistent with the 20-micron diameter electron microprobe scan, which showed occasionally high Pu concentrations superimposed on a more uniformly fluctuating Pu background. Detailed study of additional specimens is needed to confirm this interpretation.

Phase Studies. Calibration of the tensimetric apparatus with CeO₂ was completed. Several isothermal equilibrations at fixed oxygen pressures showed good agreement for the composition of CeO_x with data recently published in the literature. Results from 11 experiments yielded values of x agreeing to within 0.003 of the literature data (over the region $1.781 < x < 1.983$) with one extreme deviation of 0.005. The cryostat was used successfully in several of these experiments at temperatures between 0 and -25 C. Plots of $1/2 \log P_{O_2}$ against $1/T$ for constant compositions were in agreement with the literature cited within 0.03 log units. Equilibrium was not achieved during heating at 75 C/hr, and it appears that extremely slow heating rates or isothermal conditions are necessary to produce the equilibrium state in the Ce-O system.

Preparation of Pu-B Compounds. The difficulty of arc melting -325 mesh boron powder was circumvented by first "dynapacking" the powder in stainless steel cans. Arc melting of alpha plutonium and boron in chunk form was accomplished at the Plutonium Metallurgy Facility (231-Z) for compositions containing up to 96 at% B. Loss of boron was minimal. A series of heat treatment, x-ray analysis, metallographic, and electrical property studies of these compositions will be undertaken.

Plutonium Ceramics Metallography Facilities. Design of a special glove box facility for taking macrographs of plutonium-bearing particulate fuels is under way. Programs planned for this facility include (a) characterization of UO₂-PuO₂ fuels to determine the particle size

distribution in various particle size fractions and to subsequently correlate it with fuel rod fabrication and performance, and (b) characterization of large grained, arc fused PuO_2 . This facility is necessary to prevent gross contamination of the metallographic hood and to allow efficient handling of unmounted powder specimens. The importance of maintaining the metallographic facilities as contamination-free as possible was emphasized recently by the specimen decontamination required to permit off-site microprobe analysis of Pu-bearing and Pu-contaminated specimens in "cold" facilities.

High Temperature Dilatometer. The high temperature dilatometer was calibrated to 1500 C with a ThO_2 sample holder. It appears that the ThO_2 sample holders must be heat treated to stabilize them before use in the dilatometer. Therefore, the maximum useful temperature for this material has not yet been determined. A new incremental heating and cooling cycle was developed which allows thermal equilibrium to be attained every 100 C. This cycle increases the accuracy and stability of the measurements. The system was operated routinely, unattended, on a 24-hour cycle. However, an inadvertent loss of cooling water resulted in a system burnout. The dilatometer is now being installed in a glove box, and adequate safeguards are being incorporated to prevent a recurrence of the heating element burnout.

PuO_2 Microspheres. In order to evaluate the physical and irradiation properties of PuO_2 microspheres, a joint program has been initiated with Mound Laboratory. High density PuO_2 particles crushed, sieved, and fully characterized at BNW, are suitable feed material for the hooded plasma torch at Mound Laboratory. Mound will use this material to prepare microspheres in the 100-200 μ diameter range. The spheres will be returned to BNW for physical and irradiation testing.

Stainless Steel- PuO_2 Cermet Extrusions. The first plutonium-bearing cermets were successfully extruded. Two stainless steel-20 vol% PuO_2 cermet billets were extruded to 1/4-inch diameter pins about 30 inches long. Radiographs show the clad thickness to be fairly uniform--between 0.010 and 0.015 inch. Metallographic examination of longitudinal and transverse sections from one of the extrusions show a metallurgical bond between the clad and core.

Cermet Irradiation Test in P-7 Loop. Cermet fuel pins about 0.25 inch in diameter and 12 inches long are being fabricated for an irradiation test in the ETR P-7 loop. The cermets consist of 20 vol% PuO_2 dispersed in 304 stainless steel and are fabricated by extrusion, cold-pressing, or impaction. For comparison, fuel pins containing 20 vol% U-235 dispersed in stainless steel will also be tested.

Cold-pressed cermet pellets (82% TD) containing -65 to +325 mesh PuO_2 in stainless steel were prepared. Extruded fuel pins are also being

fabricated. An attempt is being made to fabricate integrally clad and capped fuel pins by pneumatic impaction. These pins will be about four inches long and can be welded end to end to a desired length.

Stainless Steel-PuO₂ Cermets. Capsule irradiations in the MTR of stainless steel-20 vol% PuO₂ cermets are continuing. Two capsules--one containing high density and the other low density cermets--were discharged from the reactor after an estimated 2% burnup of the plutonium. Calculated centerline temperature of the cermet was 1300 C. These capsules are being returned to BNW for postirradiation examination.

The stainless steel-20 vol% PuO₂ blended powders were pneumatically impacted to obtain high density (98+% TD) cermets. Low density (82% TD) cermets were made by cold-pressing the blended powders at 70 tsi.

Low density cermets heated to 900 C in vacuum for up to 810 hours showed no appreciable change (<0.6%) in density. High density cermets heated to the same temperature for 409 hours decreased 2.18% in density.

3. Ceramic (Uranium) Fuel Research

Electrical Properties of UO₂. The electrical resistivity of a single crystal specimen of UO₂ was measured to 1600 C in a dried argon atmosphere using the four-probe method. A change of slope of $\ln \rho$ plotted against $1/T$ was observed at 1040 C. The activation energy for conduction was calculated to be 0.47 ev between 700 and 1040 C. The activation energy between 1040 and 1600 C was 1.87 ev.

Cathodic Etching Behavior of UO₂. The cathodic etching behavior of both sintered and arc fused UO₂ is being studied. Differences between chemically-etched (9 parts 30% H₂O₂: 1 part H₂O₄) and cathodically-etched (6 hours at 2 kv, radiofrequency, in krypton gas at 10⁻⁴ torr), sintered UO₂ as observed by replication electron microscopy include (a) greater intraparticle detail and sharper intergranular contrast of the chemically-etched specimen, and (b) rounded pits produced in the grains of the cathodically-etched specimen. This latter effect may be the result of the sintered-nature of the UO₂. The effect is currently being sought in arc fused material.

UO₂ Melting Tests. A UO₂ fuel capsule was irradiated to test the hypothesis that once-molten UO₂ can solidify during continued irradiation as the effective thermal conductivity is improved through fuel structure changes. UO₂ pellets (1.36 inches OD) containing tungsten spheres (-100 to +200 mesh) in coaxial holes were irradiated

in a horizontal position in the KW "Snout" facility. After two hours of irradiation at a heat generation rate sufficient to cause initial melting to ~50% of the fuel radius, the capsule was rotated 180°, irradiated an additional four minutes, then discharged. Destructive postirradiation examination to determine the position of the tungsten spheres will reveal whether or not part or all of the molten UO₂ solidified during the two-hour irradiation.

Hot Stage Microscopy. A new hot stage specimen holder was constructed for reflection electron microscopy. The holder will permit use of the full 30° specimen tilt available in the microscope instead of the presently usable 20°. The new device features a boron-nitride sample block which can be interchanged through the vacuum lock without opening the column in a manner similar to specimen changes routinely accomplished in cold stage reflection operation. Tungsten electrical contacts are used to permit operation to well over 2000 C.

UO₂ Tube-in-Tube Element. Examination of a vibrationally-compacted, tubular UO₂ fuel element showed that failure occurred in the inner annulus of the inner tube. The defect (which occurred after a burnup of 5×10^{18} fissions/cc) was ~1/2 inches from one end, near the peak flux position of the reactor. The defect appeared to be a small angular crack with some degree of hydriding in the surrounding area.

Purification of Uranium Dioxide. The heat treating of sintered UO₂ wafers at 1750 C for 12 hours in flowing hydrogen lowered the fluoride content from 670 ppm to less than 2 ppm.

Materials and Information Exchange. As part of the exchange program with Argonne National Laboratory, approximately three kilograms of PuO₂ were prepared by the Nupac process with no unusual problems. The density obtained was 97% of theoretical, the highest density that has been achieved on PuO₂ or UO₂ fine-grained powders. Two and one-half kilograms of densified PuO₂ has been prepared for shipment to ANL, and the balance retained here for analysis and study. Characterization of the densified PuO₂ is under way.

Current requestors for single crystals of UO₂ included two sites in Japan and the National Magnet Laboratory at MIT.

The US-Japan Research Newsletter No. 3 was compiled.

4. Basic Swelling Studies

Irradiation Program. A general swelling capsule containing eight, 2.88 at% enriched, high-purity uranium specimens in several metallurgical states continues to operate successfully in-reactor at a controlled temperature of 375 C.

The third controlled pressure-temperature swelling capsule (P-4), scheduled for in-reactor operation at 1000 psi and 625 C, was assembled and shipped to a reactor for irradiation testing. It contains ten, high-purity uranium, U + Fe - Al and U + Fe - Si specimens with several histories of heat treatment.

Postirradiation Examination. Longitudinal sections of three specimens (two, high-purity uranium and one U + Fe - Al) recovered from capsule P-2 controlled at 450 C and 1000 psi were processed for optical and electron microscopy. The burnup in the specimens was 0.13 at%. A few scattered tears and pores were present in the high-purity uranium specimens, both of which were in the as-extruded condition prior to irradiation. The U + Fe - Al specimen, also in the as-extruded state, was completely free of tears and pores. "Banding" in the extrusion direction due to preferential irradiation growth was observed. Examination of unirradiated U + Fe - Al specimens revealed that recrystallization was not complete in the as-extruded condition. This accounts for the banding and for the macro striations which were observed in the U + Fe - Al specimen.

Supporting Activities. A document entitled "Physical Property Changes in Thorium Postirradiation Annealed to 950 C" was written and submitted for publication as a BNWL report. This paper summarizes work done on thorium during the past several years by Physical Metallurgy personnel.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Several nickel-base alloys are being studied to determine the effects of irradiation and environment upon their mechanical properties. During the past month specimens of Incoloy 800, Inconel 600, Inconel 625, Inconel 702, Inconel 718, Hastelloy N, Hastelloy X-280, and Hastelloy R-235 were tested at room temperature and 650 C. These specimens were irradiated at an average temperature of 740 C to an estimated exposure of 2×10^{20} nvt, fast. Data from the tests conducted at 650 C indicate a marked reduction in ductility as compared to the room-temperature tests. Similar specimens irradiated to a higher dosage were discharged from the reactor during the month.

Present and future reactor concepts utilize a liquid metal coolant. Components for a bench model of an irradiation capsule to be used to evaluate the effects of irradiation and a liquid metal coolant upon the mechanical properties of structural metals are being fabricated and assembled. The design of safety control circuits and a temperature control system was completed and reviewed by reactor personnel.

In-Reactor Measurements of Mechanical Properties

An in-reactor creep test on annealed 304 SS at 650 C and 10,000 psi is in progress. After 300 hours at test conditions it shows a creep rate of 6.8×10^{-6} /hr. The ex-reactor test at these conditions has not accumulated enough time to allow a creep rate to be determined.

The metallographic investigation being conducted on the irradiated 304 SS creep specimen is partially completed. Photomicrographs of transverse sections from one buttonhead and from near the fracture show no unusual features when compared to the unirradiated material. Longitudinal sections from the area near the fracture and the buttonheads will be made soon.

Irradiation Effects in Structural Materials

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

Tensile tests were conducted on BONUS transition joint specimens in the as-fabricated condition and after an exposure of approximately 725 hours in the out-of-reactor loop. These tests were performed at 550 F and at room temperature on specimens formed by both coextrusion and explosive welding processes. The integrity of both types of joints is very good since in the 14 tests conducted to date, all of the specimens have failed in the Zircaloy-2. No changes were noted in either strength or ductility after exposure in the out-of-reactor loop.

Several tensile tests have been conducted on stainless steels at temperatures from 400 to 750 C. The data from these tests were submitted for processing on the computer. Preliminary results indicate a continual decrease in ductility as a function of increasing test temperature up to approximately 600 C. Above this test temperature the ductility is gradually improved up to 750 C. These tests are being performed on stainless steels irradiated at 50, 290, 550, and 750 C. One of the objectives of this study is to determine the combinations of irradiation and test temperatures which produce minimum ductility. This information can be used to establish practical design limits for reactor components.

During the month, a dynamic solution of crack extension in the double cantilever beam (DCB) specimen was developed. This solution is now being used to aid in the proper design of the specimen. Basically, the solution relates the crack velocity to crack length subject to the effect of crack velocity on the energy absorption rate. The problem of developing a relationship between the energy absorption rate and the crack velocity so far has defied solution.

A computer program for processing fracture data from DCB specimens has been initiated and is now in the final stages of development. This program is being styled to fit the larger and more general REM program for inclusion at a later date.

Irradiation Effects in Nickel-Base Alloys

Twenty-four Hastelloy X-280 tensile and stress-to-rupture specimens are being irradiated in the G-7 position of the ETR at 540 F to a goal exposure of approximately 2×10^{20} nvt. These specimens are to test the effect of irradiation upon several experimental pre-irradiation treatments. These treatments induced general precipitation within the matrix, thus reducing the formation of embrittling deposits at the grain boundaries during irradiation. Additional Hastelloy X-280 specimens are being prepared in similar treatments for irradiation at 1300 F.

Ten Inconel X-750 tensile specimens, representing five experimental pre-irradiation treatments, have been fabricated and are being placed in quadrants for irradiation in the G-7 position of the ETR as soon as space becomes available. Other specimens are being prepared in similar treatments for irradiation at 1300 F.

Specimens of Inconel 600, a solution-hardenable alloy with similar chemistry to that of Inconel X-750, with the exception of lesser amounts of aluminum and titanium so that precipitation-hardening does not occur, are being prepared for irradiation at 540 F and 1300 F. These will undergo pre-irradiation treatments similar to those of Inconel X-750. It is felt that a better understanding of the nature of radiation damage in complex high temperature alloys may result from a comparison of the test data for the two alloys.

Damage Mechanisms

The objective of this program is to determine how interactions between irradiation-induced defects and dislocations modify plastic deformation mechanisms in a metal. The role of interstitial impurities in alpha iron is currently being investigated.

During this period several single crystals of Ferrovac "E" iron were grown by a strain anneal method, were indexed, and were polished preparatory to deformation and slip trace analysis. The largest of these crystals was roughly 50 mm x 5 mm x 1 mm.

A hardness recovery curve determined from an isochronal annealing experiment over the range 0 - 900 C showed two major recovery processes in Ferrovac "E" iron. The first was a broad peak over the range 270 C - 470 C which represented 37% of the hardness recovery. The second peak, which was much narrower, was centered at 680 C and represented 59% of the hardness recovery. Similar experiments on irradiated specimens of Ferrovac "E" and BMI zone-refined iron are in progress.

A study of machine effects during strain rate changes was conducted to ascertain the cause of inhomogeneous yielding in specimens of Ferrovac "E" iron tested at room temperature. Yield point effects in Ferrovac "E", which shows a yield point, and BMI which does not, were studied by simultaneously recording load-time and strain-time curves during strain rate changes, affected by the Instron automatic speed changer. These experiments indicated a slight delay in strain-rate transition, but the rate steadily increased to the new rate rather than the decrease and sudden surge observed by Bolling, et al. No differences were noted in the curves for Ferrovac "E" and BMI iron.

Similar experiments on initial yield points, where the specimen is loaded at a constant crosshead rate, showed a large increase in rate following yielding. This is caused by the elastic springback of the tensile machine following yielding at the upper yield point. The rate returns to that imposed by the crosshead after the lower yield point is reached.

ATR Gas Loop Studies

Model Loop Studies. An extensive test run was made with the model gas loop to determine the characteristics of foil insulation that contained one prototypical ATR joint, and one open joint with a 1/16-inch longitudinal gap between the stacks of foil. No attempt was made to prevent gas flow through the open joint. A preliminary analysis of the data produced a conductivity of 0.65 BTU/hr-ft-F at a mass flow per unit cross sectional area of pipe of 22,900 lg/hr-ft² and a mean temperature of 1200 F. This mass flow compares closely with prototypical flow of 5,000 lb/hr in a 6-inch pipe which is 24,500 lb/hr-ft². The tests showed a definite dependence on flow, which is in direct contrast with the previous runs which indicated no affect of flow. After the next series of tests an attempt will be made to modify the open joint in order to determine if this joint is affecting the conductivity of the insulation three feet downstream. The open joint created a hot spot on the test section that was several hundred degrees hotter than areas covered by the insulation.

During the test run the loop operated with a small bypass cleanup system that consisted of a charcoal and a molecular sieve cold trap. Several days of operation were completed with less than 1 ppm of water, and with hydrogen being the only other major impurity. Gas temperatures out of the heater ranged up to 1700 F during this period. A catalytic copper oxide bed is being installed in the bypass cleanup system to aid in the removal of the hydrogen.

The test section is being modified to incorporate a cooling water jacket. This modification will permit testing of the foil insulation with high temperature differentials and will allow testing under ATR conditions. The gas-bearing compressor has completed 270 hours of

operation over a wide range in pressure, temperature, and speed. During this operation the compressor has been started about 80 times. No problems have developed with the compressor.

Model Loop Heater Design. Detail drawings of the second generation helium gas heater for the model gas loop were completed. Orders were placed for the electrical control system and 60% of the material for major components of the heater. An analysis is being made to determine what piping changes will be necessary in the loop to replace the existing heater with the new model.

ATR Loop Corrosion Studies. Microbalance studies of niobium exposed to oxygen at low pressures was started. As-received niobium was oxidized in 10 to 12 μ replenished oxygen atmosphere. Linear rate constants calculated for these tests are: 1.96×10^{-4} mg/cm²sec (50 to 1200 min) at 1000 C, 2.44×10^{-4} mg/cm²sec (initial rate), 1.68×10^{-4} mg/cm²sec (175 to 1100 min) at 900 C, and 2.38×10^{-4} mg/cm²sec (initial rate), 7.73×10^{-5} mg/cm²sec (50 to 300 min) at 800 C. The rate of oxidation is relatively fast, the initial rate apparently controlled by the diffusion of oxygen to the sample. The sample sits in a static leg removed from the point of pressure measurement, oxygen leak, and system pumps in that order. The actual oxygen pressure around the sample is not known. System modifications will be required for a quantitative determination of the surface reaction rate as a function of oxygen pressure.

Dynamic Materials Testing Apparatus. A 10-day run with a niobium alloy was completed on February 1, 1965. Shortly before completion of the run, the guard heater failed. Upon inspection of the guard heater, it was found that grain growth of Pt-1% Rh heating wire had taken place and the wire had failed. This winding had about 800 hours of service at an operating temperature of approximately 2050 F. Wire temperature probably is several hundred degrees F above this temperature. The guard heater has been replaced, and installation of the new recorder-controller has been started.

The Cb-1 Zr sample was exposed to untreated Grade A helium in the loop for 285 hours at 2050 F and 350 psi. Gas velocities are 525 fps. It was found that the sample formed a tenacious oxide film, with no evidence of the spallation found during oxidation of Co or Ni-base superalloys. The sample showed a weight gain of 2 mg/cm², and an average increase in the diameter (normally 0.25 inch) of 4 mils.

Outgassing of Ceramics. Specimens of Coors AD-94 (94% Al₂O₃), AP-100 (100% Al₂O₃), and AHP-99 (99% Al₂O₃, machinable) were supplied by Materials Engineering for outgassing studies. These materials are candidates for model gas loop heater insulation. The AP-100 material was not supplied in samples massive enough for a meaningful test. The AD-94 and AHP-99 material was broken into pie-shaped pieces

weighing about 100 gm each. These pieces were stacked three deep, placed on a grate, and outgassed in vacuo for two hours at 400 C and 600 C. The AHP-99 material lost an average of 0.013 wt% during the 400 C test, and 0.020 wt% during the 600 C test. The AD-94 material lost 0.004 wt% during each test. The weight lost during either test, from either material, was gained back rapidly during exposure to lab air (~80% in 30 min), presumably by adsorption.

Helium Diffusion Cell. The quartz helium diffusion cell was tested as a source of ultra-pure helium. This helium will be used as gas chromatograph carrier gas and ATR reference helium for zeroing the instrument. Helium delivered from the cell was invariably contaminated with 10-20 ppm H₂. It appears the source of the H₂ is internal contamination. Discussions with another user of this same device at Bell Telephone Labs indicated the cell is capable of producing ultra-pure helium, suggesting our unit is defective. The cell was returned for repair or replacement.

6. Nuclear Graphite Studies

Gamma Irradiation Facility. An additional 86,000 curies of cobalt-60 were added to the gamma irradiation facility to bring the total to 185,000 curies. Ceric sulfate dosimetry measurements were completed in a 2-inch diameter tube which extended from the source to three feet above the surface of the water. With all of the cobalt positioned around a single, 2-inch diameter tube, the dose rate was 1.65×10^7 r/hr. The variation in dose rate over an 8-inch length was less than 5%. When the cobalt was placed around two 2-inch diameter tubes, the dose rate was 1.23×10^7 r/hr.

One tube is currently used for irradiations near room temperature, and irradiations at temperatures up to 500 C are conducted in the second tube. This facility is quite versatile and could be easily modified for irradiations at other conditions.

EGCR Graphite Irradiation. The report, "The H-3 Irradiation Experiment: Irradiation of EGCR Graphite--Interim Report No. 2," HW-71500B, was issued. This report presents information from the first seven capsules in the series of EGCR graphite irradiations.

The ninth capsule, H-3-9, in the series successfully completed four cycles of operation in the GETR and was removed from the reactor on February 1. Maximum neutron exposure is estimated to be approximately 1.35×10^{22} nvt, $E > 0.18$ Mev. Upon disassembly of the capsule, it was found that the four samples in Position 1 were frozen into the central graphite collar and cooling ring, and during removal the ends were accidentally broken off. Consequently, the hole-to-hole length measurements will be used to determine the length changes for these samples. All other samples were recovered in excellent condition.

Construction of H-3-10 is under way concurrently. The four samples from H-3-9 position will be modified by rounding off the broken ends and remeasuring. Since these samples provide data at approximately 325 C, they will be used in H-3-10. The central graphite collar will also be revised by splitting it into sections to prevent expansion from freezing the collar around the samples. A new special sample holder will hold the samples in place. The H-3-10 capsule will include 19 samples from H-3-9, three from H-3-8, and two from H-3-7, with four samples having been in all nine previous capsules. H-3-10 will be installed in the GETR the first week in March.

During the past GETR shutdown and just prior to the removal of H-3-9, a flux run was made with the capsule located in a new position, E8. The purpose of the run was to (1) measure fast fluxes by means of nickel and aluminum wires in several reactor positions and correlate these with physics calculations, and (2) determine sample temperatures in H-3-9 at several power levels to enable extrapolations to higher power levels contemplated for the future. Flux wires are still being measured; however, sample temperatures indicate that reactor operation at approximately 45 MW would be possible with the capsule temperatures remaining at the present operating levels. Consequently, additional determinations are being made by the GETR physics and engineering groups to determine in detail the feasibility of increasing reactor power levels.

High-Temperature Graphite Irradiations. The second capsule, GEH-13-11, of the high-temperature graphite irradiation series is performing satisfactorily in the K7 position of the ETR. The capsule contains 29 samples being re-irradiated plus five new samples. Maximum sample temperatures are approximately 1150 C, and all thermocouples are functioning properly.

Irradiation of Experimental Graphites. Samples from the Speer contract (DDR-164) were discharged after an initial exposure of 1650 Mwd/At_K. The samples were irradiated in the 3C test hole at KE Reactor at a temperature of 600 ± 100 C.

Two series of extruded samples were tested. The first series was prepared from one formulation of raw materials and the dimensional change during processing was controlled by varying the rate of temperature rise during graphitization. The second series, prepared using a graphitized filler and a pitch binder, were heat-treated to various final temperatures ranging from 700 to 2900 C.

An effect of graphitization rate and binder graphitization temperature was seen for both transverse and longitudinal orientations. At low-binder temperatures, the transverse orientation experienced the larger contraction. However, higher exposures were required to determine long-term effects.

The graphitization rate appears to have no effect on the radiation-induced changes to the coefficient of thermal expansion (CTE) for either orientation. Irradiation increased the CTE about 45% in transverse samples and about 24% in parallel samples.

For transverse samples in the binder series, the CTE's increased 56 and 44% for binder temperatures of 700 and 2900 C, respectively. For parallel samples, the changes were +35 and -5% for the same binder temperatures.

Annealing of Irradiated Carbon. Four carbon samples containing equal amounts of lampblack filler and pitch binder and heat treated to 1400 C were irradiated at 600 C to 2.5×10^{20} nvt and subsequently annealed. The neutron exposure caused the carbons to shrink 0.9% and the bulk CTE to increase 8.7%, but the CTE of the crystallite c spacing and the value of the c spacing remained the same. These results are most easily explained if the material between the crystallites had densified. The driving force for densification could have come from the internal stresses developed during cooling from the 1400 C heat-treatment temperature.

Subsequent annealing to 1100 C made the carbons shrink an additional 0.02% without any noticeable change in the other measured properties. Further heating to 1375 C caused the carbons to expand 0.02% with a corresponding decrease in the bulk CTE to a value 3.8% greater than the unirradiated carbons. During this further heating, the c spacing decreased 0.3%.

These observations are explainable if the small microcracks along the c planes between the subgrains had grown. Microcrack generation would be expected as a result of the decrease in c spacing. Moreover, the generation of microcracks which could absorb the crystallite thermal expansion would explain the decrease in bulk CTE.

Dosimetry for ETR Boronated Graphite Irradiations. A mockup of the shielded boronated-graphite capsule, GEH-23-8, was irradiated at low power in the ETR M6 position to measure the cobalt-cadmium ratio for the shielded facility and to determine axial and azimuthal flux distribution.

Three sets of Co, Cd-covered Co, and Fe foils were spaced 120° around the periphery of the capsule. These wires extended the full 24-inch length of the capsule. A set of foils was also placed in the center of the capsule at reactor centerline.

Analysis of the foil activities (A) gave a Cd ratio ($\frac{A_{\text{bare}}}{A_{\text{Cd-covered}}}$) of 1.71 for the sample position and a spread of 1.68 to 1.88 for the peripheral positions.

The effect of these measurements is to decrease the thermal-neutron exposure to 40% of the original estimates for the previously reported, shielded, boronated-graphite irradiations.

7. Metallic Fuel Development

Irradiation of Thorium-Uranium Fuel Elements. The irradiation of three, tubular, Zircaloy-2 clad, thorium-2.5 wt% uranium-1.0 wt% zirconium, fuel elements continued successfully in the ETR-P7 loop. The fuel elements are currently operating at a maximum temperature of only 385 C due to a decreased neutron flux in the area of the ETR-P7 loop. The surface heat flux is 30 cal/sec-cm² (4.0×10^5 BTU/hr-ft²), and the integrated exposure is currently 3.7×10^{20} fissions/cm³ (10,800 Mwd/ton). No swelling beyond that expected from solid state fission products was observed.

Nb-1 Zr Clad, Th-2.5 U-10 Zr Alloy. Nb-1 wt% Zr alloy has potential application for high temperature service in liquid alkali metal environments due to its corrosion resistance and high temperature strength. The high temperature performance of metallic thorium base fuel makes it potentially useful for liquid-metal-cooled reactor fuel applications. This material combination has not been employed previously, and it was felt desirable to determine the fabrication and bonding characteristics. One rod coextrusion was made producing a 0.565-inch diameter rod with 0.026-inch Nb-1 Zr cladding over a Th-2.5 U-10 Zr core alloy. Extrusion conditions were 800 C - 2-1/4-hour billet preheat, Cu-10 Ni extrusion can with Aquadag lubrication, at a reduction of 13.3:1 in 500 C container and die. K_{max} was 23.7 tsi. Sections of the coextrusion showed excellent clad thickness uniformity ($\sim \pm 0.0005$ inch) and complete metallurgical bonding. The Cu-10 Ni extrusion cladding was easily removed with HNO₃-H₂O without attack of either the Nb-1 Zr or thorium alloy core material.

Additional sections were heated in vacuum for two hours at 900, 1000, and 1100 C to check diffusion, bond failure due to thermal cycling, and recrystallization of the cladding. Additional bond zone growth was observed with no bond failure. Components of the diffusion zone are being determined utilizing the electron beam microprobe. The Nb-1 Zr cladding appears partially recrystallized to a fine-grained, equiaxed structure after heating at 1100 C. These results indicate that fabrication of other geometries (tubes or honeycomb shapes) is feasible.

8. Critical Flow of Steam-Water Mixtures

A report entitled "The Critical Discharge of Steam-Water Mixtures from Pipe Elbows and Tees" was submitted for publication. The report covers the study of the critical discharge of two-phase, steam-water mixtures from a piping system when the system is opened to the atmosphere adjacent to a tee or elbow. Two phenomena uniquely determine the maximum

rate at which a two-phase mixture can be discharged from such a system: (1) the critical flow phenomenon and (2) the two-phase, pressure-drop phenomenon. Experiments were performed at moderate temperatures and pressures on 1/2-inch commercial piping and welding fittings to study the effect of the presence of the fitting on these phenomena. In these experiments the range of steam-water total enthalpies considered was between 500 and 1000 Btu/lb. Flow rates ranged from 200 to 800 lb/sec-ft², and measured critical pressures were between 23 and 87 psia. The results indicated that the elbow at the discharge of the flow passage did not significantly influence the critical flow rate. The tee in this location reduced the critical discharge rates from 10% at enthalpies of 900 and 1000 Btu/lb up to 24% at 500 Btu/lb. This indicates that the tee immediately upstream of the location of the flow-choking acts to increase the degree of homogenization of the two-phase mixture. The data gathered in this investigation were correlated by two methods. In the first method the critical flow phenomenon and the two-phase pressure drop across the fitting were considered as one effect, and the critical discharge rate was related to the flow enthalpy and a "pseudo-critical" pressure upstream of the fitting.

In the second correlation the critical flow model of Fauske was combined with the pressure loss information obtained in this investigation. These combined data were referred to the momentum exchange pressure drop model of Levy and the empirical relation of Thom to obtain a general pressure loss correlation for pipe elbows and tees during choked flow. A comparison between the resulting pressure loss correlation with one proposed by Fitzsimmons for normal (nonchoked) flow in similar fittings demonstrated a surprising consistency.

9. Upstream Boiling Burnout

A survey was made of the results obtained to date in the investigation of conditions which can cause boiling burnout to occur at positions upstream of the exit of electrically heated flow channels. The following conclusions were made:

1. In tests with a 12-foot, 0.44-inch ID test section at 1000 and 1500 psig, upstream burnout was found to occur as a reproducible phenomenon with inlet enthalpies near or above saturation and at mass velocities of 5×10^6 lb/(hr)(ft²) and higher.
2. Upstream burnout was also encountered in low-pressure tests (90 to 150 psi). Fifteen burnout points, all upstream, were found with a test section composed of 0.1-inch bands of high heat flux separated by 0.9-inch bands of lower heat flux. Upstream boiling burnout was also encountered with a uniformly heated test section, but the burnout point was usually

accompanied by large pressure oscillations. When the flow apparatus was modified to decrease the pressure oscillations, the burnout points always occurred downstream.

3. Both the high-pressure and the low-pressure data indicate that the burnout heat flux must be a function of some quantity in addition to enthalpy which varies along the length of the test section. The chief significance of the data obtained to date lies in this indication that burnout depends on some factor besides those usually taken into account.

10. Advanced Reactor Concept Studies

Nitride Fuel Cycle Studies. Additional multigroup calculations were completed for the 1000 Mwe nitride fueled fast reactor study. Two cases were investigated to determine what changes in neutron energy group structure and cross sections are desirable before refining the core design further. The 16-group YOM cross section set, used initially, lacks low energy detail and does not contain the more recent high energy plutonium data. A comparison of the neutron spectrum for the carbide fueled 1000 MWe Combustion Engineering study obtained with an improved 18-group set shows a substantially softer spectrum than that predicted by the YOM set. Further, the sodium coefficient calculated for the nitride core appears to be strongly negative, a questionable result for a core of this size. It is concluded that the 16-group YOM set is not adequate to continue this study and that alterations in both group structure and nuclear data are necessary.

Study of the processes for preparation of nitride fuels revealed that they have been prepared to date only by small-scale laboratory methods involving reaction of the metal (uranium or plutonium) or its hydride with nitrogen. Alternate methods which appear feasible but require development include a carbon-nitrogen reduction of the oxide, or reacting the chlorides with nitrogen or ammonia. As one phase of the nitride fuel cycle study, the feasibility of potential nitride conversion methods will be analyzed and conversion costs estimated. The establishment of a reasonable conversion cost for nitride fuels may well be the key element in establishing their economic position relative to the carbides and oxides.

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Plutonium and U-233 Fueling of a Fast Compact Reactor. Survey calculations on Pu- and U-233-fueled, compact reactors, including the effect of tantalum on core size, were completed. A linear extrapolation of size based on volume fraction of Ta appears satisfactory. Three cases, corresponding to the three fuels at 0.95 total pressure ratio, were selected for core zoning. They were zoned using the SWISS CHEESE computer code and are being checked for criticality.

Definition of the minimum possible size core based upon fuel temperature limitations and reduction of tungsten content was not possible. The calculated thermal conductivity of 30 vol% tungsten is still high enough at the design point to preclude fuel temperatures of more than 200 F higher than the tube wall.

The core zoning program, SWISS CHEESE II, was completed and checked against the base design, and an outline of the final report was drafted.

Fast Flux Test Reactor Studies. Information on testing capabilities of the 200 Mwt core was developed and is presented below:

	<u>Open Loops</u>	<u>Closed Loops</u>	<u>Reflector Test Holes</u>
Coolant	Na	Na	Optional
No. of Test Facilities	8 (33)	4 (12)	6 to 20
Power per Loop, Mwt	5-10	5-20	($\phi=2 \times 10^{15}$)
Inlet Temperature °F	450 (750)*	600-1000	Optional
Outlet Temperature °F	800 (1100)*	900-1400	Optional
Net Velocity ft/sec	25-60	20-50	--
Test Specimen "Diameter," In.	3.3 (6)**	3.3 (6)**	1 to 6
Fueled Length, Inches	35	35	--
Total Length, Inches	70	70	70++
(including blankets & plenum)			

*Nuclear preheat; multipass or series flow.

**1 or 2 peripheral positions.

A fast reactor core design-analysis program is being written jointly with Systems Research for use on the Beckman Ease 2133 analog computer. The program is intended to permit rapid parametric surveys of a wide range of design variables. The initial scope of the program will include coolant temperatures, flow rate, and pressure drop; geometry of fuel pin, fuel element, and flow tube; critical mass and flux as functions of core size and composition; temperatures and temperature gradients in film, clad, and fuel; and burnup limits as functions of fuel temperatures and power distributions.

Although aimed directly at optimizing the FFTR core, the program will be generally adaptable to other fast reactors.

Computer studies to determine the decay heat generation in candidate driver and test fuels for the fast test reactor are under way. The information will be important to the design of fuel handling and fuel examination facilities.

11. Nuclear Rocket Fuels Studies

Studies conducted for AEC-SNPO on tungsten-UO₂ fuel element development and on fueled graphite problems are reported via a separate distribution list.

D. DIVISION OF RESEARCH - 05 PROGRAM

1. Radiation Effects on Metals

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

Radiation damage in rhenium, a cph metal, is being studied in a manner analogous to that of molybdenum, a bcc metal. Rhenium foils have been irradiated to an exposure of 7×10^{18} nvt and examined by transmission electron microscopy. Defect structures which could be attributed to neutron irradiation were not observed. This is in marked contrast to observations made on fcc and bcc metals where damage in the form of spots and loops is observed after equivalent exposures. The absence of observable clusters in irradiated rhenium may be explained in several ways. Clusters may exist but may be too small to be resolved by the electron microscope. It is also possible that clustering has not occurred because point defects are not mobile at the irradiation temperature. The irradiated foils are currently being annealed in an effort to promote growth of clusters.

Single crystal specimens of molybdenum which have been used for length-change-lattice parameter measurements have been cut into shorter sections on which crystal faces have been ground and polished. These faces are parallel to (110) and (100) planes. The intensity of diffracted x-rays from these faces will, it is hoped, provide an indirect means for observing the movement of isolated defects during low-temperature annealing. Measurements currently are being made on samples annealed four hours at 200 C, and differences in intensity have been observed.

Studies on the changes in the length of irradiated molybdenum due to annealing have been extended to 220 C. The results, however, were anomalous. An initial decrease in length followed by an abrupt increase was observed. The possibility exists that the oil bath used for annealing may be responsible for contamination of the molybdenum. In subsequent experiments the crystal rods will be sealed in glass ampules before immersion in the oil bath.

A paper, "Line Broadening in Irradiated Molybdenum," was prepared for presentation at a meeting of the X-ray Diffraction Group of the Fuel Element Design Committee convening in Richland in February.

The equations for the two-beam dynamical theories of electron diffraction through a foil were programmed and evaluated for foils of various thickness and with various values for the elastic scattering coefficients. Parametric studies have been made for various foil thicknesses with the electron intensity transmitted through the foil plotted as a function of the deviation from the Bragg reflection condition. Several attempts have been made to correlate experimental and theoretical intensities in order to determine the foil thickness. Kikuchi lines have been used to measure the deviation from the Bragg reflection condition. Difficulty was encountered in obtaining representative two-beam diffraction patterns and verifying predicted foil thicknesses.

Tests on the rate- and temperature-dependence of the flow stress in irradiated polycrystalline molybdenum continue. The drop in load occurring upon initial yielding has been correlated with exposure and with postirradiation annealing treatment based on current rate theory.

2. Plutonium Physical Metallurgy

The objective of this program is to derive fundamental information relative to (1) the kinetics and mechanics of the phase transformation in plutonium, and (2) the mechanisms by which monoclinic plutonium deforms.

The influence of plastic deformation of a stable plutonium phase on subsequent phase transformation has been investigated and related to the mechanism of the phase transformation. Previously it was reported that plastic deformation of the beta phase retards the beta to alpha transformation and that plastic deformation of the gamma phase accelerates the gamma to beta transformation. Currently it was shown that plastic deformation of the alpha phase by compression, e.g., 30% reduction in thickness, accelerates the alpha to beta transformation and that plastic deformation of the beta phase accelerates the beta to gamma transformation. The accelerating effect of plastic deformation

of the stable phase on its subsequent transformation is a characteristic of diffusional transformations. The retarding effect of plastic deformation of the stable phase on its subsequent transformation is a characteristic of diffusionless transformations.

Studies on the effect of applied uniaxial tensile stresses on the low-temperature, allotropic transformations of plutonium were performed and the data were compared to similar compression studies. The results were related to the phase transformations of other metal and alloy systems. Applied stresses during beta to alpha transformation produce a crystallographic texture and a microstructural shape texture. The alpha grains are elongated in a direction perpendicular to the applied tensile stresses. They are parallel to the applied compressive stresses. The elongated grains are aligned in bands which are perpendicular to the direction of applied compressive stresses and parallel to the direction of the applied tensile stresses. The (010) plane is parallel to the direction of the applied tensile stress and perpendicular to the direction of the applied compressive stress. The data also indicate that texture can develop as a result of alpha to beta, beta to gamma, and gamma to beta transformations under either applied uniaxial tensile and compressive stresses.

Some phase transformation creep occurs during the transformations when subjected to an external stress. Phase transformation creep is the strain during transformation that cannot be attributed to either normal creep or anisotropic volumetric changes. More alpha to beta transformation creep occurs under a tensile stress than under an equal compressional stress. Little or no beta to alpha transformation creep occurs under compression, but some beta to alpha transformation creep does occur under tension.

The effect of applied tensile stresses above about 4000 psi on the alpha to beta transformation is to cause all of the transformation volume changes to occur in a direction perpendicular to the applied stress.

The temperatures for the start of the beta to alpha transformation under applied uniaxial compressive and tensile stresses were determined and related to the mechanism of transformation. Both applied uniaxial tensile and compressive stresses raise the temperature for the beginning of the beta to alpha transformation 3C/1000 psi at a cooling rate of 2 C/hr. This can be explained qualitatively on the basis of the transformation being a shear-type transformation. A similar effect has been observed in shear transformations in other alloys. Shear and normal compressive components favor the beta to alpha transformation, but a normal tensile component opposes the transformation. Accordingly, the shear stress in the beta to alpha transformation in plutonium aids the transformation more effectively than the normal stress opposes it. For this explanation to be valid, the shear displacement would have to be much larger than the dilational displacements.

It was previously reported that the cycling of a textured alpha plutonium sample into the beta phase does not restore the alpha to a random condition. This was borne out in recent investigations with added indications of a definite crystallographic relationship existing between alpha and beta. For example, it was noted that in alpha samples, oriented by beta to alpha transformation under compressive stress, an excursion into the beta region results in a 90° shift in the [010] direction. While this was not found in every specimen examined, it was repeated a sufficient number of times to indicate that certain temperature and stress conditions arising during the beta to alpha transformation can result in this rather unusual crystallographic phenomena. The fact that the [010] direction appears to be so sensitive to the direction of applied stress during the formation of alpha from beta plutonium suggests that maximum change occurs in the [010] direction.

A study of slip lines and directions was undertaken on textured alpha plutonium samples. Alpha pressing resulted in the formation of slip lines which are evident in photomicrographs. These have not, as yet, been completely analyzed.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations. The results of routine examinations and measurements are or will be reported as part of the sponsoring research and development programs. During the reporting period, examination and testing included the following:

Photomicrography	- - - - -	46
Photomacrography	- - - - -	68
Photomosaics	- - - - -	7
Autoradiography	- - - - -	18
Replication	- - - - -	7
Fission Gas	- - - - -	12
Hydrogen Determination	- - - - -	15
Density	- - - - -	11
Dissolution	- - - - -	31
Microdrilling	- - - - -	55
Microhardness	- - - - -	5
Macrohardness	- - - - -	122
Tensile Testing (Rm Temp)	- - - - -	28
Tensile Testing (Hi Temp)	- - - - -	29
X-ray	- - - - -	3

Project CAH-136, Service Addition - 327 Building. The project bid package was completed. Bids are due in March.

"E" Cell Metallographic Facilities. Operational tests were completed on the hand-operated fine grinder. Metal removal rates were good and remote operation of the paper rolls was satisfactory. Two additional grinders were fabricated and installation of these and the belt sander is in progress.

Molding of Spline Seals. A mold was designed and fabricated and two prototype spline seals were molded from polyvinyl-chloride. The seals will be tested by IPD Plant Engineering personnel to determine whether the new seals will solve the spline-seal leaking problem.

Strain Gage Spot Welder. Strain gages were successfully attached to cold Zr-2 process tubing using the remote spot welder in a mockup. In-cell testing of the spot welder is in progress.

"B" Cell Modification. Preliminary drawings were completed for a cell tray, heater-stirrers and hood for the south end of "B" cell.

2. Metallography Laboratories

During the report month 398 samples were processed, a total of 732 micrographs and macrographs taken, 1949 negatives printed, and 7032 prints processed.

3. High Temperature Lattice Test Reactor (HTLTR)

Fabrication and installation of another set of graphite heating elements was completed--two of the four heating elements presently installed are of a new "hourglass" design, while the remaining two are of the rectangular design used in Run No. 1. Inadvertent fabrication errors including (1) slippage of a jig holding an element, (2) breaking an element by bumping, and (3) cutting an end taper backwards, resulted in the loss of three heating elements. Electrical resistances of the finished elements as measured with a Wheatstone bridge indicated the individual resistances varied $\pm 1\%$ from an arithmetic average of the measured values. When installed, the individual heating element circuits matched within $\pm 5\%$ of an arithmetic average.

The molecular sieve-type gas cleanup system was designed and installation is 70% complete.

The following startup items were completed prior to start of evacuation:

1. Vent line relocated.
2. Tygon cooling water hoses replaced.
3. Heating elements replaced.
4. Ground bus isolated from mockup shell.
5. New thermocouples installed.

6. Sized three orifices and installed.
7. Safety and control rods installed.
8. All recorders calibrated.
9. Gas analysis equipment adjusted.
10. Recording-controlling pyrometer calibrated.
11. Blower bypass installed.
12. Graphite sample system installed on west door.
13. Install doors.

The safety rod assembly was installed on the mockup preparatory to testing. Piping and wiring of the rod was completed, and cold testing (on the mockup) was started. The horizontal control element was disassembled and all components weighed prior to installation in the large mockup.

The examination of TD Nickel cladding from fuel and poison rod specimens from the HTLTR environmental test described last month was completed. There was no evidence of reaction between the cladding and UO_2 or Al_2O_3 - Gd_2O_3 mixture.

Metallic specimens which were exposed in the initial run of the HTLTR mockup in the 314 Building were found to have been affected by an oxidizing atmosphere. In this environment the chromium-containing alloys were the least affected because of their oxidation resistance, whereas the alloys, TD Nickel, Hastelloy B, and Nickel 200 were severely corroded. Surveillance of materials will continue in future runs of the 314 mockup. The chemical nature of the atmosphere in this test chamber is expected to change as the water content is lowered with continued operation.

4. EBWR Fuel Elements

EBWR Plutonium Recycle Fuel. The next shipment of plutonium fuel rods to Argonne is scheduled for March 15, and will complete the plutonium fuel required for the EBWR core in support of the Plutonium Recycle Demonstration Experiment. More than 700 depleted UO_2 -1.5 PuO_2 fuel rods and 20 special rods (Table I) will make up the shipment. Most of the 700 rods were returned from Argonne National Laboratory and repaired by replacing the top end cap.

TABLE I

SPECIAL RODS

<u>Quantity</u>	<u>Core Composition</u>	<u>wt% Pu-240</u>
4	Al-3.35 Pu-2 Ni	8
4	Al-3.35 Pu-2 Ni	26
4	UO_2 (Normal)	--
4	UO_2 -1.5 PuO_2	20
4	UO_2 -1.5 PuO_2	26

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Fabrication of 150 spare fuel rods will begin after the above shipment is made.

Irradiation-Testing of EBWR Prototypic Fuel Rods. Maximum exposures of the in-reactor (MTR) and discharged, short EBWR specimens are 4.1×10^{20} fissions/cm³ (15,000 Mwd/ton of fuel) and 4.8×10^{20} fissions/cm³ (17,000 Mwd/ton of fuel), respectively. Capsule irradiation tests are continuing and indicate that the EBWR fuel will perform satisfactorily. Seven capsules with short sections of EBWR fuel rods are still being irradiated and 25 have been discharged from the MTR.

A 21-rod bundle (5 x 5 array with corner rods missing) with full-size replaceable EBWR fuel rods is being irradiated in the PRTR rupture loop. The bundle is operating with a heat output of 450 Kw, which is equivalent to an average rod surface heat flux of 51 w/cm². The maximum operating conditions to date have been: heat output - 520 Kw, rod heat flux (average) - 59 w/cm², and rod heat flux (maximum) - 65 w/cm² (60 w/cm² is maximum value for EBWR). The bundle has operated for 552 hours in the loop. Upon completion of the one cycle in the loop, it is planned to position the bundle in the PRTR core and continue the irradiation test. Tentative plans call for rod replacements after the following exposures: one rod at 0.4×10^{20} fissions/cm³ (1500 Mwd/ton of fuel), two rods at 1.4×10^{20} fissions/cm³ (5000 Mwd/ton of fuel), and two rods at 2.7×10^{20} fissions/cm³ (10,000 Mwd/ton of fuel).

Twelve, 3-rod clusters of full-size EBWR rods commenced irradiation. Proposed goal exposures for the four-cluster groups are: 0.4×10^{20} fissions/cm³, 1.4×10^{20} fissions/cm³, and 6.8×10^{20} fissions/cm³ (25,000 Mwd/ton of fuel). Rods in all clusters are of the repaired type.

EBWR Cladding Stress Studies. Tests are continuing to determine the nature and magnitude of stresses developed in EBWR cladding during vibrational fuel loading using either electro-dynamic or pneumatic variable impact vibration (VIV) shaker units.

Many resonance frequencies were found when using the electro-dynamic system for vibrating EBWR tubes. The resonance frequencies induce two general types of waves in the Zircaloy tubes: (1) horizontal standing waves with maximum deflections normal to the tube axis and (2) vertical waves which are superimposed on the sinusoidal wave generated by the shaker system. Cracking from fatigue is easily produced from the horizontal motion of the horizontal standing waves but sustained shaking with only vertical deflections will not induce cracking. Stress-coat analysis of cladding strains during shaking has helped delineate areas of high stress, while subsequent strain gauge measurements in these areas show stresses of 20,000-40,000 psi.

It was further observed that stress values in the tubing near the clamp assembly and resonance frequencies changed during UO_2 loading of the tube. Particularly bad stresses ($\sim 20,000$ psi) were recorded in the cladding near the clamp at 350-400 cps for full fuel rods. Specific clamp arrangements, designed to minimize stress concentration, showed some promise of success. It also appears that solid lubricants such as MoS_2 are effective in reducing cracking, probably because of reduced fretting action between the stainless steel clamp and the Zircaloy tubing.

Studies with the pneumatic VIV shaker show different stress patterns. Momentarily high stresses ($\sim 20,000$ psi) are obtained during impact, but they dampen out within 0.1 sec. Impact pulses are produced every 0.25 sec and decrease in intensity as the rods are filled with fuel. Initial studies indicate that the integrated fatigue during a normal loading operation is much less for the VIV system than for the electrodynamic system.

EBWR End Cap Cracking. Metallographic examination is being performed on selected rods exposed to thermal cycling studies. Four of these rods showed ultrasonic defect indications in the end cap region prior to cycling. Of these end cap indications, two remained relatively unchanged after thermal cycling and two propagated extensively. Weld-bubble voids were found in nonpropagating defects, and cracking was found in end caps with propagating defects.

Subtle differences appear to exist in ultrasonic trace records made on these rods. Efforts to determine if cracks can be distinguished from voids by analysis of ultrasonic signals are being made.

One rod, originally free of defects prior to fabrication, demonstrated defects 2-3 mils deep in the cladding. Preliminary examination of the cladding "defects" indicated that they are surface roughness caused by oxide fuel impinging on the interior surface while loading.

5. Other Customer Work

Saxton Fuel Element Fabrication. Forty-eight kilograms of UO_2 -6.6 wt% PuO_2 were blended and sent through the various steps of the Nupac process. Eighty-four individual rod loads were prepared from this material and are now on hand.

Six fuel rods, loaded by a special technique, resulted in contamination of the counterbore areas. This problem did not occur in loading three additional rods by the standard technique, which uses a sleeve insert to protect the counterbore. X-rays of these three rods showed that a 1/8-inch alumina spacer, which is inserted in the fuel tube prior to loading, had shifted in position. Fine Nupac powder had worked its

way around the spacers and moved them 1/16 to 3/16-inch up from the bottom of the tube. Permission to eliminate these bottom spacers is being sought from Westinghouse.

Ultrasonic examination of these rods showed no crack indications in either the weld or clamp areas.

Weld samples were sent for approval to Westinghouse and were rejected on the basis of not having 100% penetration. Examination showed that all welds did have 100% penetration, but that the thickness from the throat of the weld to the outside surface of the end cap was less than 100% of the counterbored tube wall thickness in a number of samples.

A second series of weld specimens was prepared, changing the weld parameters to eliminate this problem. Photomacrographs at 5X showed the weld throat to be equal to the wall thickness. These samples are now being examined by Westinghouse. An attempt is being made to develop nondestructive test to ultrasonically measure the cross section of the weld metal.

Approximately one-half of the 72 kilograms of Nupac material, which was originally rejected for high gas content and low density, has been given a 1500 C hydrogen furnace treatment as the first step of recycling. In general, the gas content is being lowered in the range of 0.03 to 0.07 cc/g, although most of the results are above the 0.06 cc/g specified for Saxton.

Manager
Reactor and Materials Technology

PHYSICS AND INSTRUMENTS DEPARTMENTMONTHLY REPORTFEBRUARY 1965FISSIONABLE MATERIALS - O2 PROGRAMREACTORNPR Coproduct Studies

Four-group diffusion theory calculations, performed on PHYSICS CHAIN, yielded a cold, clean, green k_{∞} that is 25 mk lower than that measured in the PCTR for the NPR 1.95% U-235-enriched-coproduct fuel element. This difference is five times larger than that found in comparisons made on the 0.95% U-235-enriched-standard NPR fuel element. A possible source of error in this calculation may occur in the TEMPEST calculation of the thermal neutron spectrum in the strongly absorbing target of the coproduct fuel. This target and approximately 50% of the cooling water between it and the driver fuel annulus are homogenized in the spectrum calculation. This intuitive recipe may yield too soft a spectrum and, hence, an abnormally high absorption with a resulting low k_{∞} . In future calculations, thermal cross sections obtained from THERMOS are to be incorporated in the few group diffusion calculations for the cell, in an attempt to improve the calculational technique.

Power peaking in the proposed NPR 1.95 coproduct block test (a 3 x 7 array of columns) has been investigated using the two-dimensional 9-ANGIE code. Group constants which had been previously obtained from two-group diffusion theory cell calculations were used. A previous error in reflector absorption, which caused severe buckling of the reactor flux, has been corrected. The new predicted power peaking appears to be excessive for a test calculation on the present spike load in the NPR, and can be expected to be excessive for the block test also. This overprediction is not understood at present. It could be due, in part, to the apparent excessive thermal neutron flux peak in the reflector. This flux peak appears in all diffusion theory calculations on N-Reactor, and is not observed in operational experience.

A split exponential experiment has been proposed for the purpose of predicting power peaking in the block test. The proposed exponential would consist of three vertical rows of ten tubes each of the coproduct fuel flanked on each side by three vertical rows of standard 0.95 N-Reactor fuel tubes. Additional rows of 0.95 fuel tubes could be added to improve spectral

equilibrium and flux flattening. Two-group diffusion theory calculations on this split exponential indicated gross spectral equilibrium would be achieved to within a few percent in the middle column of the three coproduct columns. The predicted power peaking (U-235 fission product activity) is approximately three percent. These perturbations are relatively small indicating the experiment can be expected to be difficult. The final decision to carry out the experiment has not been made. Negotiations between Reactor Lattice Physics and N-Reactor Department are under way.

Experiments to determine the material buckling B_m^2 and control rod strengths ΔB_m^2 in the coproduct lattice using the N mockup exponential pile have been completed. The measurements were made for these cases: Air coolant, water coolant, water coolant plus six control rods, and water coolant plus four stages of simulated flooding in the steam vent holes. The vertical relaxation distance, buckling, and all three extrapolation distances have been computed from the measurements by program ICEDT. The results have been made available in rough draft form. A full report is now being written.

Nuclear Analyses of Graphite-Water Systems

To obtain a better understanding of the physics of heterogeneous water-cooled, uranium-graphite lattices, detailed analyses of some published PCTR results have been started. Initial calculations were carried out for a solid rod geometry in a graphite cell with and without water. The idealized cell dimensions are given below:

<u>Region</u>	<u>Outer Radius (cm)</u>
Natural uranium	3.178
Aluminum	3.278
Water (or air)	3.840
Aluminum	4.003
Graphite	15.050

Group cross sections were obtained using the TEMPEST-GAM routine, and k_{∞} was determined by means of one-dimensional diffusion theory (HFN). The TEMPEST group constants were based on successive composition iterations using flux depression corrections obtained in the HFN calculation. The results are given below and compared with the published experimental data.

<u>Iteration No.</u>	<u>k_{∞}(wet)</u>	<u>k_{∞}(dry)</u>
0	1.0238	1.0174
1	1.0180	1.0233
2	1.0172	1.0237
3	1.0171	
Exp.	1.015 ± 0.002	1.024 ± 0.002

The calculations have now been extended to tube-in-tube geometry. Early results indicate gross discrepancies of about 4% Δk .

Development of NPR Cell Models

A series of calculations for the tube-in-tube NPR cell geometry has been started to obtain more reliable cell models. Comparisons will be made between diffusion and transport calculations, single and multiple thermal groups and overlapping thermal groups.

Instrumentation

Two power supplies and a line filter distribution chassis were completed and made ready for installation with the N Reactor subcritical neutron monitoring system. General requirements were established for a radiation monitoring instrument to be used at the N Reactor heat exchangers.

Development continued for the reactor irradiated fuel cooling age monitoring system. Circuits were completed for the pulse preamplifier, the phototube signal current metering and alarm, and for the ratio metering and alarm sections. Fabrication continued on the analog output circuits. The phototube anode current metering and alarm circuits have a full scale current sensitivity of 5 μ A with an available alarm-trip selection from 80 nA to 3 μ A. The ratio metering circuit provides an alarm-trip over four decades with a trip-point hysteresis of 10% of full scale. A majority of the system has now been completed and tested.

System Studies

It has been established that N Reactor Phase II (conversion) system simulation should consist of lumped steam and condensate headers. With this simplification the problem can be handled on the EASE 1132 computer such that over-all system transients due to turbine trip-off or scram conditions can be studied.

The export system simulation consists of two export line models (one for each turbine) and one condensate return line model. Each export line is

represented by a two-node line model. The line volume for each set of four export lines is divided into two equal parts. The partial volumes are connected by a frictional resistance equal to the total frictional resistance of the four lines in parallel. This representation provides a good approximation of the time lag existing between turbine load changes and the resulting changes in main steam header pressure.

Several minor changes were made in the primary-secondary coolant loop simulation model to set it up for Phase II conditions. These include larger surge tank volume, different operating pressures and temperatures and different heat transfer coefficients.

Accurate data on some of the conversion system components, such as condensate return valve characteristics, are still not available. Parametric studies to determine the effects of these variables are planned.

A set of scaled drawings was completed for the three surge tank secondary simulation of N Reactor. No pot set or constants lists are completed as yet. This simulation will be used to check out methods of pressure and level control when using three surge tanks.

SEPARATIONS

Criticality Experiments with PuO₂-Polystyrene Assemblies

Further critical experiments were performed with PuO₂-polystyrene compacts containing ~8 wt% Pu²⁴⁰ in Lucite reflected assemblies. Since previous experiments were performed with 2.2 wt% Pu²⁴⁰ material, these results yield data for the effect of Pu²⁴⁰ on criticality. The Pu concentration in the plastic compacts comprising the core material was 1.03 g/cc with an H/Pu²³⁹ atomic ratio of ~16. The experiments thus provide data for nuclear safety guidance in handling undermoderated Pu mixtures; the clean systems also provide data needed for checking criticality calculations.

In the initial experiments, the fuel compacts were contained in 20 mil thick aluminum cans. The cans were removed and the compacts coated with a thin layer of plastic. This was done to evaluate the worth of the aluminum cans and the void spaces present in the cans as well as to permit measurements of cleaner systems. Corrections for the effect of the aluminum cans and the decreased core density are not trivial; these were found to amount to about 20% in mass for near cubic geometry when the difference in core density was ~15%.

Criticality was determined for various reflected rectangular parallelopipeds with geometrics ranging from a near cube, to a long parallelopiped,

to a thin slab. The critical mass due to these geometry variations ranged from 17 kg to 37 kg Pu. Analyses of the results are proceeding.

Calculation of Neutron Decay Constant by Monte Carlo Technique

In connection with the analyses of pulsed neutron source experiments and measurements of reactor noise, a Monte Carlo technique has been devised for computing neutron decay constants for comparison with experiments. It is assumed that n neutrons are in the system initially with a fission spectrum energy distribution. These neutrons are then traced and the fractions which fission or are removed from the system by absorption or leakage are recorded as a function of time. After these quantities have been found, the decay constant may be determined by a double summation process. The advantage of this method is that only one neutron generation is traced in determining the decay constant.

This technique has been inserted in the SMC computer code. A sample calculation was carried out for a critical system of plutonium moderated by polystyrene ($H/Pu = 15$). A pulsed neutron measurement yielded an α of $1763 \pm 200 \text{ sec}^{-1}$. The corresponding α calculated by the Monte Carlo technique was $2045 \pm 300 \text{ sec}^{-1}$. The estimated error limits for the calculated result include statistical uncertainties only and are large because only 2000 neutron histories were traced to determine the fission and removal probabilities.

Criticality of U Rods in Aqueous U Solution

A version of the GAMTEC-2 Code has been modified that permits calculation of resonance absorption within the uranium solution of a U-rod, U-solution system. The new code, GAMTEC-3, is presently being checked with test problems.

Critical Mass Laboratory Instrumentation

Development work on a time-to-pulse height converter circuit was completed. The converter will be used with the TMC 256 channel pulse height analyzer to measure neutron lifetimes in the 10^{-8} to 10^{-6} second range. The converter was designed using 500 kc digital modules which limit the time interval between two successive neutron detections to a two microsecond minimum. However, since intervals less than two microseconds will be measured, five Mc digital modules have been ordered for the converter.

Instrumentation was assembled for determining subcritical reactivity of systems using the p-method of noise analysis. This method makes use of scaler gating to record a count whenever counts are present in a given time

interval and to record a zero whenever no counts occur in the interval. A bistable element was used to gate the scaler and a variable frequency pulse generator provides the time interval.

The operation of the noise analysis system at the Critical Mass Laboratory has been improved with the installation and calibration of new output smoothing filters of longer time constants (10 cps filters).

Consulting Services on Nuclear Safety - Criticality Hazards

1. Nuclear Safety in Battelle-Northwest

The following nuclear safety specifications were reviewed:

- a) Specification B-13, for Experimental Reactors, covering a 2 wt% $\text{PuO}_2\text{-UO}_2$ fuel rods
- b) Temporary Specification No. 7, for Ceramics Research and Development, on reprocessing a special batch of 93 wt% enriched UO_2 .
- c) Temporary Specification No. 8, for Reactor Lattice Physics, on loading fuel cans with 20 wt% Pu-Al alloy discs and polyethylene discs.

2. Nuclear Safety of Off-Site Shipments of Fissile Materials

Two shipments were reviewed and approved of slightly enriched fuel elements to Idaho Falls; the categories of shipment were Class I and Class III.

3. Nuclear Safety Training and Education

A course of lectures in nuclear safety is currently being given to personnel of Vitro Engineering Company. The lectures are designed to acquaint technically oriented personnel with the technical bases and computational methods used to ensure nuclear safety and prevent criticality accidents. The material is being presented in 10 lectures, each of two hours duration. The classes are being held in the outer conference room of the 703 Building, with 21 persons attending.

Instrumentation and System Studies

Tests were completed on the remaining eight BF_3 neutron counter tubes for the neutron counting system used during the final inspection of material.

Tests with the instrument developed to measure Am^{241} distribution in a chemical process column indicated the sensitivity was greater than necessary; thus, a smaller GM tube detector was installed and shielding was added. Recalibration of the instrument was completed and it was again moved to the 234-5 Building for further in-use tests.

METALLURGY - Nondestructive Testing

N Fuels Testing

Fabrication of a fuel handling system for use with the surface contamination tester (AC-3) has been completed and checkout of the entire system is about 60% complete. Delivery has been obtained on all components and two test heads have been installed, checked out, and calibrated. Checkout of the control circuits, which tie in with the test chamber to operate the elevator, has been completed. The mechanical operation of the elevator and the light tightness of the test chamber are satisfactory. A microgram planchett, used for system standardization, will probably require recalibration because its alpha activity is about 10% low. This activity must be accurately known because it is used to establish the basic sensitivity of the tester. A check print of the nuclear circuits has been completed and the tester control schematic is partially drawn. A preliminary draft of the instruction and maintenance manual has been prepared.

The current clad-to-core bond test performs satisfactorily; however, the usage rate on the internal bond transducer is considered excessive (over \$5000 per year.) An attempt will be made to operate the clad-to-core bond testers using a through-transmission method instead of the current pulse echo technique. This will reduce the transducer performance requirements and decrease the number of chassis required by a factor of two.

The cap-to-clad closure test, which is used to detect porosity in the braze metal, is still plagued with reproducibility problems caused by equipment design.

It is planned to replace both existing pulse echo tests with one through transmission test. The required transducer design is complete and the transducers were ordered during the latter part of February.

Design of a TIG braze closure tester, which will be used to determine whether or not the braze wire melted during the welding cycle, has been completed. A prototype crystal mount and manipulator has been designed, detailed, and sent to the shop for fabrication. Work orders have been written to cover procurement and fabrication of all components except transducers. Procurement of transducers was postponed pending evaluation

of engineering samples manufactured by a local vendor.

The chevron end tester, which is used to test the bond quality of the beveled portion of the chevron end cap, was used to evaluate two trucks of engineering material. The test has been equipped with a new type of transducer and new transducer holders. The test is still more difficult to align than is desirable for a production line tester. A crystal mounting manipulator has been designed, detailed, and is being fabricated.

Fabrication of the mechanical portions of the remotely operated irradiated fuel test station (UT-10B), for use in the 105-N fuel examination basin, has been completed and the unit delivered to 189-F to be assembled with parts from the 100-K basin fuel tester.

Development is continuing on an eddy current test to detect cladding stria on irradiated N fuels. There is some evidence that irradiated fuels could encounter outer-clad-splitting failure as a result of circumferential stresses concentrated at longitudinal stria in the cladding. Efforts this month were directed towards selecting an optimum test frequency. Measurements of phase angle between stria and surface depression vectors were made at 120 kc and at 240 kc. These angles were found to be approximately 130° and 90° , respectively. Although 90° is considered to be ideal, the depth of penetration at 240 kc was reduced to where sensitivity to the smallest depth stria was unsatisfactory. A loss of depth sensitivity cannot be tolerated and development will be continued using the 120 kc system. If project funding is sufficient, strong consideration will be given to a two-frequency tester--the high frequency tester to be used on the thin wall NOT's and the low frequency tester to be used on the thicker walled NIT's and driver tubes.

Experiments were conducted on the yoke design probe head which will be used on the irradiated fuels (rather than a tape head as used on preirradiated fuels). All experiments were performed out of water. The yoke head appears to function as well as the tape head. Other problems such as wear on the neoprene bumper and floating debris interfering with the slide mechanism are yet to be investigated. It is desirable to provide some protection for the sensing coils which are positioned within a few mils of the fuel element. A thin stainless steel cap was evaluated for this purpose, and tester response was found to be substantially unaffected.

N Reactor Department - Test Engineering Support

An engineering study to establish the design criteria for an enrichment tester to measure the enrichment of fuel elements prior to reactor charging, is approximately 50% complete. Satisfactory identifications were obtained

at the four enrichment levels (natural, 0.95, 1.25, and 1.96) on all but the 1.25 enriched, driver fuel element. The measured gamma activity from this driver fuel is only 1% higher than the measured activity from a 0.95 enriched fuel element; an insufficient difference for reliable identification. The problem is caused by the 4% smaller diameter and 5 mil thicker cladding of this fuel compared to a regular NOE (outer enriched) fuel. This problem may disappear because present manufacturing schedules do not include further production of 1.25 enriched driver fuels.

Design data have been obtained on two test head parameters which are important in obtaining a sufficiently accurate statistical count. The effect of gamma detector size has been investigated over a diameter range of 1-1/2 - 3 inches. The spacing between fuel element and detector has been investigated over a range of 1/4 - 3 inches. A 3-inch diameter detector and a 1/2-inch spacing have been tentatively selected since they yield the largest count which has a statistical error of less than 1% at the 3 σ probability level. Currently, variations in fuel element dimensions such as diameter, clad thickness, and support height are being investigated.

AlSi Fuels Testing

A prototype design of a new handling mechanism to position AlSi fuels for bond and penetration testing has been requested. Engineering scope drawings are 95% complete and will be submitted for customer approval.

Work continued on an improved bond test to simultaneously monitor the internal and external bond quality of AlSi fuels. Approximately 50% of the electronics has been designed, breadboarded, and tested. This includes the timer, pulser, first and second delays, RF amplifier, gate pedestal generator, and flaw gate. Nuvistor tubes have been used exclusively with the exception of the pulser tube. Construction of prototype model of these circuits is nearing completion. A memo report, comparing the performance characteristics of several different Nuvistor amplifier configurations, is being prepared.

Work continued on the development of a spectrophotometer for determining the feasibility of detecting the formation of pinhole weld defects in AlSi fuel elements. The technique consists of monitoring the emission spectrum of the welding arc. A literature search was conducted to define the optical requirements of such an apparatus. Due to the uncertain nature of the problem, it was deemed advisable to employ a system with a fairly high resolving power (2-10 Å), high sensitivity, and a fair degree of versatility. A comparison was made of the characteristics of filter systems, dispersive systems, diffraction gating systems, and prism systems. A review of the

engineering catalogues failed to disclose any commercially available equipment suitable to the present problem; therefore, a special system was designed which is an adaptation of a Dual-Channel Comparative Spectrophotometer used to detect stellar flares. This system has a primary channel width of 10 Å with resolution determined by the source intensity. It is adjustable over the entire visible spectrum and has two differential outputs, or four absolute outputs, making a variety of experiments possible. Procurement and construction of prototype equipment has been temporarily curtailed pending receipt of continued funding.

IPD - Test Engineering Support

A study to determine the most promising method of measuring the thickness of the aluminum process tubes in the older Hanford reactors has been completed. A final report has been written recommending that ultrasonic pulse-echo equipment be developed for this purpose.

NEUTRON CROSS SECTION PROGRAM - 02/04 PROGRAM

Triple-Axis Spectrometer

The new third-arm counter shield and 2-inch diameter end-window BF₃ counter has been installed and put into operation as the scattered-neutron detector of the spectrometer. Initial measurements of neutron scattering from vanadium indicate an increase of about 50% in counting efficiency over the small He³ counter previously used. Measurements of inelastic neutron scattering from H₂O taken with the He³ detector showed systematic differences of 10- to 50-percent for different conditions of collimation of the initial monoenergetic neutron beam.

Time-of-Flight Spectroscopy for Slow Neutrons

The study of the scattering of neutrons from zirconium hydride was continued for an initial neutron energy of 0.50 eV. The angular distribution of neutrons scattered after excitation of the first three Einstein levels was observed. This is the first observation of the excitation of the third level insofar as known. A preliminary analysis of the results indicates good agreement with the theoretical expectations of the scattering from a simple harmonic oscillator. A scheme has been developed for calculating the expected scattering on the basis of a multiphonon treatment which includes the acoustical modes of the lattice. This scheme uses the computer programs LEAP, ADDELTA, and SCS.

Further measurements were made of the resolution elements of the spectrometer system. The measured elements were in good agreement with expected values.

Several days running time were lost during the month because of failures of the time-of-flight analyzer and punched tape readout.

Fast-Neutron Cross Sections

All of the total cross section data obtained to date have been processed with the program LIL NED and the final card decks of averaged and corrected data have been prepared. There are now at least two acceptable measurements on 27 samples and single measurements on 41 others. The results on the separated isotope Cr^{53} were determined to be acceptable and this sample will be returned to Oak Ridge.

A review of the error analysis used in the data-reduction program BIG NED was completed. An improved analysis was developed and is now being incorporated into the program.

An optical-model program ABACUS-2 was received from Brookhaven. The program calculates cross sections and makes comparisons with experimental values.

Work continued on the preparation of a B_2O_3 glass sample for measurement of the oxygen cross section. Samples of Li^6 and Li^7 were received from Los Alamos for measurement. Procurement was started on samples of Th^{232} , U^{233} , U^{235} , U^{238} , Np^{237} , and Pu for future measurement.

REACTOR DEVELOPMENT - O4 PROGRAM

PLUTONIUM RECYCLE PROGRAM

PuO_2 - UO_2 Graphite Lattice Studies

Data analysis for the 8-3/8" lattice experiment containing 0.9 w/o Lx PuO_2 is complete. The infinite multiplication factor for this lattice is $1.21 \pm .01$. The largest contribution to the uncertainty in the value of k_∞ is due to an estimated $\pm 1.6\%$ uncertainty in the value of

$\left(\frac{\sigma_a}{\sigma_f} \right)^{\text{Pu}}$ calculated by the computer code THERMOS. The results of this experiment have been summarized for presentation at the June meeting of the American Nuclear Society.

The effective cross section ratios calculated with THERMOS, using the graphite scattering kernel, now agree with the 8-3/8" lattice experiment to within 2%. Previously, using the gas scattering kernel, the discrepancy between THERMOS and experiment was 10%.

$\text{PuO}_2\text{-UO}_2$ fuel utilizing plutonium containing 29.6% Pu-240 will be ordered from Plutonium Fuels Engineering. Calculations to help determine the PuO_2 concentration which should be specified has begun. THERMOS is being used to generate η curves as a function of PuO_2 concentration, and moderator-to-fuel volume ratios for single rods in a graphite lattice. These η values will be combined with values of $P(238)$ generated from Hellstrand's data, and $P(240)$ from Nichols' data, to yield a curve of k_∞ as a function of fuel-to-moderator ratio.

Critical Experiments with D_2O

The Plutonium Recycle Critical Facility became critical on February 17, 1965, with seven, 19-rod clusters of 1.8 w/o Pu-Al moderated by D_2O . The lattice is hexagonal with a center-to-center spacing of 8", and the clusters are located in the seven center positions. Three safety rods are at a radius of 13.86", and three control rods are at a radius of 22.17". This loading is critical at a moderator level of 97.25".

Kinetics Parameters Measured in the H_2O -Moderated 1.8 w/o Pu-Al Core

Data from a step change of reactivity (boron arrow) were analyzed using a seven-group reactor kinetics model. All seven groups were identified from the data. The value of β/λ of 18.3 obtained from this analysis agreed with the value obtained from the two-group analysis. A value of 1.46 mk was obtained from the IBM code STEP for the reactivity inserted into the reactor by the boron arrow.

An equation that would allow a value of λ to be obtained from the parameters of each of the delayed groups of neutrons can be derived from the seven-group reactor kinetics model. However, an attempt to use this equation showed that a small error in either the slope or the intercept of the delayed group would give a large error in λ .

Summary of D_2O Experiments

A summary of D_2O lattice experiments performed at Battelle-Northwest is being prepared. This summary is to be included in the D_2O lattice data compilation of the European-American Committee on Reactor Physics, EACRP-L-42.

The Beta-Ray Spectrum of Zirconium-97

An understanding of the decay of zirconium is important because of the use of zirconium as a flux monitor, and because of the occurrence of zirconium as a fission product. The isotope Zr^{97} decays by beta-ray emission to Nb^{97} with a half life of 17 hours. The Nb^{97} , in turn, decays by beta

emission to Mo^{97} with a half life of 72 minutes.

The momentum distribution of the beta-ray spectrum of Zr^{97} has been measured using the Double-Focusing Beta-Ray Spectrometer. The data are being analyzed to obtain a Fermi plot of the results. A quick entry gate has been added to the spectrometer to allow sources to change without loss of vacuum. The addition is in preparation for measurements using Nb^{97} sources. This research is being done cooperatively with personnel from Washington State University.

PRTR Burnup Experiments

Work has been initiated during the last month with Applied Mathematics to establish an adequate statistical model to describe the propagation of burnup analyses errors in the present method of determining cross section ratios. Also, more than one method of determining the ratios is available and one would like to know if one method is significantly different or better than another method.

Analysis of Thermal Neutron Spectra in Pu Poisoned, Light Water Systems

Infinite medium spectra have been calculated for the Harwell system (AERE-R-4668) using the Nelkin scattering model in the code SPECTRUM. Pu-239 cross sections (Sher and Leonard) averaged from 0 to 0.683 eV were obtained from both the calculated and measured spectra. It was found that the calculated spectra were rather insensitive to the cross section set (Sher or Leonard) employed in the calculation.

Values of η_L Pu-239 Calculated from SPECTRUM

<u>Cross Section Set Used in SPECTRUM</u>	<u>1197</u>	<u>H/Pu Ratio 402</u>	<u>131</u>
Sher	2.0525	2.0036	1.9391
Leonard	2.0532	2.0048	1.9400
% Diff.	.03%	.06%	.05%

(Subscript L denotes Leonard value.)

Comparison of Average Pu-239 Parameters Using
Calculated and Measured Spectra

	$\sigma_a(L)$	$\sigma_a(s)$	$\alpha(L)$	$\alpha(s)$	σ_f/σ_a	σ_f/σ_a
<u>H/Pu - 1197</u>						
Calc.	1009.1	1037.1	.4076	.4565	.7104	.6866
Meas.	1016.9	1045.1	.4171	.4666	.7057	.6818
% Error	-.8	-.8	-2.3	-2.2	+.7	+.7
<u>H/Pu = 402</u>						
Calc.	1034.3	1064.2	.4416	.4949	.6937	.6689
Meas.	1047.1	1078.2	.4565	.5107	.6866	.6619
% Error	-1.2	-1.3	-3.3	-3.1	+1.0	+1.1
<u>H/Pu = 131</u>						
Calc.	1012.3	1040.5	.4897	.5485	.6713	.6458
Meas.	1052.6	1086.3	.5054	.5652	.6643	.6389
% Error	-3.8	-4.2	-3.1	-3.0	+1.1	+1.1

As indicated by the author of the published data, the calculated spectra are too "soft." This is reflected in the values of α , since α increases as it approaches the 0.3 eV resonance.

In the H/Pu - 131 system, the error in calculated $\bar{\sigma}_a$ is $\sim 4\%$, however, the ratio of $\frac{\sigma_f}{\sigma_a}$ (tabulated as η) is in error by only 1.1% which is within the experimental error ($\sim 5\%$).

Further calculations are intended to compare the Wilkins and Wigner-Wilkins models with the measured data.

REACTOR PHYSICS CODE DEVELOPMENT

1. Program TEMPEST

This code as revised has been adapted to the PHYSICS CHAIN and is now also available on the Special Procedures Library. The SPL version is named SPLTEMPEST. Further modifications have been made to the SPL version to enable the user to read the TEMPEST cross section library

from (1) either decimal or binary cards as in the original code, (2) FORTRAN tape N, in binary card image, or (3) a Composite Library Tape on unit B as is presently done by the chain version. Which option to use is determined by the contents of columns 70 through 72 of the binary library data card. For the respective options, these columns should contain the following (right adjusted) entry: (1) Blanks, (2) N, or (3) -9. Of course, if options (2) and (3) are used, the tape unit, location, and reel number must be specified on the Scientific Applications card.

2. ZUT and TUZ

Programs ZUT and TUZ, which calculate resolved and unresolved resonance contributions, respectively, are now on the Special Procedures Library. Their respective SPL names are SPLZUT and SPLTUZ.

3. PHYSICS CHAIN

Revisions have been made to computer program COMBO which enable it to read TEMPEST and GAM multipliers directly from the Sigma-3 Library. This eliminates previous restrictions concerning the selection of Composite Library Tapes to be used when running COMBO PHYSICS CHAIN jobs. Revisions further allow SIGMA-3H and HFN only, or SIGMA-3C and CALX only, jobs to be run on a PCT containing this COMBO without the heretofore required COMBO input, namely, the first six entries of COMBO data as found on page 39 of HW-80968. Also, standard runs on this COMBO no longer require an entry in location 17 of the input instructions on page 39, HW-80968.

4. Error Function

The error function (ERF(x)) distributed as SHARE 3115 was obtained, converted to FORTRAN IV, checked out, and both versions were included in the FORTRAN Library. The accuracy of this error function routine is eight figures with an error of ± 1 in the eighth figure. The complementary error function should not be evaluated using this program since it is a single precision routine and an answer of one is returned for arguments larger than 4.17.

5. Hanford Basic Library

This was updated several times this past month to eliminate some inconsistencies between certain RBU records, and the processing requirements of BARNS II. Th-232 was updated to contain cross sections that conform to the latest published information, and an infinite dilute

resonance absorption integral of 82.5 barns was obtained from the data. Protactinium was updated to smooth the curves around the 0.4 eV resonance. When the last update had been completed and a new library tape generated, a decimal library punch tape was made along with a new RBU Library Editor listing.

Pu-238 is being updated using methods suggested by B. R. Leonard. The updated Pu-238 and Np-237 should be available before the middle of March.

6. TWENTY GRAND

A program interrupt and restart option feature has been added to the two-dimensional diffusion theory code, TWENTY GRAND, recently adapted to Hanford use. This feature permits salvage of those runs which would otherwise be lost when terminated by the operator at an intermediate stage because maximum allowable computer time was exceeded. This advantage is obtained at the possible cost of increased turnaround time, because tape mounting is now needed.

Another added option is the ability to read in a flux guess from tape at the start of a run. This option has potential value in reducing operating time in certain situations. Only minor modifications in input instructions are needed with the new options. These modified instructions will be distributed to current holders of binary decks along with modified decks as soon as they are available.

7. Graphite Scattering Kernel for THERMOS

A graphite scattering kernel from the computer code, SUMMIT, has been made available on the THERMOS Library in location 755 (reel number 4378) under the identification 612-60. These data are for 20°C graphite. Data for 425, 600, and 1000°C graphite are also available.

A code, GASUM, was written which accepts as input the output kernels for SUMMIT and converts these kernels into the proper form for use by THERMOS. The results have been checked by comparison with the gas model and the new data are now in general use (HTLTR, PCTR, NPR).

Isotopic Analysis of PRTR Samples

Isotopic analyses were performed on 31 samples of PRTR-irradiated fuel elements: 11 plutonium and 13 uranium burnup samples from PuO₂-UO₂ fuel element number 5185 and 7 uranium burnup samples from PuO₂-UO₂ fuel element number 5213.

Isotopic analyses were provided on four samples in cooperation with the development of a double-spike isotopic dilution technique by Analytical Laboratories.

PRTR System Studies

Development was started on a mathematical model of a steam pressurizer for the PRTR primary loop. A study is being made to determine if the PRTR helium pressure control system can be changed to steam pressure control. The model will be useful in determining the steam generator size and the heater requirements.

Assistance and consultation were given in connection with a MIDAS program being written for the PRTR kinetics for hazards analysis purposes.

EBWR PROGRAM

Critical Experiments

The effects of various materials in the EBWR core have been evaluated. The following is a list of all the materials that affected the reactivity of the EBWR loading. A value of 0.2831 mk per fuel rod was used to obtain the reactivity values listed:

<u>Material</u>	<u>Reactivity (Fuel Rods)</u>
Lexan Templates	-11.3
Lucite Templates	- 2.0
Safety Sheet Followers	- 1.6
Safety Sheets and Control Rods at Full Out Position	- 0.4
Saran Tubing (Wire Traverse Thimble)	- 0.2
Temperature Differential Between PRCF and TTR Approach Tank	- 0.1
Removable Center Section	- 2.0

The critical mass in the PRCF before the removable center section was installed was 495.9 fuel rods. The corresponding value for the Critical Approach Facility was 484 ± 1 fuel rods. By making appropriate corrections from the above table, the critical mass of the EBWR fuel in a clean core and at a lattice spacing of 0.71 inch is 482 ± 1 for the Critical Approach Facility, and 482.3 ± 0.4 for the PRCF.

The reactivity worth of a copper rod was measured at several radii in the PRCF during the EBWR critical experiments. These data are being analyzed

to obtain the radial distribution of the function $\phi_{th}\phi_{th}^+$. A similar analysis is being done for the vertical distribution. When these curves are divided by the measured thermal flux, this will yield the radial and vertical distributions of the adjoint flux.

The worths of several rods which contained various reactor poisons (hafnium, europium, copper, aluminum) have been measured in two light water-moderated cores in the PRCF. These data are being analyzed to obtain the sensitivities of the cores to these materials. The worths of some of these rods have been calculated by three-group perturbation theory. The calculated values are in good agreement with experimental values. Thus it appears that the three-group perturbation theory calculations can be used to reliably predict the worths of these and other poison rods in future H_2O moderated cores.

Approach-to-Critical Loading Calculations

An HFN calculation, for an EBWR-fueled approach-to-critical lattice, using more than one thermal group, has been successfully run. The flux now converges in approximately 450 iterations. This represents about nine minutes of computer time for a two-region, eight-group problem. The effective multiplication for the 0.55 inch lattice changes from 1.004 to 1.002 when going from one thermal group to five thermal groups. The analysis will be extended to the other measured lattices.

Age Calculations

The calculations of age using HRG as reported in the January Monthly Report have been extended to other materials. The ages calculated with GAM-I are also included for the sake of comparison, since the differences between HRG and GAM-I represent the RBU Basic Library vs. General Atomic's cross section data.

TABLE I

Age to Indium Resonance (γ) cm²

Material	Source	HRG	GAM-I	
H ₂ O	Fission	25.1	25.8	27.8 $\pm$ 1.8 ^{(1)*} , 26.48 $\pm$.32 ⁽²⁾
	Po-Be 1	55.9	56.3	48.8 ^{(1)*}
	Po-Be 2	58.4	58.7	
	Ra-Be	55.1	55.6	48.4 ^{(1)*}
D ₂ O	Fission	114.1	112.3	109 $\pm$ 3 ^{(1)*}
Graphite	"	294.6	294.2	321.0 $\pm$ 8.0 ^{(1)*} , 307.8 $\pm$ 1.9 ⁽²⁾
Be	"	92.5	73.2	80.2 $\pm$ 2.0 ^{(1)*}
BeO	"	101.4	90.9	93.4 $\pm$ 4.7 ^{(1)*}
H ₂ O+Al W/M=1.0	"	59.7	56.0	76.8 ⁽¹⁾
H ₂ O+Al W/M=2.0	"	41.9	40.7	49.6 ⁽¹⁾
H ₂ O+Zr W/M=4.0	"	30.3	31.5	35.7 ⁽¹⁾

PHOENIX FUEL PROGRAMPu-Al Polyethylene Experiments in the PCTR

Analysis of the U-235/Al experiment is still under way. The double P₃ flux depression correction to the thermal group was insufficient to lower the k_{∞} of 1.0888--obtained for the homogeneous cell--to unity. About ± 0.015 in k can be explained by uncertainties in the composition. The presently calculated k_{∞} value of 1.0332 is considered unsatisfactory and a thorough reexamination of all calculations is now under way. The reactivity of the test cell is currently being recalculated using both Monte Carlo and transport (S_n) methods.

Pu-Be Studies

Determination of PuO₂ grain size effects in Pu-Be systems is under continuing study. Published results on the effect of grain structure on resonance absorption have been verified using the ZUT code. This method of calculation makes it possible to account for changes in resonance absorption due to lumping of the absorbing species in small grains embedded in a matrix of moderating material. The method has been verified experimentally out to absorber volume fraction as high as ten percent. Input tapes of resonance parameters for thorium and for the uranium and plutonium isotopes are being prepared.

(*) Ave. ge of all measured values in Ref. (1).

(1) McCarthy, A. E., et al., "Neutron Resonance Integral and Age Data," Reactor Physics Constants Center Newsletter No. 1.

(2) Paschall, R. K., "The Age of Fission Neutrons to Indium Resonance Energy in Water," NSE 20, 445 (1964).

Although used to calculate the grain effect, the ZUT code is a general code in which homogeneous and heterogeneous, Doppler-broadened resonance integrals are calculated by solving the integral slowing down equation for the collision density numerically. The primary output is limited to the individual resonance integral values and their summation, but ZUT output can be spliced into GAM-I and HRG through a suitable choice of input cross section self-shielding factors. Another desirable feature is that a Dancoff factor can be used as input. For closely packed absorber arrays, this amounts to a much more specific treatment of the problem geometry than the customary use of the Bell correction.

As part of the general study of Be systems analysis of the Livermore U²³⁵-BeO critical experiments was initiated. Work on setting up several cases for RBU analysis continues.

FAST REACTOR STUDIES

Compact Fast Reactor Studies

The neutronics analysis of U²³⁵O₂-W, U²³³O₂-W, and PuN-W reactors is nearing completion. During the past reporting period, most of the emphasis was placed on estimating the reactivity life for the three reactor types. It was found that an expression of the form

$$\delta k \sim C \frac{P(\text{mw})}{M_0(\text{kg})} T(y)$$

gives a good approximation to the reactivity change during T years of operation at P megawatts with an initial loading of M₀ kilograms. For U²³⁵(7% 28), U²³³, and Pu²³⁹ (5% 240) fueled reactors, the parameter C has the values .24, .25, and .3, respectively. Such a simple expression for the reactivity life is possible because: (1) The fission products have essentially no effect on reactivity; (2) the concentration of fertile fuel is very small, hence, the buildup of the higher isotopes is negligible.

Physics Analysis of a Fast Fuel Test Reactor

Reactor life and various reactivity coefficients were computed for PuO₂ fueling, varying core sizes, and compositions. For test reactors of interest, the life can be estimated from the equation

$$\delta k \sim .27 \frac{P(\text{Mw})}{M(\text{kg})} T(y) .$$

Because the fast test reactor has a high specific power and very little fertile fuel, a batch loaded core would only last about one month.

The reactivity change caused by core expansion, beryllium oxide addition, and sodium voiding was computed using a transport code with 16 energy groups. Due to the high neutron leakage fraction in the FFTR, any increase in material concentration, except a strong poison, will significantly increase reactivity. For example, a one percent uniform expansion in the core will decrease the reactivity by roughly .6%; a complete loss of sodium (initially 50 v/o) will decrease reactivity by about 8%.

Pu-Al Polyethylene Experiment in the PCTR

The 22 fuel cans for the first Pu-Al experiment have been loaded with fuel, decontaminated, and inserted into the PCTR. The major difficulty encountered during loading of the cans was the high radiation dose rate to the hands of those working in the glove box. Exterior decontamination of the loaded fuel cans was much easier than expected. There has been very little, if any, leakage of contamination from the fuel cans, but the cap joints of all cans have been sealed as an added precaution. Preliminary flux distribution measurements show that flattening of the axial thermal flux will be necessary before measurements can be made.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Mockup Tests

Preparations have been completed for the second run on the HTLTR mockup assembly. The run will include further tests of heater elements and thermocouples, and initial high temperature testing of the prototype control and safety rods. The safety rod, consisting of a blade of B_4C -graphite matrix clad in TD nickel, is mounted on top of the system and can be dropped into and removed from the hot graphite region with the prototype actuating system. The control rod is located horizontally in the graphite as it will be in the reactor. Its actuating system is mounted externally and operates the control rod through an opening in the mockup door. The run will consist of operation at 700°C for four days, a shutdown and visual inspection of the rods, and finally operation at 1000°C for five days if the inspection at 700°C permits it. During the course of the run samples of graphite and B_4C -graphite will be inserted for periods of a few hours, then removed and weighed. This will provide a progressive measure of the corrosion of these materials. In addition, samples of ceramics and metals will be placed at various locations in the mockup and left in for the duration of the run.

Materials Tests

Samples of UO_2 -graphite matrix material were tested for 100 hours at 1100°C in nitrogen containing 2.4% H_2O by volume. There was no visible deterioration. A weight loss of 4 mg cm^{-2} was observed. It is planned to make a similar test of B_4C -graphite matrix material.

Pulsed Source Experiment

The analysis of the pulsed source experiment on PCTR has been completed. Due to the weak source used and high background present, only the results for the least shutdown configuration (\$2.75 shutdown) possess reasonably good statistics. For this least shutdown case, the value for ten pulses per second is computed to be 2.51 ± 0.78 with lesser values for lower pulsing rates where the violation of the assumptions of the theory become important. It would appear that:

1. The $\frac{k\beta}{\lambda}$ method of determining shutdown will be quite useful in the range in which the delayed neutron contribution may be separated from the background with fair statistical certainty even with a reflected, highly heterogeneous reactor. With the much larger source used in HTLTR, this should permit use of the method to much larger shutdowns than observed in this case.
2. The analysis of the data will be fairly simple to program on a small computer like PMACS.
3. The integrated effect of pulsed source causes an increase in neutron level which could cause a period scram. (This occurred in one case with PCTR.) This effect appears to be the same as that which occurs when a normal startup source is inserted. If the output of the pulsed source varies greatly from pulse to pulse, however, (as was true in the PCTR experiment), this effect could be a recurring one and cause difficulties.
4. The measurement of $\alpha (= k\beta/\lambda)$ can be made quite rapidly with good precision even in the presence of rather large backgrounds and thus should be quite useful for lifetime measurements which, however, would be characteristic of the entire, quite complicated, reactor arrangement.

HTLTR Design

Detailed designs of the two pile oscillators that go with the reactor were reviewed this month. The light duty oscillator design is now nearly identical to the heavy duty oscillator design. According to the architectural engineer, who is designing most of the details of the two oscillators, the design and construction costs are lower for two similar oscillators. A larger and more versatile light duty oscillator will thus be obtained at no increase in cost.

Other parts of the detailed design, including the steel reactor case, doors and insulation, parts of the neutron spectrometer assembly and the motorized flux traverse wire reels, were reviewed for conformance with experimental needs.

HTLTR Driver Fuel Design Calculations

Reactor multiplication of the HTLTR has been calculated for several conditions of driver fuel enrichment and temperature. Three conditions of the test cavity have been investigated, and the end-to-end flux-flattening requirements are being studied. The values of multiplication when calculated in a rectangular two-dimensional model have been compared with those calculated using an area-equivalent one-dimensional model. Effective group cross sections were calculated by methods which not only are most explicit, but which also produce good results when incorporated into a computation of the k_{∞} of a tested, natural uranium, graphite-moderated lattice. At present, the Gd requirements to compensate temperature defects are being determined.

FUEL CYCLE ANALYSIS

Calibration of the MELEAGER Burnup Code

Some initial computer calculations, using the THERMOS-HRG codes to determine effective cross sections for mixed oxide fuels, UO_2 - PuO_2 , in light water moderated lattices with varying degrees of moderation, have been completed. These calculations were done at a moderator temperature of 20°C--the temperature of the scattering kernels on the present THERMOS library. A new THERMOS library with scattering kernels at 300°C is being generated. These same test cases will be re-examined at this higher temperature.

As a check on the ability of the THERMOS-HRG codes to handle low-moderator-to-fuel ratios, an attempt was made to reproduce the experimentally determined value of the resonance escape probability, p , for uranium-water

lattices, published in BNL-486. Table I presents the reported values of p from BNL-486 and the values determined by the HRG and MELEAGER codes. Figure 1 represents a graph of these data. Although the curves do not appear to be in good agreement, the largest deviation from the reported value of p is only 6%

TABLE I

(NO $1/v$ CONTRIBUTION OF CROSS SECTION INCLUDED IN p)

V_m/V_f	P_{BNL}	P_{HRG}	$P_{MELEAGER}$
4	0.909 ± 0.008	0.893	0.904
3	0.868 ± 0.003	0.861	0.877
2	0.841 ± 0.004	0.805	0.829
1	0.720 ± 0.027	0.679	0.720

($1/v$ CONTRIBUTION OF CROSS SECTION INCLUDED IN p)

	P_{HRG}	$P_{MELEAGER}$
4	0.884	0.888
3	0.851	0.856
2	0.792	0.800
1	0.657	0.669

The values of p reported in the BNL document also had the fast effect in U^{238} removed. The exact procedure for removing the fast effect portion of the U^{238} cross section from the resonance escape probability when the HRG cross section data are collapsed down to a one-group cross section is not entirely clear. For this reason, these cases are being recalculated to remove the high energy (greater than 0.1 MeV) portion of the U^{238} cross section.

If the data from HRG and MELEAGER are replotted with the $1/v$ portion of the U^{238} cross section included in the value of p , Figure 2 is obtained. From these two curves, it is evident that the only difference between the HRG value of p with the $1/v$ portion of the cross section removed and the corresponding value of p from MELEAGER is the value of the $1/v$ part of the cross section in these two codes.

The difference in the value of the $1/v$ portion of the U^{238} cross section in the two codes is the upper energy at which the cross section averaging is cut off. In the HRG code, the upper energy cutoff was set at 10

RESONANCE ESCAPE PROBABILITY WITH 1/v PORTION OF U²³⁸ CROSS SECTION REMOVED

FIGURE 1

RESONANCE ESCAPE PROBABILITY WITH 1/v PORTION OF U²³⁸ CROSS SECTION PRESENT

FIGURE 2

2 of 2

MeV while in the MELEAGER code it was set at 0.2 MeV. Thus the HNG code averages the U^{238} cross section over a greater energy range and its $1/v$ contribution to the U^{238} cross section is larger than that obtained in the MELEAGER code.

Fuel Utilization Studies

The VESTA code has been revised to include improved plutonium recycle techniques in thermal reactors. Several changes have also been made to the code to improve the nature and form of the output. For example, the annual cascade separative work requirement is printed out for each period in the hypothesized future nuclear power industry.

Table II shows the cumulative natural uranium requirements for the next hundred years, using the growth curve from the Report to the President⁽¹⁾ and various projected breeder reactor development schedules with the characteristics of a pressurized water reactor (PWR) as parameters. It is assumed that the PWR will supply the nuclear power capacity in the near future and will recycle all available plutonium. When breeders are introduced, all plutonium will be fed to the breeders so that they can be added as rapidly as possible. Later, when the available plutonium is greater than needed for breeder inventories, the PWR's will return to plutonium recycle operation. As seen in Table II, the lower α is for Pu^{239} , the more in situ burning of plutonium is favored and the less is made available for export to breeder inventory. However, enrichments are lower, which tends to offset the effect on total uranium requirements of less plutonium being available for fast reactors. The data reported here are developed with plutonium valued at the thermal reactor value without using an increased selling price of plutonium when fast reactors are assumed to purchase all of the plutonium. Cases with the latter assumption are being run for which uranium utilization will improve as α of Pu^{239} is reduced. Also, increases in specific powers of thermal reactors tend to reduce the uranium usage by virtue of the reductions in reactor inventories. Work is continuing to more fully evaluate the effects of the interactions of parameters and the effects of fuel fabrication costs on uranium requirements. One further point is evident in Table II: The type of breeder reactor introduced has a much greater effect on the uranium requirements than the date of its introduction.

(1) Civilian Nuclear Power - A Report to the President, 1962, USAEC, November, 1962.

TABLE II

NATURAL URANIUM REQUIREMENTS FOR VARIOUS FAST BREEDER INTRODUCTION DATES AND PWR PARAMETERS
(An incomplete listing of cases)

		Pressurized Water Reactor Parameters					
		0.57	0.57	0.51	0.46	0.51	0.57**
Specific Power, MW _t /ton		20.	40.	40.	40.	80.	40.
Fast Breeder		Total Natural Uranium Used by 2060 - 10 ⁶ tons					
Doubling Time, yrs	Introduction Date						
20	1980	2.16	2.14	1.92	1.77	2.04	1.99
7	1990						
20	1975	5.12	6.12	5.56	5.05	4.72	5.40
20	1990	6.02	6.42	6.14	5.83	6.40	6.23
7	1975	1.91	2.03	1.77	1.57	1.97	1.58
7	1990	3.39	2.94	2.73	2.57	3.00	2.80
None (PWR only)		29.83	27.19	23.59	20.58	22.09	27.13
Tons Fuel/MW _t *		.0000343	.0000273	.0000292	.0000305	.0000517	.0000373
Tons Natural U/ton Fuel*		4.314	5.370	4.651	4.204	2.460	4.033
Tons Natural/MW _t *		.000148	.000147	.000136	.000128	.000127	.000151
Gram Pu/ton Natural*		2223.	1970.	2114.	2215.	2218.	2292.
Gram Pu/MW _t for Sale*		0.329	0.289	0.287	0.284	0.281	0.245

* For optimum exposure in a PWR with 25% recycle of plutonium, \$23.50/kg uranium, and \$125/kg fuel fabrication cost.

** Same with \$60/kg fuel fabrication cost.

Nuclear Characteristics of Plutonium Fuels as a Function of Geometrical Concentration

Six RBU, MONTE CARLO, calculations were performed to estimate the alpha of plutonium, the fast effect, and resonance escape probability. The first three runs represented a typical center rod of the Westinghouse 1000 MW_e reactor with 4% enrichment. The first used uniform enrichment, the second used a natural UO₂ core with PuO₂ annulus, and the third used a PuO₂ core with natural UO₂ annulus. The fourth run was a repeat of the third using cross section data which were placed on the cross section data tape this month. The fifth and sixth were typical 0.6-inch-diameter rods, as described in GNL-486 (1957, using 1.3% enriched uranium metal). The fifth represented a water-to-uranium ratio of 1.0 while the sixth represented a ratio of 4.0. The low values of fast effect obtained from the first four calculations (see Table III) prompted the other two as a check on the method. The comparison to experiment is presented in Table IV, which also indicates the fast effect computed in MONTE CARLO is low. Since these calculations are a result of only one hour in MONTE CARLO, this may be a statistical variation; so, additional time will be used to improve this calculation.

TABLE III

MONTE CARLO CALCULATIONS WITH PLUTONIUM

Run Number	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Plutonium location	Uniform	Annulus	Core	Core
Cross section set	1	1	1	2
Plutonium alpha	.555	.536	.503	.523
Flux ratio*	.971	.958	.651	.757
Fast effect**	1.047	1.028	1.026	1.038
Resonance escape**	.575	.622	.658	.568

* Ratio of flux in plutonium region to flux in the cell.

** Based on the reaction rates in U²³⁸, Pu²⁴⁰, and Pu²⁴².

TABLE IV

MONTE CARLO CALCULATIONS WITH 1.3% URANIUM

Run Number	5	6
Water-to-metal ratio	1.0	4.0
k infinity	1.064 (1.119)	1.107 (1.158)
Fast effect	1.064 (1.099)	1.037 (1.042)
Resonance escape	.743 (.72)	.871 (.91)
Thermal utilization	.945 (.946)	.792 (.795)

NOTE: The parenthetic values are derived from experiments as reported in BNL-486.

Some High Conversion Ratio Reactor Concepts Utilizing Metallic Uranium Fuel

To achieve high conversion ratios, fuels must have a high neutron yield ($\eta\epsilon$) and a high thermal utilization (f). With metallic uranium, high values of epsilon (ϵ) are easily achieved by packing the fuel close together. However, closed packed fuel arrangements are not always consistent with a high value of eta (η), especially, where plutonium is part of the initial enrichment, or is formed during irradiation due to the hardening of the neutron spectrum. Reactor designs which allow a clustering of the fuel with separate moderator regions, such as typical D₂O or SGR designs, present the possibility of cluster fuel and a soft neutron spectrum. The D₂O system also leads to especially high thermal utilization.

In this report, three possible reactors with pressure tube designs were considered: (1) D₂O moderated, D₂O cooled; (2) D₂O moderated, H₂O cooled; and (3) Graphite moderated, sodium cooled. The reactors were designed to have conversions ratios in excess of 0.90 at a reasonable fuel cycle cost. In all cases, the exposure of the uranium metal was limited to 4000 MWd/M.T. exposure to ensure fuel durability. The short exposure, in term of conversion ratio, means low loss of neutron-to-fission products and because fuel is changed at short intervals, very little excess reactivity is required, thus, small losses to the control systems. In terms of fuel cost, low fuel exposures generally result in high fuel cycle costs if fabrication and recovery unit costs are high. But the short exposure also leads to high throughput in both fuel fabrication and separations which will lower unit costs and fuel cycle costs.

To achieve a high eta, a soft neutron spectrum is required. Thus, the fuel clusters are spaced farther apart than in typical D₂O or SGR designs.

This increases the moderator cost, especially for D_2O designs. To account for this, the cost for moderator is included in the fuel cycle cost. With large clusters of fuel, flux depressions and a hardening of the spectrum occur in the central fuel area. For this reason, an especially soft spectrum is used along with some enrichment variations to minimize these effects.

The fuel configuration used in all cases consisted of a central rod surrounded by two concentric rings, all clad with zirconium (0.025 thick) utilizing a low cost coextrusion fabrication process similar to NPR fuel. The pressure tube was also made of zirconium to maximize thermal utilization. To minimize reactivity changes, plutonium recycle fuel enriching depleted uranium (0.253% U^{235}) was used. Table V shows neutron balance for each reactor type. The D_2O has a unity conversion ratio whereas the D_2O moderated with H_2O coolant and the SGR design are both at 0.95. The impacts of a high fast effect and soft neutron spectrum show in the neutron yield.

Although these systems do not breed, they do export large quantities of plutonium while consuming about one-half U^{235} in the tails uranium. Thus, if plutonium price is increased, the fuel cost of the system is reduced. The enrichment requirements are very low because the neutron spectrum is very soft and the nonfuel absorptions are low. For this reason, these systems are not very sensitive to the interest rate on fuel. To make a proper comparison between these, the cost of moderator and coolants were included. D_2O was priced at \$0.044/cc (\$20/lb) with a leakage rate of 4%/year for D_2O cooled system and 1% of the H_2O - D_2O system. Graphite was priced at \$0.017/cc with sodium at \$0.005/cc. Table VI shows the fuel cost and coolant-moderator cost for each system under five sets of unit cost condition and fuel prices. The first and second cost condition represents the lowest and highest probable unit costs, respectively. The third case represents a more reasonable intermediate unit cost. The last two cases show the effect of an increase in plutonium credit for each system. The full study includes a wide variety of jacketing and separation costs. The figures presented are for a fully developed close-coupled industry.

TABLE V

NEUTRON BALANCES FOR D₂O AND SGR REACTOR USING METALLIC URANIUM FUEL
 (Basis - 100 Neutron Absorbed in Fissile Material)

	<u>D₂O-D₂O</u>	<u>H₂O-D₂O</u>	<u>Sodium Graphite</u>
Fission Neutron			
U ²³⁵	57.3	50.9	78.1
Pu ²³⁹	124.7	125.3	126.5
Pu ²⁴¹	22.7	27.3	34.0
Total	204.7	203.5	198.6
Net from U ²³⁸	14.2	13.9	18.3
Total	218.9	217.4	216.9
Absorption in Fissile Isotopes			
U ²³⁵	28.1	25.1	19.3
Pu ²³⁹	62.1	53.0	66.0
Pu ²⁴¹	9.8	11.9	14.7
Total	100.0	100.0	100.0
Fertile Capture			
U ²³⁸	91.1	84.9	80.4
Pu ²⁴⁰	9.6	11.2	14.7
Total	100.7*	96.1*	95.0*
Nonfertile Capture			
U ²³⁶	.1	.1	.1
Pu ²⁴²	.4	.5	.8
Other Isotopes	.2	.2	.3
Total	.7	.8	1.2
Nonfuel Absorption			
Samarium and Xenon	5.9	6.1	5.9
Other F.P.	3.4	2.8	1.8
Structure and Moderator	3.5	7.3	8.6
Control Leakage	4.7	4.4	4.3
Total	17.5	20.6	20.7
Total Nonfissile Absorption	118.9	117.4	116.9

* Average conversion ratio.

TABLE VI

FUEL CYCLE COST FOR HIGH CONVERSION RATIO REACTORS USING METALLIC URANIUM (TAILS) ENRICHED WITH PLUTONIUM

Working Capital Rate %/year	Economic Parameters			Operating Cost, Mills/kwh _e								
	Fabrication Cost, \$/kg	Separation Cost, \$/kg	Plutonium Credit, \$/g Fissile	D ₂ O-D ₂ O (26%)*			H ₂ O-D ₂ O (26%)*			SGR (45%)*		
				Fuel Cycle	Moderator Coolant	Total	Fuel Cycle	Moderator Coolant	Total	Fuel Cycle	Moderator Coolant	Total
10	5.00	2.00	10.00	-.046	.234	.188	.201	.166	.367	.212	.038	.250
10	24.00	32.00	10.00	1.770	.234	2.004	1.953	.166	2.119	1.395	.038	1.433
10	10.00	5.00	10.00	.467	.234	.513	.529	.166	.695	.410	.038	0.448
10	10.00	5.00	20.00	-.135	.234	.099	.351	.166	.517	.396	.038	0.434
10	10.00	5.00	30.00	-.549	.234	-.315	.174	.166	.340	.383	.038	0.421

* Thermal-to-electrical conversion efficiency.

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NEUTRON FLUX MONITOR PROGRAM

Two regenerating neutron flux monitors installed five months ago in a test facility at KW Reactor are having difficulty with voltage breakdown. It appears that insulation deterioration is occurring and data are now being obtained intermittently. It appears that completion of planned in-core testing may require fabrication of new chambers. An interim technical progress report was completed.

Insulation of cables used for beta current neutron flux monitors was analyzed through an activation process; it was determined that the insulation was alumina rather than the magnesia as stated on the purchase specifications. The cable manufacturer was contacted and has agreed to replace the cables with the proper magnesia insulation. Analysis of data from the last KW Reactor irradiation indicated that aluminum in the exterior surroundings is contributing a considerable negative current; whereas, the alumina insulation is a significant source of positive current. Another source of positive current was identified as the manganese in the stainless steel center conductor. Because of this strong material dependence on the background currents, consideration is being given to preparing cables of other materials including Zircaloy-2, Inconel, nickel, and nichrome.

Another boron-11 detector chamber is being designed for experimental use. In this case, a symmetrical chamber concept will be employed to attempt to achieve effective cancellation of background currents. For demonstration purposes, graphite will be used in place of boron-11 in the initial attempt. Thus, it will be possible to determine the magnitude of the background currents and the methods to be used for simple measurement and cancellation. Work will be extended to using boron-11 in the chamber. An interim technical report on beta-current generator neutron flux monitors was nearly completed in rough-draft form.

For the microwave neutron flux monitors, no further experimental work was conducted; however, analysis of data from prior experiments was continued in an effort to establish plans for future tests. The necessary hardware is being assembled for an attenuation-type, in-reactor test at KW Reactor. An interim technical progress report was completed covering the microwave neutron flux monitor accomplishments to date.

NONDESTRUCTIVE TESTING RESEARCH PROGRAMElectromagnetic Testing

A study of the current flow around flaws within eddy current test

specimens and the effects of these currents on the components of the magnetic field adjacent to the surface of the test specimen has been started. The purpose of the study is to obtain new information about the electromagnetic fields in the vicinity of test specimens for different flaw conditions. This information will be used as an aid in test coil design and in the preparation of a series of semi-empirical nomograms for use in applying eddy current tests to practical problems.

The multiparameter principle makes it possible to use more information from the electromagnetic field than is possible with single frequency tests. It is thus believed that the multiparameter tester range can be extended by use of new coil assembly design techniques. The present need to extend the limited flaw (hole) diameter range over which the multiparameter tester now functions has been factored into the new study.

Theory indicates that eddy current flow in the general case has a complex direction of flow which for sinusoidal excitation would result in the current elements flowing in elliptical paths. Measurements of IR voltage drops using a miniature voltage probe in the vicinity of a simulated longitudinal "crack" in molten Wood's metal clearly showed this effect. Vectorscope plots of the voltage plane loci of the probe output as a function of probe orientation at several locations show that the current element paths on each side of the simulated crack were elliptical. As expected, no evidence of complex direction of flow was noted for the case of circular symmetry.

Fundamental Ultrasonic Studies

To evaluate the recently developed ultrasonic wave model, waveforms of ultrasonic pulses were recorded and compared to those predicted by this model. In fitting the experimental pulse data to a Fourier series assumed for the wave model, fourteen terms were used in the original trials. The number of terms was subsequently increased to twenty-five in order to better represent the measured pulses. In addition to this change, the computer input data were also modified to avoid overlap of the printed output pulses. Since attenuated pulses became longer in time, care had to be taken to ensure that the time duration for high attenuation did not exceed the time provided for pulse print-out. The input data were augmented by adding time segments, having zero amplitude, at the beginning and end of the input pulses. With these changes, some improvements in pulse shape were obtained.

It is believed that additional experiments will be necessary to determine the cause for the remaining differences between calculated and measured pulses. To facilitate further measurements of attenuation, the influence

of shear wave attenuation on longitudinal pulses and longitudinal attenuation on shear pulses was also studied. It was found that shear wave attenuation has little effect on longitudinal pulses and that the longitudinal attenuation influences the shear pulses only when the longitudinal waves are incident near the critical angle. This indicates that the more difficult shear measurements will not be necessary when determining pulse distortions due only to longitudinal attenuation. Previously it was thought that experimental data on both types of waves would be necessary. Previous experimental data on longitudinal attenuation in aluminum and Zircaloy-2 samples will, therefore, be useful in obtaining pulse distortion measurements and calculations for these materials.

Heat Transfer Testing

Development of a stainless steel foil sinusoidal thermal wave transducer has continued. Current varying between 0 and 120 amperes was passed through the foil, which was sandwiched between a cold plate (-195°C) and the sample under test. Copper constantan thermocouples made of 0.003 in. diameter wire were used to record temperatures at the transducer foil and at the sample surface. An attempt was made to use a 0.005 in. thick air gap between the transducer and sample surface, but buckling of the foil due to thermal expansion caused contact to occur between the sample and foil at a number of points. For this reason, it was necessary to electrically insulate the foil to prevent contact with metal samples. Films of paper and teflon 0.0005 in. thick were used for this purpose.

Approximately sinusoidal thermal waves were generated in samples of teflon, steel, glass, and lead with this prototype equipment. However, variable distortions in the waves occurred, apparently due to variations in gap thickness due to thermal expansion of the foil. A foil transducer with the foil clamped in a spring-loaded holder is now being fabricated. Continuous tension exerted by the springs should prevent buckling of the foil.

The first foil transducer had a liquid nitrogen coolant storage tank and cold plate made of stainless steel. Thermal stresses due to temperature gradients generated in the tank during filling caused distortion which warped the cold plate in the bottom of the tank. A copper tank and cold plate are being fabricated for use in connection with the new spring-loaded foil transducer. The high thermal diffusivity of copper should reduce thermal gradients and warping.

Quantum Physics Devices

The possibility of using a pulse laser for generating high intensity acoustic pulses of very short duration is under study. A memorandum has been prepared on the subject. These calculations show that the effect of the radiation pressure of the laser light pulse produces an acoustic pulse of intensity greater than can be generated by conventional methods. This analysis indicated that the radiation pressure of the light beam is not the predominant effect. For an absorbing medium, a thermo-elastically generated acoustic wave has a greater amplitude than that associated with the radiation pressure. Other workers have reported the coherent generation of intense hypersonic waves with intensities sufficient to damage the target material. These phenomena may have potential value for nondestructive testing purposes.

Theoretical and experimental work is progressing on a device to measure transit acoustic phenomena. A device of this type will be useful to investigate the acoustic phenomena associated with the pulse laser as well as conventional ultrasonic pulses. By measuring the transient acoustic phenomena in the propagating medium, the effect of the receive transducer upon the acoustic information can be eliminated.

The device employs a coherent laser light beam which illuminates a modified Michelson interferometer. One leg of the interferometer contains a transparent cell in which the acoustic pulses are transmitted. The pulse traveling in the medium varies the index of refraction of the material; this effectively changes the path length of one leg of the interferometer. This change in turn alters the light distribution in the fringe pattern. A phototube monitoring the fringe pattern produces a signal which corresponds to the acoustic pulse intensity as it travels through the light beam. By minimizing the diameter of the light beam and magnifying the resulting fringe pattern, in principle one should be able to follow the wave motion of 30 to 50 Mc acoustic phenomena. For a liquid medium, the sensitivity of the device for ultrasonic waves would be about 10^{-4} watts/cm². It is planned that this technique will be investigated as soon as the work can be scheduled on the continuous wave laser.

Nondestructive Testing of Plutonium Metal

A liquid chemical FC-73 (C₈F₁₆O) appears to satisfy all requirements for testing Pu metal by immersed ultrasonic techniques.

Available hood space in the 200 Area was investigated and a transducer manipulating system is being designed for use in a hood. This system will

be versatile and will permit evaluation of all the ultrasonic tests which are planned.

An electrostatic transducer has been used to generate and detect ultrasound in two plutonium alloy test rods. Both the shear and longitudinal waves were clearly defined, indicating the technique may be used in determining the elastic module of plutonium alloys.

The results obtained on the plutonium alloy test rods indicated that the pulse superposition method would be more efficient than the "sing-around" method for velocity measurement. A high repetition rate (40 kc to 50 kc) pulser is being developed which will permit the use of the pulse superposition method for measuring ultrasound velocities in the plutonium. Waveforms observed in the plutonium alloy were similar to those observed earlier in copper, in which case excellent results were obtained with the "superposition" method.

DETECTION OF COOLANT IMPURITIES AND TEMPERATURE MEASUREMENT PROGRAM

A bare cavity was used in the environmental chamber to determine microwave frequency variations as a function of chamber humidity. Over the controlled humidity range from 30% to 100% (relative) a significant frequency change was noted. The actual variation was not measured as plans were already established to provide a considerably more sensitive cavity with included temperature compensation. The design of the new cavity was partially completed.

Investigations were accelerated on the use of infrared techniques for determining the moisture content of gas impurities. Plans were established to utilize a spectrophotometer to measure several known samples.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

A scavenging test (REDOX-4) using the REDOX stack off-gases was conducted in the 0.42 mm/hr snow of January 10, 1965. The washout coefficient measured for inorganic gaseous iodine was $4.2 \times 10^{-5} \text{ sec}^{-1}$, which is nearly the same as measured in the 1.0 mm/hr snows of December 14, 1965 (PUREX-2 and PUREX-3).

Assuming these measurements are accurate, we need to explain why the coefficient in REDOX-4 was not 0.4 times that in the PUREX tests. We noted that the predominant crystals in these two snowfalls were columns and spatial dendrites, respectively, and theorize that the lower terminal

velocity of the columns (and correspondingly longer residence in the plume) provides a higher washout per depth of snowfall.

Work has been completed on an operable MIDAS program for the 7090 computer that will calculate the evaporation of a raindrop as it falls through arbitrary observed temperature and humidity profiles, within the following restrictions:

Temperature	0 - 25 deg C
Height	0 - 6000 m
Diameter	0.1 - 4.4 mm

The program is capable of handling 50 changes in the slopes of the temperature and humidity profiles during the droplet's fall. It may be used to calculate variations in rain spectra, radar reflectivities, and washout coefficients with elevation.

The readout gives height, diameter, rate of change of diameter and of mass, and terminal velocity. Provision is made for introduction of down or updraft velocities of the air.

Detailed analysis is now completed on the final two field diffusion tests using fluorescein and fluorescent zinc sulfide. Equal masses of these two tracers were simultaneously released from a common low-level source. The agreement in masses collected over downwind sampling arcs was generally good; the ratio of fluorescein to sulfide varied from 0.83 to 1.24 even though the mass varied over two orders of magnitude. The discrepancies observed should not greatly limit the use of this dual-tracer technique out to two miles.

Radiological Physics

Another whole body counting trip was made to Alaska to count the Eskimos at Anaktuvuk Pass. The average adult body burden, 637 nCi, was 10% lower than it was last November but it was 40% higher than it was in January 1964.

An experiment to study the human metabolism of technetium was started in cooperation with the University of Washington. A shadow shield counter was transported to the University of Washington and is being used to make linear differential scans of radioactivity along the body and to determine the whole body retention of the isotope with time. By using the 60-day half-life Tc^{95m} , measurements can be made for at least one month whereas previous studies by other investigators had to stop at three days because of the short-lived isotope they were using. These studies will be helpful

in determining the behavior in the body of all technetium isotopes, in particular the long-lived Tc^{99} which will be separated in large quantities from radioactive wastes at Hanford. These studies will not be complete for another month.

Calibration of the plutonium counter for lung burdens was made using distributed sources in plastic foam inserted in the lung cavities of a Remcal phantom. It was found that differences in distribution can cause errors of as much as 30% and differences in isotopic composition can change counting rates by as much as a factor of two when using the 17 keV X-ray peak. This indicates that an accurate measurement of the plutonium burden must be accompanied by a measurement of the isotope composition of some of the excreted plutonium. Methods to eliminate the errors caused by non-uniform distribution are being studied.

A heat baffle was incorporated into the small calorimeter for measuring alpha-beta radiation by completely surrounding both cells by 1/8" of copper. This greatly improved the performance. The temperature variation is only $\pm 5 \times 10^{-6}^{\circ}C$ which corresponds to $\pm 5 \times 10^{-7}$ watts.

The capacitor adaptor of the pulse X-ray machine was repaired and the unit can now be pulsed satisfactorily. A high frequency oscillation on the leading edge of the pulse display was found to be due to an electromagnetic disturbance in the oscilloscope and is not part of the pulse. The unit is now ready to be calibrated.

The positive ion accelerator is now back in operation after being down several weeks after flood waters entered the building. The accelerator had to be completely realigned and recalibrated and this work has been completed.

The precision long counter, which gives accurate measurements of a neutron flux having random directions, gives errors of ± 20 percent for flux coming at small angles (0° - 90°) from a $D(d,n)$ reaction target. This is probably due to non-uniform illumination of neutrons across the finite face of the counter due to the sharply peaked angular distribution of neutrons from the target. An attempt will be made to determine the exact cause in future experiments.

A calibration cask, which provides a reproducible thermal flux, was built for the He^3 neutron spectrometer. This will permit checking the He^3 counter for changes in efficiency and for bias adjustments of the anticoincidence counters. The final calibration of the spectrometer is being made in preparation for making field spectrum measurements.

A computer code for analyzing data from optical discrimination experiments was written. The code calculates the channel number, height, and the width of a gaussian curve to fit the pulse height distribution of the transmitted light.

Instrumentation and System Studies

Circuit development was essentially completed for the blood pressure measurement channel of the animal physiological function and radio-nuclide uptake telemetry system. A linear modulator was designed to convert the amplified bridge output voltage to dc as required to operate the voltage-controlled sub-carrier oscillator. A simulated blood vessel is being used to test operation of the channel. In addition, a miniature 500 Vdc power supply was designed for use with the GM tube in the thyroid uptake measurement channel. Fabrication began on all four channels of the system.

One developmental electrometer circuit was fabricated for the instrument to be used to measure both the charge and velocity of falling raindrops. The electrometer uses an insulated gate field-effect transistor as the input stage and an input resistance of 10^{11} ohms in parallel with 3 pF capacitance was demonstrated. A 6-inch long by 1-inch diameter brass cylinder will be used as a "pass-through" detector for the raindrops; charge induced on the cylinder will be amplified by the electrometer circuit. Raindrop velocity will be determined through the use of a second cylinder with the time of passage measured by similar induction methods. Problems involving rainfall on windy days remain to be resolved.

Investigation regarding an animal activity monitoring system indicates that a matrix of metal plates, wherein a capacitance variation is detected, will not provide a satisfactory solution due to completely inadequate resolution. The use of infrared detectors is now being considered for monitoring the small caged animals.

Development was initiated on a solid state circuit, using insulated gate field-effect transistors, for use with ionization chambers of the 500 cm³ to 1000 cm³ size.

A low noise, charge sensitive voltage preamplifier circuit was developed using an insulated gate field-effect transistor in the input stage. The unit is non-inverting, provides a voltage gain of 12, and has a rise time of about 60 nsec. The circuit will be the basis for development of a complete low noise pulse amplifier.

Circuit fabrication was continued on the single electron counting system to be used with multiplier phototubes in dosimetry studies. Most of the circuits, except the low voltage power supply, were completed.

Analysis of the coordinate transformed meteorology wind turbulence data is nearly complete. The transformed data tapes were speeded up by a factor of 16 during transformation and the analysis of the data at three different tape playback speeds gave results of wind variations from as low as .01 cps to as high as 20 cps. Two portions of the analyses were completed; that of determining the variance and the covariances of the three wind components, and the power spectra determination. Yet to be completed are two analyses consisting of covariance spectra and cross correlation spectra determinations.

Additional computer runs were made to update the biological radioisotope system analog model for Tellurium¹³² and Iodine¹³². These parent and daughter radioisotopes were injected into the bloodstream in the standard man model yielding curves that appeared typical. A complete paper tape program was punched which allows this problem to be quickly set up.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

Isotopic analyses were provided on program samples as received during the month. The mass spectrometer for this program was shut down during the last half of the month in order to install the new ion source and ceramic vacuum-lock sample-changer. A paper on the vacuum-lock system was accepted for presentation at the ASTM Committee E-14 Conference on Mass Spectrometry in May, 1965.

The analytical study of ion and electron trajectories in the scintillation-type ion detector was continued.

EXPERIMENTAL REACTOR PHYSICS FACILITIES

PRCF

The D₂O conversion and the outage items for the H₂O irradiated fuel handling capability were successfully completed. A total of 17,616 pounds of 99.78% D₂O water was added to the moderator system following a 1000 pound rinse. Composite moderator samples indicate there has been no degradation of the heavy water since its addition early in the month.

A load-to-critical experiment was performed using PRTR type Pu-Al fuel elements in the D₂O moderator. Critical was achieved with a total of seven of the 19-rod clusters surrounded by the six safety mechanisms in the reflector. Calibration of the flux instruments, moderator, and safety mechanisms is in progress.

Work on Project CAH-119, which includes the work area and office wing addition to the PRCF, has been temporarily terminated with exception of the PRTR basin extension. No significant operating time has been lost due to construction interference.

PCTR Operation

Operation of the PCTR continued routinely during the month. The experiment to measure the relative worths of special Pu-Al and PuO₂-UO₂ fuel rods was completed. The Phoenix fuel experiment was started. The uniformity of the boron content of the borated polyethylene disks was measured and the loading of the fuel assemblies (Pu-Al disks, polyethylene disks and borated disks in aluminum tubes) was completed. Loading of the assemblies into the reactor was started.

TTR Operation

The TTR was operated on a one night a week basis for the University of Washington Graduate Center.

A group of coaxial cables was irradiated to investigate the possibility of identifying the insulation used inside the stainless steel sheaths. Analysis of the induced radioactivity was shown to be a successful non-destructive testing technique.

Critical Approach Facility

The Critical Approach Facility was not operated during the month.

COMPUTER FACILITIES

A roll of PRTR punched paper tape was processed on the G.E. 412 computer using an edit program, which corrects punch errors and tapes non-uniformities. The corrected tape is to be sent to EDPO to determine the extent to which PRTR data processing can be speeded by preliminary editing on a small computer.

Circuits to simulate a low-speed data transmission system were designed and installed between the 321 Building data logger and the G.E. 412.

This link will be used to develop methods for the acquisition and reduction of data time-shared with concurrent process control programs. A conference was held with engineers of Northwest Bell Telephone Company to learn the characteristics of available data transmission systems, particularly the Bell dataphone equipment.

Analog computer utilization was as follows:

<u>EASE 1132</u>	<u>EASE 2133</u>	
136	72	Hours Up Time
72	96	Hours Scheduled Down Time
<u>0</u>	<u>0</u>	Hours Unscheduled Down Time
208	168	Hours Total

The EASE 1132 uptime includes 40 hours of swing shift usage. Problems considered during the month were:

1. Meteorology Study.
2. N Reactor Conversion Study.
3. ANN Optimization Study.
4. Simulation of a High Temperature Chemical Reaction System.
5. Model Development for Biological Study.

The operational checkout was completed on both computers after their move to 3201 Building and they are now essentially in full operation. Improved air conditioning system of the new facility has reduced ambient noise.

The use of semiconductor diodes and varistors (Thyrite) to simulate bidirectional flow in piping systems was investigated briefly. There is a need for simple devices which provide an approximate square or square-root characteristic with an accuracy compatible with calculated frictional losses in pipes. Studies of multisection fluid header flow rates and pressures such as are encountered in the N Reactor secondary coolant system would be considerably simplified if simple two-way fluid resistance could be obtained. Studies to date indicate that relatively inexpensive operational amplifiers used with Thyrite varistors and linear resistive elements will be suitable for this purpose.

Assistance was given to personnel of the Graphite Research and Development Unit in programming for the MIDAS program a simulation of high temperature chemical kinetic equations involving reactions between hydrogen ions, hydroxyl ions, hydrogen, carbon dioxide, carbon monoxide, and water.

Nine of the ten sessions were completed on the winter quarter of the Electronic Computer Applications class. During the past month, each student spent an average of four evening hours solving problems on the EASE 1132 analog computer.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Consultation was provided to 1) Engineering Development, Reactor and Fuels, on the magnitudes of surface solar energy and energy released by thunderstorms, and to 2) Industrial Hygiene on the sludge removal and waste transfer facility (244-AR vault).

Weather Summary

<u>Type of Forecast</u>	<u>Number Made</u>	<u>% Reliability</u>
8-Hour Production	84	83.0
24-Hour General	56	85.3
Special	105	90.5

February was much warmer and drier than normal. The month was also windier than usual with notably high speeds occurring on the 5th, 8th, and 27th.

Mass Spectrometry

Analyses were provided on three plutonium samples of material used in critical experiments with PuO_2 -polystyrene systems.

Instrumentation and System Studies

Investigation was started for Fabrication Development of Ceramic Research and Development regarding suitable methods for measuring PuO_2 grain sizes in PuO_2 - UO_2 fuel element samples. The use of an electron microscope appears to be feasible; however, consideration is also being given to the use of a modified "metallograph."

Fabrication was partly completed on a gamma monitor that is to be used in moving ground vehicles by Environmental Monitoring. All bench tests were successfully completed on the required solid state circuitry.

A study was made of the fuel rupture monitor and contamination monitor requirements of the proposed Advanced Test Reactor high temperature gas

loop at the request of Metallurgy Research. The radiation level experiences at DR Reactor (Loop No. 1) and at the General-Atomics loop at GETR were considered. Informal recommendations regarding the ATR requirements were provided. Further engineering advisory effort appears probable on this problem.

Fabrication was completed on a solid state prototype low level current-to-frequency converter proposed for use in the HTLTR instrumentation. The instrument has six input current ranges from 10^{-9} to 10^{-4} A, where in each case, the full scale value equates to 10^5 pulses/sec. The "zero" pulse rate for constant temperature and for short term measurements is only 6 pulses/sec on the most sensitive range. Input currents as low as 10^{-11} A can be measured to $\pm 1\%$ accuracy on the 10^{-9} A full scale range. The accuracy for 10^{-12} A is $\pm 10\%$. Thus, in effect, the instrument readable coverage extends from 10^{-12} to 10^{-4} A. Further developmental work is planned in an effort to reduce the zero offset and improve stability.

Basic flow charts were prepared for the programs for the HTLTR Programmed Measurement and Control System (FMACS). Work continues on developing the flow charts in greater detail.

A relay multiplexer is being designed for the Containment Systems Experiments to switch temperature and pressure signals into analog-to-digital converters at high speed. Initially built to handle 160 inputs, the unit can be readily expanded to receive more points as required. Specifications were written for four analog-to-digital converters to be controlled and accessed by a digital computer. The combination of several converters under computer control provides an economical method of gathering pulse height and time-of-flight data.

Design was completed on an interface to allow computer control of a mass spectrometer and a cost estimate of required equipment was compiled. One manufacturer's high-speed counting equipment was evaluated and found to be unsatisfactory in its present form; however, there appear to be other suppliers whose systems can meet the specifications.

Development work continued on a data processor to be used with the micropositioner readout portion of the creep capsule data logger. The function of the data processor is to register and store the existing digitized analog quantity each time the digitizer is interrogated. The new data processor, when complete, will replace an old Coleman processor that uses mechanical relays. Mercury wetted relays and a simplified wiring layout in the new unit will improve the reliability of the system. Fabrication is 60% complete.

Work continued on testing and fabrication of a wattmeter transducer assembly to be integrated with the creep system instrumentation at the KW facility. These transducers, when installed, will measure the power dissipated in each of the capsule heaters. Fabrication is 20% complete.

Development work continued on an instrument to measure the small mechanical displacements of a metallurgy test specimen inside a high temperature furnace. Fabrication of a high voltage, radio frequency oscillator-amplifier has been completed. A new nitrogen filled transducer was excited with 300 volts (RMS) at 28 megacycles. Preliminary testing indicated the output of the transducer (d-c volts) was linear with the displacement of the external sliding ring. The resolution of the transducer will be limited only by the resolution of the readout instrumentation. A sensitivity of 14.5 volts per millimeter was noted over a 15 mm displacement range. Further effort in this project will be aimed at tuning the output of the excitation source to the transducer using a two-foot transmission cable. A radio frequency voltmeter will also be integrated into the system.

Failure of an electrolytic capacitor in a Dressen-Barnes 300 volt power supply caused a malfunction of the Minneapolis-Honeywell data logger at KW. The logger has operated properly since the fault was corrected. As a precautionary measure, other electrolytic capacitors in the power supply have been replaced.

At the request of Equipment Development Unit of Reactor and Materials Technology Department, an evaluation was run on a model SD-101A variable frequency filter. The unit is built by Spectral Dynamics Corporation of San Diego. The unit is specified to have fixed bandwidth ranging in fixed steps from 2 cps to 500 cps and a frequency range of 2 cps to 25,000 cps. The accuracy of the bandwidth was not determined specifically but the frequency range met the specifications. Gain as a function of frequency was checked and found to be as specified. A given spectrum that is speeded up by means of tape recordings and then observed at the higher frequencies also reproduced as expected.

Two meetings were held with N Reactor Fuels personnel to discuss the possibility of applying systems analysis and modeling techniques to fuel element fabrication problems. Two areas suggested as possible targets are the induction-heating mechanism in the brazing operation and the extrusion processes. N Reactor Fuels personnel are now preparing more detailed information on these processes. This information will be used as the basis for determining whether modeling techniques are applicable to the proposed process improvement studies.

Work is continuing on the design of the nondestructive testing station for the waste calcination tanks. As presently scoped, the station will monitor tank wall thickness and temperature, and will read out on either an X-Y recorder or a typewriter printer. An auxiliary, contact type, ultrasonic probe will be provided for miscellaneous in-cell testing. An oscilloscope will be used for general checking and readout while a selsyn indicator will provide pot position information. Two ultrasonic detectors will be used on the waste tank system; a high frequency transducer for vessel wall thickness and a low frequency transducer for analysis of pot contents. Temperature measurement will be attempted with both a contact thermocouple and a radiant detector. Present plans are to utilize the tank welding mechanism for vessel rotation and to build an added drive mechanism to provide for vertical transducer travel.

A liquid level probe, installed in a waste storage test tank, was used successfully to indicate liquid level during calcination. A device was developed to convert the oscilloscope readout for presentation on a strip chart recorder. Although some drift was encountered, the recorder operated well during the three-day calcination period.

A study is being conducted to determine the feasibility of employing nondestructive techniques to measure, with $\pm 1\%$ accuracy, the thickness of plutonium deposited on a metallic substrate. A preliminary analysis of the measuring problem was conducted and the development of appropriate tester concepts has begun. It appears that the primary design criterion will be high accuracy while stability, simplicity of operation, and reliability will be strong secondary requirements. A prototype eddy current system is being assembled from available equipment to analyze the factors affecting accuracy and stability.

Two ultrasonic methods for performing this measurement are also being investigated. One method, using a pulsed oscillation, through-transmission technique, was evaluated unfavorably due to the extremely short pulse transit time involved.

A second approach was found to offer fair prospects of working. A simple single crystal pulse echo technique, using wide-band equipment, will resolve echoes spaced as closely as about 0.3 microsecond apart; this is equivalent to about 10-1/2 mils of Pu. Possibly, this resolution could be improved to about 0.2 microsecond or about 7.2 mils of Pu.

Optics

The N Area traverse equipment has been calibrated and is ready for service. The traverse equipment for the old reactors was modified to use

the new IRC readout equipment. This unit was also calibrated and the new equipment has been transferred to Irradiation Testing Operation of IPD.

Debugging of the computer program used to calculate displacements from the traverse data is continuing. A document is being prepared which describes the traverse mechanisms, develops the theory of operation, and explains the Fortran program used to make the calculations.

The profilometer at 105-C is used for diameter and warp measurements of irradiated fuel elements. Battelle-Northwest has been asked to modify this instrument by adding photocells to provide automatic diameter and warp readout. Photocells to be used for this purpose have been tested in an optical system similar to that of the profilometer. This test has been demonstrated to the customer.

Assistance has been given on the design of a special spectrophotometer to be used to monitor the progress of chemical separations in a hot cell. A complete specification of lenses, mirrors, lamps, slits, prisms, and photomultipliers has been provided together with information on spacing and positioning the various components.

Factory acceptance tests were made on the new NRC vacuum system, and the system has been received and installed. Installation tests are complete and the system has been put into routine operation.

Some problems are being experienced with the lens design program. Two sample problems were run. The first sample problem ran successfully. The second problem did not. The trouble should soon be corrected, and the program made ready for routine lens design.

During the four-week period covered by this report, the following work was performed:

1. Two Redox crane periscope heads were repaired.
2. A Purex crane periscope head was repaired.
3. The fuel basin periscope for PRTR was repaired.
4. The fuel examination borescope for PRTR was repaired.
5. A Lenox borescope for reactor process tubes was repaired.
6. Several front surface mirrors were fabricated.
7. A plastic sphere was silvered and gold coated.
8. Two glass seals were fabricated.
9. A TV mount and lens adapter was fabricated.
10. A boron oxide block was ground and polished.
11. A glass plate was scribed with 1 mm divisions.
12. Two shutters were repaired.

Physical Testing

Seamless Inconel tubing is scheduled for installation in the sixth cell at N Reactor. Representative samples from seven heats, both initial hollow and finished tube, were nondestructively tested. All indications were metallographically inspected. None of the flaws exceeded three mils in depth.

A three-week time study was made using ultrasonics to detect formation of air bubbles in 10 mil thick water columns. Bubbles in the water introduce errors in the neutron dispersion measurements. None were found.

A holder for point focus type transducers was made for testing various diameter tanks or pipes. It eliminates having to reshape the holder or change the transducer. Its versatility should prove successful in monitoring metal surfaces with various geometrical shapes.

Quality control of stress rupture and tensile specimen material was continued with ultrasonic inspection of both Inconel-X, Series 600 rod, and Hastalloy-X, Series 200 plate.

Field radiography was provided to eighteen different customers indicating an increase in demand for this type of service.

Fluorescent penetrant tests include those made on reactor nozzles, boron stainless steel control rods, 8,516 N Fuels braze rings and a boric acid tank to detect surface discontinuities.

Field ultrasonic thickness measurements were taken on equipment in PRTR cells A and B as part of the first and second quarter odd-year pressure vessel survey. In addition, measurements were made on a 300 Area crib waste trailer and a titanium tank in the 321 Building for detection of corrosion.

A total of 33 tubes were eddy current tested in steam generator 1A during a recent shutdown. Fifteen tubes had inside indications not in evidence during the tests made in 1963. A split tube was located, verifying that suspected from a visual check.

Engineering services were provided in investigating a faulty dilatometer furnace program controller. A temperature lag was found to exist; the sensing thermocouple was imbedded in an aluminum oxide rod, which was good thermal insulator. The problem was corrected; the thermocouple was brought to the surface of the rod.

INSTRUMENT EVALUATION

Fabrication progress by an offsite contractor was reviewed regarding eight hand and shoe contamination monitors, six of the combined alpha-beta-gamma type and two for alpha-only monitoring. Full test procedure information was given to the contractor and several tests were demonstrated. The instruments are nearly completed.

Long-term operational evaluation tests were continued on a commercial battery-operated portable radiation survey meter made by Physics International Company of Berkeley, California. General operation continues to be acceptable, in particular for alpha survey work. Similar tests are in progress on an air proportional type alpha detection instrument manufactured by International Telephone and Telegraph Corporation. Breakdown difficulties with the air proportional probe are apparent and are increasing in frequency.

Evaluation testing was completed on a "Hilam" high level gamma area monitor of the Eberline Instrument Company. The unit has two ranges of 0-10³ and 0-10⁴ R/hr. Final calibration errors were determined to be + 5% to + 23% on the low range and -3% to + 6% on the high range. The instrument employs a momentary contact switch for energizing the detector. During activation at 10⁴ R/hr, the reading was noted to decrease at a 0.5%/min. rate.

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CHEMISTRY DEPARTMENT
RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Overheated N-Reactor Fuel Study

Two additional irradiated co-product target elements were subjected to rapid thermal transients to simulate the effect of overheating followed by delayed emergency cooling. The two elements were heated to 935 C in individual experiments over a period of about 5 minutes then quenched immediately with a flowing stream of 66 C water. In both tests the elements failed and the core material flowed away from the element at a temperature near 800 C. About half of the metal displaced from the molten core is believed to have been washed or otherwise quickly ejected out of the furnace tube to the downstream catch tank by the action of steam and water during the quench. This was evidenced by a mat of metal particles conforming to the bottom and sides of the catch tank. Cooling was rapid enough to solidify a portion of the aluminum alloy and bond the Zircaloy jacket to the Vycor furnace liner. Virtually all of the core alloy had flowed from the jacket during rupture and quench. A major reaction between jacket and core metal to form the higher melting $ZrAl_3$ did not occur. This was in contrast to the observation of significant $ZrAl_3$ formation in earlier heating tests with radiative cooling alone. Swelling prior to rupture increased the diameter by about 0.04 inch.

A cooperative study was initiated to establish internal bursting pressures of target elements under temperatures expected in loss-of-coolant incidents.

Reactor Studies

Bromide ion was added to the process water used in one-half of H Reactor and the Br-82 formed was measured in the effluent water from both halves of the reactor to determine the extent of cross mixing. These data are important in assessing the validity of half-reactor tests. If no channels of cross mixing are available other than the reactor front and rear face piping, about 6% of the bromide ion is found in the control side. A correction factor must be applied to all half-reactor tests to take this cross mixing into account.

SEPARATIONS PROCESSES

Disposal to Ground

Only minor changes, most of them showing decreasing concentrations, were observed in radioisotope analytical results for the ground-water monitoring program during the current month.

The source of low-level radionuclides in several wells near the Yakima River is apparently the river itself. Isotopic analyses of water samples from these wells show the presence of natural uranium and potassium-40. The Yakima River is somewhat higher in uranium content than are project ground waters.

Plutonium Aerosol Studies

Two additional plutonium rods were ignited at about 600 C in air flowing at 3 cm/sec. The objective of these tests was to determine the particle size distribution of the oxide when plutonium ignition occurs at higher temperature. On a weight basis less than 2% of the oxide consisted of particles smaller than 44 microns in diameter. Particles smaller than 15 microns accounted for less than 0.24%. The size distributions of the large particles were essentially the same as obtained from earlier ignitions.

One observation of interest was that oxidation could be terminated after ignition by cooling with a current of air. The cooling of the air appeared to be sufficient to lower the temperature below the point required for oxidation.

Thorium Dissolution Studies

As part of the study to develop a flowsheet for co-dissolution of aluminum cladding and thoria core, the influence of high concentrations of aluminum on the dissolution rate of thoria was determined. In boiling, vigorously agitated 12M HNO_3 -0.025M HF, the rate of penetration of Sol Gel thoria particles is reduced by a factor of about 10 as aluminum concentration is increased from 0.1M to 0.64M. When the hydrofluoric acid concentration is increased to 0.08M, a penetration rate in 12M HNO_3 -0.65M $\text{Al}(\text{NO}_3)_3$ equivalent to that in 12M HNO_3 -0.1M $\text{Al}(\text{NO}_3)_3$ -0.025M HF is attained. Use of the high hydrofluoric acid concentration may not be practical from a corrosion standpoint.

Acquisition of data on the range of nitric acid, hydrofluoric acid, mercuric nitrate and nickel nitrate concentrations over which activation of aluminum can be attained was completed. Aluminum dissolution rate studies are in progress.

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Polonium Chemistry

Two more potential polonium extractants, di-n-octylamine and di(sec-butyl) phenyl phosphonate, each representing a class of compounds, were examined. Maximum extraction of polonium by 10% di-n-octylamine is at 6M HNO_3 with a polonium-bismuth separation factor of six. Ten percent di(sec-butyl) phenyl phosphonate in xylene gave distribution ratios of less than 0.1 for both polonium and bismuth. Neither compound appears as good as dibutyl carbitol. A report covering flowsheets using the latter compound has been written by Richardson, Sheppard and Guay (BNWC-44).

Polonium Recovery Development

Pulse column development was continued on the bismuth-polonium solvent extraction process described last month. The work this month demonstrated considerably improved performance with organic-continuous column operation with respect to capacity and dispersion when compared with aqueous continuous operation. Special flowsheet modifications were made to permit estimation of nitric acid HTU's in the extraction and stripping contactors. The HTU's (overall raffinate basis) were about 1 to 1.3 ft. in both columns. In addition, the bismuth scrubbing efficiency was measured in the scrub column using 2M HNO_3 as scrub. The scrub denatination factor was 280, giving a bismuth HTU of 1.6 ft., in line with that previously reported with 0.1M HNO_3 scrub. The overall bismuth DF was 1.2×10^4 in the simulated polonium product streams.

In laboratory studies batch contacts simulating the extraction, scrub and strip columns of a proposed flowsheet for separating polonium from bismuth (dibutyl carbitol as solvent) were performed using uranium (U-233) in place of polonium. Favorable distribution ratios indicate uranium may be used as a stand-in for polonium in Cold Semiworks pilot-plant studies. Additional nitric acid extraction data were obtained to aid in defining process conditions.

As part of the development of an aqueous process for recycling bismuth at Hanford, successful pilot plant runs with the 14-in. spray calciner showed good denitration of $1.52\text{M Bi}(\text{NO}_3)_3 \cdot 5 \text{H}_2\text{O}$, 3.2M HNO_3 feed at 3.0, 4.0 and 5.0 gph at 750 C wall temperature. The bismuth nitrate contained 1.2, 1.1 and 1.4% residual nitrate respectively. Calciner wall deposits were low with no sticking observed.

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WASTE MANAGEMENT AND FISSION PRODUCT RECOVERYSemiworks Support - Manganese Removal

Two alternate processes to the controlled pH, D2EHPA solvent extraction process for manganese removal from rare earth fission product concentrates have been scouted in laboratory scale experiments. One is based on rare earth oxalate precipitation and the other on sulfate precipitation. The more promising sulfate process was carried through hot-cell testing (with Semiworks feed stock) and has now been used in the plant, apparently with quite satisfactory results.

The development of a process was complicated by lack of a centrifuge in the Semiworks, which required use of decantation for phase separation. The final process, which is not pH sensitive, involved:

1. Rare earth precipitation by addition of $5M NaHSO_4$ followed by 15-minute digestion at 80 C and one hour for precipitate settling.
2. Decantation (drainage of the tank through a dip leg, followed by a $1M Na_2SO_4$ wash).
3. Metathesis with $3.5M NaOH$ (80 C).
4. Dissolution with HNO_3 -HEDTA. The HEDTA is required because some residual sulfate and lead remain because of inability to completely drain the tank.

Dissolution is followed by batch extraction of the total rare earths into D2EHPA followed by cerium oxidation and trivalent rare earth stripping with an oxidizing solution of $2M HNO_3$, $0.2M Na_2S_2O_8$, $0.02M Ag^+$. Preliminary results from the plant indicate 98% rare earth recovery through step 4 and a manganese decontamination factor of 50 (which is adequate).

Strontium Semiworks Solvent Treatment

Previous work showed that washing degraded D2EHPA-TBP-diluent solvents with HNO_3 - $KMnO_4$ solutions restores cerium(IV) extraction capacity. It is desirable to solubilize solid manganese dioxide formed in this treatment to aid organic-aqueous phase separation. Addition of oxalic acid, citric acid or hydrogen peroxide to the aqueous phase prior to phase separation reduces manganese to soluble Mn(II). Oxalic acid appears to be the most satisfactory of the three reagents.

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Persulfate Solubility

In both laboratory and plant cerium removal (via cerium oxidation and D2EHPA extraction), potassium persulfate ($K_2S_2O_8$) has been used as oxidant, and the relatively low solubility of the compound has required large solution volumes. The solubilities of both the potassium and sodium salts were measured (as a function of HNO_3 concentration) and $Na_2S_2O_8$ found to be much more soluble than $K_2S_2O_8$ in all cases. Solubility (at 25 C) of $K_2S_2O_8$ in 0, 1, 2 and 4M HNO_3 was 0.22, 0.264, 0.285 and 0.37M, whereas $Na_2S_2O_8$ was soluble to the extent of 2.4M in water and 1.7M in 2M HNO_3 . Hot cell performance of the two chemicals was identical. Use of $Na_2S_2O_8$ is accordingly recommended. Besides much lower solution volumes and improved operating flexibility, a saving in reagent cost may also be realized.

Cobalt D2EHPA Extraction Behavior

The production of Co-60 as a by-product of uranium irradiation (via use of a thin cobalt-containing "sweater") was recently suggested. The D2EHPA solvent extraction behavior of cobalt was briefly investigated to ascertain the feasibility of its recovery from Purex waste. The cobalt extraction coefficients (E_a^0) increased from very low values at $pH \leq 1$ to over 100 at pH 5 to 6. Acetic, tartaric and hydroxyacetic acid (added as possible complexants) had very little effect on its extractability while EDTA and HEDTA effectively suppressed the extraction at all pH 's and citric acid was intermediate in its effect. It appears that cobalt can be readily extracted and stripped and that separation from other metals should be straightforward.

In other studies, a wash with 2M HNO_3 , 0.2M $K_2S_2O_8$, 0.02M Ag^+ was shown to regenerate Cold Semiworks D2EHPA solvent which was used in cerium pulse column studies.

Clean U-233 Studies

Continuing tracer-level experiments were aimed at determining: (1) the effect of nitrate on the uranyl sulfate anion exchange process (for recovering ultra-pure U-233 protactinium decay product); and (2) the role of "unadsorbable" Pa species in Vycor adsorption.

Target elements of cold-pressed spray-calcined thoria (67% theoretical density) were prepared for reactor irradiation as a "proof of principle" and, more importantly, to see whether irradiation will have any adverse affect on the excellent dissolution kinetics of spray-calcined thoria. A series of additional calcination runs is being made in the 8-in. spray calciner to: (1) explore further the effects of calciner variables on

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product quality (particularly cold-press density and ability to form a sol); and (2) test the calcination behavior of the thorium nitrate product of the recent 6-ton Purex processing campaign. The latter effort (calcination of Purex thorium nitrate) will also provide material needed for certain radiological measurements.

In-Tank Solidification

Tests were completed in the 10-ft. diameter model tank to extend the data on circulation patterns to include the effects of circulator submergence. The general circulation pattern was similar to that observed in earlier studies. The test results show an approximate 40% decrease in circulation velocity as the water level above the circulator is decreased from 8 inches to zero. When the water level was reduced below the top of the circulator, the velocity pattern reversed its expected trend and showed a slight increase above that measured at the point of equal level.

The purpose of these recent tests is to provide information for the prototype in-tank solidification unit which may be operated at various levels of circulator submergence.

Recovery of Neptunium and Plutonium from Purex Plant Stored Waste Sludge

Recovery of neptunium and plutonium from acidified Purex process stored waste sludge (PAS solution) by extraction with D2EHPA-diluent is under study. Plutonium(IV) is extracted into an equal volume of 0.4M D2EHPA-0.2M TBP-Soltrrol extractant. Without the organic phase being removed, a reducing agent is added to the aqueous phase to reduce Np(V) to Np(IV). On recontact of the two phases neptunium is extracted without appreciable stripping of plutonium. Hydrazine alone will not reduce Np(V) to Np(IV) in PAS solution at 25 C. Either ferrous sulfamate (0.01 to 0.05 M) or ferrous sulfamate (0.01 to 0.05M)-hydrazine (0.01-0.05M) will reduce Np(V). Greater than 99% extraction of both neptunium and plutonium from simulated PAS into an equal volume of solvent was attained with either of these reducing conditions.

Cerium-Rare Earth Partitioning

The Strontium Semiworks currently uses a batch persulfate oxidation-D2EHPA extraction process to separate cerium [as Ce(IV)] from the other rare earths. A continuous process has been proposed for this operation using the 1A column to extract the rare earths into the solvent (0.4M D2EHPA-0.2M TBP in a n-paraffin hydrocarbon diluent) and a persulfate strip in the 1C column to selectively remove the trivalent rare earths from the solvent. Cerium would be left in the solvent as Ce(IV), to be

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removed later by batch washing with a nitric-nitrous acid solution. "Cold" pilot plant runs, using a 3-in. diameter pulse column with a 16-ft. tall cartridge (having plate geometry similar to that of the Semiworks 1C column)- have been made to test the feasibility of the process. Most of the runs have been made with the 1CF heated to 60 C; with the aqueous phase continuous; with a strip solution containing 2M HNO_3 , 0.2M $\text{K}_2\text{S}_2\text{O}_8$, and 0.02M AgNO_3 ; and with a 1CF/1CX flow ratio of 5/1. An organic scrub stream (1CS) was introduced at the bottom in runs in which the feed was introduced above the bottom. The flow ratio was then modified to give 1CF/1CS/1CX = 4/1/1. The primary variable studied so far has been feed introduction point. The following cerium DF's and rare earth losses were obtained, with Ce-144 tracer used to follow the rare earth behavior:

Feed Point, Inches Above <u>Bottom</u>	Feed Temperature, C	Cerium DF	% Rare Earth Loss ^(a)
0	25	3	-
0	60	6	-
22	60	≥ 12	-
30	60	≥ 27	0.5
102	60	135,190 ^(b)	1.1

(a) Measured in separate runs with no $\text{K}_2\text{S}_2\text{O}_8$ and AgNO_3 present.

(b) Duplicate runs.

EQUIPMENT AND MATERIALS

Rejected 304-L Stainless Steel Pipe

Several lots of 2-in., schedule 80, 304-L stainless steel pipe intended for use in the Purex annular dissolvers were rejected because results of Huey tests (made off-site) were borderline. Corrosion test specimens from the various lots were sensitized (1250 F for 1 hour and water quenched) and exposed to boiling 65 w/o nitric acid in the multisample tester. Corrosion rates for these samples, measured after 665 hours of exposure, varied from 6.6 to 20.7 mils/mo. Most of the attack occurred on the inside surface of the pipe samples. The corrosion rate for a sensitized "standard 304-L" coupon exposed along with the pipe sections was 0.6 mil/mo.

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Non-Metallic Materials

Two polypropylene filter materials were exposed to boiling 60% nitric acid. A 100-micron woven cloth (B-3403 National Filter Media) disintegrated in less than 1 day. A mechanically felted material (Troy Blanket Mills) disintegrated in about 10 days.

Two graphite bearing materials (Graphitar 145C and AP-80) were not damaged during exposure successively to 2 M HNO_3 -0.1 M HF and 20% TBP- CCl_4 solutions. Exposure to each solution was about 200 hours at 50 C and 120 hours at ambient temperature.

Pump Development Studies

Development studies are being performed on a variety of liquid and slurry pumps for pilot and production plant service. Chempumps with heat exchangers installed in the bearing recirculation line have been tested for the pumping of hot, radioactive, corrosive slurries. Modifications to the originally installed seals have been completed and the life of the seal faces has been extended well over three times their original life, with no evidence yet of degradation. New bearings of boron carbide have been ordered to further improve life.

Tests with water as the test fluid have been conducted on the Integral canned motor pump. Operation has continued for approximately 1700 hours at a flow rate of 5 gpm against a head of 40 psig with no appreciable wear resulting on the pivoted shoe, modified Kingsbury thrust bearing. A specialized jet type pump manufactured by the Deming Pump Company for handling of corrosive hot light slurries has been under investigation, following modifications to improve performance with slurries. Although operation is continuing, insufficient data have been compiled for final evaluation.

Preliminary design evaluations have been completed for a pump required for moving concentrated wastes from one underground storage tank to another in the in-tank solidification program. A model pump has been specified for test. This pump is a composite unit combining the best features of several vertical slurry pumps.

Computer Process Control Studies

The feasibility of using the GE-412 computer for process control studies of remotely located systems, utilizing telemetering techniques, was investigated. The reference system for the study utilizes a data logger to scan and digitize 20 instrument signals at a rate of 2 signals per

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second in a fixed sequence. The digitized signals can be transmitted via a telephone data transmission sub-set using a voice quality telephone circuit on a dial-up basis. A device to simulate the telephonic transmitting equipment was built and installed on the data logger currently used to scan the experimental C-column control instrumentation. Using the simulated transmitter, the programs that have been written to re-assemble and process telemetered data are being evaluated.

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REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM UTILIZATION PROGRAMSalt Cycle Process Studies

Several laboratory attempts have been made to prepare 250-gram batches of 20 w/o PuO_2 in UO_2 by electro-codeposition from chloride melts. Thus far, the attempts have been unsuccessful although material of this composition had been made previously on a much smaller scale. The work has demonstrated the need for very careful control of the electro-codeposition process.

Salt Cycle Process Control Studies

Two approaches to the problem of instrumenting a salt bath for Salt Cycle process control are being studied. In one approach, measurement of the "back emf" of the electrolytic cell gives promise of serving as a guide to the PuO_2/UO_2 ratio in the cathode deposit during co-deposition. Preliminary investigations give reproducible results in the range of > 0.5 w/o PuO_2 . In measurements to date, the "back emf" has proven to be independent of variations in total current, current density, and sparge gas composition. Anode-to-reference potentials suggest that the variations in melt conditions which control the composition of the deposit also control the cell potential. Further investigations will determine the effects of plutonium and uranium concentrations in the melt on the cell potential.

In the other approach, a new light pipe photometer sensing unit for operation in a salt melt has been designed. The present system uses a GE No. 1962 quartz light bulb, encased with its leads in a hermetically sealed tube containing helium and immersed in the melt, as a light source. A prototype source has been maintained at 600 C for over 2 weeks with no sign of deterioration.

Close-Coupled Fuel Processing Economic Study

A report entitled "Economic Evaluation of Close-Coupled Fuel Processing with Plutonium Recycle" has been completed and will be issued as BNWL-28.

This report investigates the relative fuel cycle costs (and elements thereof) associated with two alternative reactor fuel cycles employing light water moderated reactors and either central plant reprocessing-refabrication or close-coupled reprocessing-refabrication. Three reactor types, a typical water reactor with Zr clad fuel, a typical

water reactor with SS clad fuel, and an advanced pressurized water reactor with SS clad fuel were examined. Two fuel management concepts were considered: one with several reactors at a single site with staggered fuel discharges averaging 1 to 3 months between discharges; and the other, a single reactor discharging fuel at 12-month intervals.

During the terminal stages of this study a scope design and preliminary cost estimate was made of the construction cost of a close-coupled plant serving 1000 Mw(e) reactor capacity and providing the process functions of reprocessing, remote fabrication of mixed oxide fuels and UO_2 to UF_6 conversion. Approximately \$6,200,000 was indicated.

One of the principal findings of this study was that the fuel exposure for a close-coupled plant cycle optimizes (depending on input data) at 15,000 to 20,000 Mwd/ton resulting in an annual throughput of 40,000 to 50,000 Kg/year per 1000 Mw(e) installed capacity. This optimum is largely independent of close-coupled plant investment. Since the close-coupled plant shares, with most chemical operations, a great economic sensitivity to throughput, increases in throughput rapidly escalate profits and incentives, but lower throughputs result in the opposite effect. The study results indicate that a \$6.2 million close-coupled plant serving a 1000 Mw(e) reactor complex can be competitive with central plant processing. Less reactor capacity (or lower throughput) can eliminate the incentives but more reactor capacity (or higher throughputs) can increase incentives markedly.

The annual fuel cycle savings (as compared to a central plant cycle) with the \$6.2 million close-coupled plant, and exposures of up to 20,000 Mwd/ton in a 1000 Mw(e) reactor ranged from \$0.6 million to \$1.6 million 0.09 to 0.22 mills/kwh, depending on reactor type and using the high range costs and high decontamination factor input data. Based on low range cost inputs and a close-coupled plant decontamination factor of 6, the maximum annual savings were \$0.3 million, 0.037 mills/kwh, and in the worst case required the close-coupled plant investment to be reduced to \$4.9 million to break even. It should be noted that the savings quoted are over- and above-normal profits, amortization, taxes, insurance, etc.

The incentive for close-coupled processing is substantial if fuel exposure is limited to 15,000 Mwd/ton in the central-plant cycle (possibly as high as 0.4 mills/kwh) but drops off sharply at higher fuel exposures.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENTHot-Cell Glass Experiment and A-Cell Renovation

Decontamination of A-Cell (following completion of the full-level Hot Cell Glass Experiment and the earlier pilot-plant and pot calcination studies) is proceeding in a very satisfactory manner. An invaluable innovation in equipment removal has been the remote use (with manipulators) of a portable rotary saw, with alundum abrasive cut-off wheels, to cut almost all of the equipment (including the spray calciner) into pieces of suitable size for disposal as dry waste. Result is virtual elimination of personnel radiation exposure and scheduling problems which would otherwise have been involved. Following removal of most of the equipment the cell was further decontaminated to a low level (prior to personnel entry) by vacuum cleaning and high-pressure spraying with a series of suitable decontaminating solutions. This operation was made possible (and practical) by the smooth integral stainless steel liners and "hot" floor drains which were an important design feature of the High Level Radiochemistry Facility. Although full-level Purex waste from several tons of irradiated uranium had been calcined in the cell (and there had been numerous leaks and general gross contamination), only 2 months of day-shift effort were required from termination of the glass experiment until entry into the cell. The importance of stainless steel cell liners (vice painted concrete or mild steel) can hardly be overemphasized.

In other work, leaching studies are continuing on the glass samples from the Hot Cell Glass Experiment, and the final report is being written.

Systematic Glass Studies

Work is proceeding on the determination of the glass-forming characteristics of the sulfate-free Purex waste system. One series of melts has been prepared for this purpose. The drip temperatures of these were found to be lower than the corresponding sulfate-bearing compositions. Solubilities, however, were only slightly lower, both the sulfate-containing and sulfate-free glasses showing a minimum at $M^+/P = 2.3$. The sulfate-free system shows an additional minimum at $M^+/P = 2.1$.

Preparation of an interim formal progress report on the phosphate work to date is about half completed and will soon be issued.

Preparation of Glasses

A series of melts containing simulated Nuclear Fuels Services (NFS) waste and phosphate was prepared. As the metal ion equivalent to phosphate

ratio in the melts increased from 1.7 to 3.1, the fusion temperatures increased from about 960 to 1040 C. Appearance of the solid products changed from glassy at the 1.7 ratio to rock-like and somewhat non-homogenous at the 3.1 ratio. All melts appeared sufficiently fluid at 1050 C to pour satisfactorily.

Coupled Spray Calciner-Melter Solidification

The range of phosphoric acid concentration which produces an operable solidification flowsheet was tested in two runs with simulated 80 gal/ton U 1WW containing 0.4M and 0.63M H_3PO_4 ($M^+/P = 2.9$ and 1.8). The lower phosphoric acid concentration ran smoothly during a 12-hour run at calciner wall temperatures of 700 C and a melter susceptor temperature of 1125 C. The feed rate was maintained at 5 gallons per hour. Filter pressure drop reached equilibrium at 15 inches of water during the first 3 hours. Melt product was a gray crystalline solid with a density of 3.2 g/cc. When the phosphoric acid concentration in the feed was increased to 0.63M ($M^+/P = 1.8$) in the second run, the calciner filters showed excessive pressure drop within an hour. Adjusting the filter blowback time cycle from 5 minutes to 2 minutes and switching to repeated manual blowback did not reduce the filter pressure drop. The melt was an olive-green crystalline solid with a density of 2.9 g/cc.

The operability of phosphoric acid-bearing 1WW has now been studied in spray calciner-melter runs with concentrations ranging from 0.4 to 0.96M PO_4^{3-} ($M^+/P = 2.9 - 1.1$). Phosphate ion concentrations from 0.4 to 0.52M ($M^+/P = 2.9 - 2.3$) result in workable feeds causing minimal calciner wall deposition and filter pressure drop (~ 15 inches water). All produce free-flowing melts at 950 - 1050 C which cool to dense crystalline solids.

Solidification of simulated 1WW waste was tested in the coupled spray calciner-melter by the separate addition of 0.5 inch borate glass balls to the melter where they fused and dissolved the calcined 1WW. A two-phase glass with an additive-to-waste calcine weight ratio of 1.6:1 was formed. The glass consisted of a dark brown glassy phase and a lighter yellow crystalline phase tentatively identified as sodium chromate.

The extremely high viscosity of the melt at 1050 - 1100 C ($\sim 200-250$ poises) caused very slow drainage of the melter and prevented continuous calciner-melter operation at 3 gallons per hour of 80 gal/ton U 1WW. High viscosity was caused by glass balls of unknown boron concentration rather than the 100% borax glass ($\text{Na}_2\text{B}_4\text{O}_7$) as previously supposed.

Waste Solidification Engineering Prototype

Design verification tests were continued with the completion of a 12-inch pot calcination test with a TBP-25 waste as feed. The evaporator and fractionator were coupled to the pot calciner and overall equipment performance was good. Minor flow difficulties are being corrected. Power and water outage was experienced during the run without incident other than delay. The pot contained a 5-inch draft tube for promoting circulation and an experimental pulse reflectometer for ascertaining the liquid or foam level. Twelve hundred and thirty-five liters of feed were calcined in 86 hours of furnace operation. Pot feeding was continued until "off-on" feeding began after about 46 hours of operation (average rate of 27 liters/hour). Performance of the pulse reflectometer was excellent. A continuous steam purged weight factor dip tube was used as a check on the reflectometer and was also successfully used for automatic pot level control.

Data Logging of Waste Solidification Test Runs

Three experimental runs with the Waste Solidification prototype were successfully data logged with the GE-412 computer. The runs included a simulated waste process run, the acid fractionator and evaporator test, and the pot cooling test runs. Twenty-five analog input signals were monitored and converted to engineering units for log out and for use in the special calculation routines. Included in the calculations were periodic averages of furnace power, material and flow balances, conductivity measurement, off-gas accumulation, liquid levels, steam flow, and overall processing rates. Evaluation of the furnace power calculations pointed up the need for recalibration of the instrumentation measuring power input to each zone. A statistically-designed calibration procedure was provided for this purpose. The computer operated continuously during the several tests without failure, although building power transients caused temporary failure of the peripheral typewriters.

Intermediate Level Waste Treatment

A pilot scale run was completed to demonstrate the use of a horizontal plate filter (Sparkler Model HRC-10) with a steam stripper and two bed anion-cation ion exchange system for decontaminating alkaline condensate. Feed rates to the stripper, filter plate and ion exchange beds were 1 gal/min, 1 gal/min/ft² and 4 gal/min/ft², respectively. The filter operated continuously for 64 hours. The filter effectively removed the particulate matter in the waste.

All radioisotopes were reduced below their respective MPC_w values except Ru-106, which was 1.8 MPC_w. The Ru-106 exceeded its MPC_w value after

about 100 bed volumes through the anion exchange column. The Ru-106 breakthrough coincides with chloride ion breakthrough. Since nearly all of the chloride is introduced by the addition of hypochlorite, anion bed performance is expected to be improved under plant conditions where hypochlorite or chlorine treatment would not be necessary (added in pilot scale tests to kill bacteria in feed storage system).

CONTAINMENT SYSTEMS EXPERIMENT

Major Facilities and Equipment

Chicago Bridge and Iron Company, contractor for the three-part containment vessel, requested and was granted an extension of the contract to March 30, 1965. The extension request was based on further delays in receipt of the material from the steel supplier. Also, the installation of the bulkheads and reinforcement steel of the wet-wells took longer than was estimated, even though the construction crew was increased to speed production.

Struthers-Wells Corporation reported that they are on schedule with the design of the reactor simulator vessel. The stress analysis performed on the first design revealed several high-stress areas. The design of the vessel is being revised at present to eliminate the trouble spots.

Installation of the laboratory equipment was started with the goal of having beneficial use of the laboratory by April 1, 1965. Purchase of the material and equipment for the auxiliary systems to be located on the canyon-deck level and crane-cab level was started.

Mathematical Models

The blowdown portion of the digital program was altered to include the effect of vessel sensible heat and to indicate temperature gradients in the vessel walls. Results indicate that severe gradients may develop in that portion of the vessel in which water is retained after the blowdown (gradients perhaps as great as 260 F/in). These calculations will be refined to take into account the various regimes of boiling that may occur at the surface.

Program Planning

An analysis of the interrelation of instrumentation sensitivity and testing time was made to aid in developing tests whereby the leakage rate of the containment vessel is determined by the rate of change of the differential pressure manometer used in the reference vessel method. This method, first used in last year's test of the PRTR leakage rate,

shows great potential advantages in decreased testing time; and it may also permit less attention to changes of air temperature and humidity during tests of only a few minutes duration

Fission Product Simulation

Two runs were made during February in the Aerosol Development Facility. In both runs stainless steel clad UO_2 specimens irradiated to 2×10^{17} nvt were melted as a fission product aerosol source. In one case the furnace atmosphere was pure steam; in the other it was a mixture of steam and air. The furnace effluent aerosol was blown into a 1.5 cubic meter tank containing a 1:1 mol ratio of steam and air at 1 atmosphere pressure.

Samples were taken from the tank as a function of time in order to obtain measurements of "air-borne" fission product concentrations, concentrations in the steam condensate, iodine forms, deposition rate on various materials, condensation nuclei concentration, and organic composition and concentration in the gas atmosphere. In addition, grids were exposed for electron microscope pictures. The samples are in the process of being analyzed. No firm conclusions can be made until the results are available, but a very tentative observation is that the iodine from "real" fission product release sources did not behave noticeably different from that from earlier synthetic source releases. No visible fumes were deposited in the furnace tube with the pure steam atmosphere; this is in contrast to abundant fume deposition in air atmosphere.

Two "cold" Zircaloy-clad UO_2 specimens were melted to see whether the water-cooled quartz furnace method of heating was suitable. No problems were encountered due to this different cladding material, but it was noticed that the cladding oxidized in place and retained its cylindrical shape. Four unirradiated specimens were melted in steam or steam-air atmosphere for as long as 17 minutes. The quartz furnace tube was damaged in each case due to insufficient cooling but did not fail during the heating period.

OFF-GAS DISPOSAL TO GROUND

An analysis of errors in flow paths and times due to: (1) ignoring compressibility of the gas in obtaining the potential pattern, (2) using finite difference approximations, and (3) excluding the well casing effects was outlined and is being made. The effect of excluding the well casing appears to cause an error of $< 1\%$ in potentials. It is expected that the error due to ignoring compressibility will be somewhat larger but tolerable. The error due to using finite difference approximations is expected to be quite small.

Analysis of a gas flow problem is under way for homogeneous soil in which the horizontal permeability is greater than the vertical permeability.

SHIPPING HAZARDS

Experiment designs were developed for heating tests of specimens of calcined and vitrified wastes. In this study the release of specific isotopes from various waste calcine and glass formulations will be measured. Equipment required for the study was being located, fabricated and assembled.

COLUMBIA RIVER STUDIES

A submergible television camera was proven to be of value in Columbia River sediment studies. A very clear picture of the river bottom can be seen on the screen in the boat with the camera submerged to depths greater than 70 feet. This closed circuit system will be useful in locating silt deposits and in characterizing the types of sediments to be found on the river bottom.

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BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The program "Steady State Flow Through Soils" was evaluated to determine the inherent error due to the approximation of continuous derivatives with discrete segments used in the solution of finite difference equations. Also, the error magnitude related to averaging permeabilities about the central node in solution of flow through heterogeneous soils was determined. Statistical analysis, using exact closed-form solutions for comparisons, showed a standard error of difference of 2.56×10^{-5} due to the derivative approximations and a standard error of difference of 8.88×10^{-4} due to permeability averaging. These relatively small errors lend confidence to the nearness of approach to exact solutions for this particular program.

A tracer test was run in three piezometers which are sand-packed in well 699-10-E12. The purpose of the test was to determine qualitatively the interflow of water to a pumped middle tube from tubes above and below. Fluorescein and nitrate traces were added to the upper and lower tubes, respectively, while the middle tube was being pumped at a rate of about 10 gpm. Fluorescein was detected after pumping about 1100 gallons. Nitrate analyses are not yet completed. Comparisons of the concentrations of the two tracers in samples should provide evidence of the relative amount of water entering from above and below the pumped section.

Geologic mapping west of the Hanford Project up to the Yakima Firing Center disclosed that the four Priest Rapids basalt flows pinch out within a short distance. The stratigraphic section thus thickens rapidly toward the Pasco Basin center, and the Pasco Basin on the west is structurally more sharply downwarped than earlier believed.

Activation Analysis

Four samples of basalt taken from different locations in two characteristic basalt flows were analyzed by neutron activation, simple chemical separation, and measurement of multidimensional gamma ray spectrometry. Preliminary values of Sc-46, La-140 and Eu-152m indicate significant differences between the basalt flows may exist large enough to provide a characteristic identification technique. More samples are being analyzed for a greater variety of radionuclides to verify the value of this method.

Soil Chemistry

The mineralogical composition of the clay fraction from surface horizons of two Hanford soils was determined by X-ray diffraction analysis. One soil represents surface soil material in the valley south of Gable Mountain; the other was collected near the Columbia River in the vicinity of 100-F Reactor. Clay fractions were separated by a centrifuge procedure. The fractionation followed treatments which removed exchangeable divalent cations, free carbonates, iron oxides and organic matter.

Semiquantitative analysis of minerals showed that montmorillonite is the dominant clay mineral. Vermiculite, mica, chlorite and traces of kaolinite also were detected.

RADIOLOGICAL AND HEALTH CHEMISTRY

Radiation Chemistry

Additional measurements of the ESR spectra of the product from the dithionite reduction of p-nitroaniline confirmed that both it and the radical obtained by γ -radiolysis of p-nitroaniline in air-free solutions are the same. A similar comparison of the spectra from dithionite-reduced p-nitrobenzoate and the γ -radiolysis of p-nitrobenzoate showed that the radicals obtained by these two procedures are not the same. As yet no structure has been found with reasonable splitting constants to give a synthetic spectrum which matches that obtained experimentally by γ -radiolysis. Interest in this radical is considerable since it has a half-life of 8 minutes in oxygen-free water and its structure could give important clues as to what features stabilize a radical.

Germanium (Li) Detector Studies

Gamma ray spectra of gross fission products of U-235 and Pu-239 have been accumulated as a function of time after bombardment for the period of 4 to 32 days. The spectra are very similar but with some notable differences. The principal observed differences are the much higher yields of Ru-103 and Cs-136 which are observed in the fission of Pu-239.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Aerosol Sampling Studies

The study of subisokinetic sampling errors was terminated. A first draft of a report of this work was completed.

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Turbulent Deposition

Markedly improved precisions in deposition measurements were achieved through using very narrow size range of particles prepared by the spinning disc generator. Some important observations were made from experiments to accurately determine the deposition velocity, K , for particles moving in a 0.5 inch diameter tube. The length of tube from the entry point required to develop fully turbulent flow conditions was shown to be of the order of 75 tube diameters for the 1/2-inch tube used. This was much longer than anticipated and suggests that initial measurements of deposition velocity for larger than 1/2-inch diameter tubes may have been determined for the entry conditions rather than for fully developed turbulence. Beyond the entry section, the derived exponential equation for decreasing concentration with distance fits the data nearly perfectly. Of even greater significance is the higher dependence of deposition velocity on average flow velocity. For a correlating parameter, ϕ_{S+} , of less than about 100 the deposition velocity, K , is dependent on a higher power of the average air velocity than the 5th power initially determined from a composite of all earlier data. Beyond S^+ of about 100 the velocity dependence is considerably less. The improvement in accuracy achieved may require revision of the models previously held for turbulent deposition. Studies on larger tubes and particles are planned.

Fallout Studies

A major increase in the Ba-140 concentration of air was observed on two air filter samples collected from January 25 to February 1, and from February 1 to February 5. An increase in Co-60 was also noted for these two samples. These increases may be from the USSR underground test reported to have been exploded on January 15, with considerable venting to the atmosphere.

ISOTOPES DEVELOPMENT - O8 PROGRAM

PROMETHIUM ISOTOPIC FUEL DEVELOPMENT

Promethium Ion Exchange

Two column runs were completed to determine the effect on degree of rare earth separation of substitution of sodium hydroxide for ammonium hydroxide for adjustment of eluant (EDTA) pH. The effect, if any, was very small. These runs complete the comparative study of effect of elutriant choice (EDTA, DTPA or HEDTA) and operating conditions on promethium purification. A final formal progress report is being prepared.

Compatibility Studies

Following the preliminary screening studies reported earlier, a large number of samples have been prepared for more intensive testing of the compatibility of proposed promethium core compounds with candidate cladding materials. Neodymium oxide (Nd_2O_3), which has been through the same chemical preparation steps that will be used for Pm_2O_3 (to insure a proper distribution of trace impurities), is being used as stand-in core material in these long-duration heating tests. Metals and alloys under test at 1100 C include Hastelloy-C and -F, 304L and 310 SS, Haynes-25, Inconel, Incaloy-804, Ta, Ta-10W, W, W-25 Re, Mo, Pt, Carpenter-20, and TZM. The coupons are pressed in Nd_2O_3 compacts in mullite crucibles and the latter are then sealed in Vycor tubes, under a 150 mm He atmosphere. (Purpose of this pressure of helium is to provide mechanical support to the Vycor at 1100 C)

Equipment is being assembled for higher temperature (up to ~ 2000 C) testing of Ta, Ta-10W, W, W-25 Re, Mo, and Pt with Nd_2O_3 and Sm_2O_3 . The samples will be packed in graphite crucibles which will serve as the susceptors for induction heating in a vacuum induction furnace. A somewhat larger number of metals and alloys will be tested at intermediate temperatures (ca. 1500 C).

Additional metals and alloys will be added to the test program as samples become available. Samples will be removed from the furnaces after various periods of time and subjected not only to metallographic examination and measurement but also to mechanical testing and micro chemical analysis (to determine the identity of diffusing species, new phases, etc.).

Besides the above experimental program, thermodynamic calculations have been made in an effort to determine the probability that various conceivable clad-core reactions will occur. The free energies of reactions of the type

where M was Cu, Ni, Mo, Fe, W, Cr, Ta, Ti, Zr, etc., were in all cases positive, and generally strongly so, implying that the reactions should not occur to a significant extent (i.e., equilibrium should be far to the left).

Compound Properties

A first attempt at direct experimental measurement of the surface tension of neodymium fluoride (by the sessile drop technique) has given a value

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far in excess of the expected value. More refined equipment is being readied to study this in more detail, as well as to measure the thermal emittance of rare earth metals and compounds.

Re-entry Burn-Up and Cool-Down

A part-time effort has continued to be devoted to computer studies of re-entry hazards to guide our clad and core development effort. Calculations appear to be in reasonable agreement with those of General Electric MSVD, Hittman Associated, Martin Nuclear, and Sandia. Whether or not burn-up will be achieved is found to be highly dependent on the initial conditions assumed. These include not only the dimensions of the capsule and the particular clad and core materials but also the assumed velocity, angle and altitude of release.

The computer program calculations bear out the belief that promethium metal will burn up more easily (or more completely) than strontium fluoride (SrF_2), because of its higher density and lower melting point. Whether burn-up will occur in either case will depend greatly on the clad. Thus, a 2-inch x 6-inch promethium capsule clad in Rene-41 will be only partially burned up for an expected exposure altitude of 75,000 meters, descent angle of -1° , and velocity of ≤ 6700 meters/sec. With lower melting clads, the situation is much more favorable (however, the lower melting clads may be ruled out because of launch pad abort safety considerations). Even with lower melting cladding materials, burn-up cannot be relied on for lower-altitude, lower-velocity release conditions. Intact re-entry accordingly appears to merit serious consideration.

In other calculations, the impact velocity and capsule temperature were calculated, as a function of capsule size, for assumed release at a speed just below Mach 1 at 12,000 meters altitude. The promethium (metal) was assumed to be at the melting temperature at time of release and both isotopic heating and aerodynamic cooling were included in the computations. It was found that impact velocities and temperatures will be fairly low and the core solid in all cases considered (right circular cylinders up to 3 inches in length and diameter). Impact velocities range from 40 to 70 meters/sec and surface temperatures from 300 to about 600 C.

Prototype Heat-Source Fabrication Equipment

Completion of design of the equipment items required for promethium prototype heat source fabrication, writing of specifications, placement of orders, and liaison with the shops has continued to receive top priority and to occupy considerable time during the month. Most

designs are now in the shop, and delivery and assembly is confidently expected by the time decontamination of A-Cell is completed.

As a result of conversations with the Chemical Processing Department of the General Electric Company and the Division of Isotopes Development of the AEC, the promethium purification capability of the rework columns, which are a part of the A-Cell package of equipment, was evaluated. (The rework columns were initially provided to remove daughter samarium and other impurities from reclaimed prototype heat sources.) If a feed no higher in total ionic impurities than that produced by the Semiworks is available, purification capacity should equal the requirements of the experimental prototype program (estimated as about 3 megacuries per year beginning FY 1966). With certain relatively minor equipment changes, capacity could be increased several-fold. An important feature (besides capacity and reliability implicit in a manipulator operation) of A-Cell for promethium purification is the fact that its heavy shielding precludes the need for prior cerium removal.

Pneumatic Impaction Process Development

All bolts on the 1220-F Dynapak machine were torqued to the specified level. Preliminary tests of frame mounted strain gages indicate that this system has promise as a method for measuring punch pressure during impact.

In other activities, design of the can heating and loading mechanism and scope of the mechanism for remotely changing the punch and die were completed. Cerium oxide and cerium fluoride have been produced with the multipurpose precipitation-filtration equipment from a nitrate feed solution. The suitability of the product for feed to the pneumatic impaction process is currently being evaluated.

Characterization of the Dynapak impaction machine performance is under study. Initial emphasis is on measurement of stresses using cemented strain gages and a memory oscilloscope for readout. Eight successive impactions of a steel cylinder resulted in 100 microstrains initially, increasing to 300 microstrains, in the lower platen. The initial strain pulse occurred over a time interval of about 2 milliseconds, corresponding to a frequency of 250 cps; additional higher frequency vibrations were also observed. Preparations are being made for additional measurements using strain gages and accelerometers with readout by means of a light beam oscillograph.

M. T. Walling, Jr.
Manager

Chemistry Department

MT Walling:cf

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BIOLOGY DEPARTMENT

A. ORGANIZATION AND PERSONNEL

P. G. Haeg, Technician, joined the Metabolism Section on February 5th.

W. R. Wiley, Research Scientist, joined the Radiocology Section on February 8th.

L. G. Smith, Technician, transferred from the Metabolism Section to the Experimental Animal Farm on February 8th.

B. P. Neal, Technician, joined the Metabolism Section on February 23rd.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - O2 PROGRAM

Columnaris

Monthly samplings of fish from Patterson, Hanford, and Priest Rapids continue to show a general titre about 1:50 and a 70% frequency of fish having titre. However, an extreme variation was noted in samples taken from Patterson one month apart, 67% in January and 28% in February. Samples of 33 Hanford fish taken during February show a slight suggestion of higher titre and a higher frequency of fish with titre.

From a group of over 100 fish taken from the Priest Rapids Dam fish ladder a 6% incidence of columnaris recovered from gill scrapings was obtained. This compares with a value of 8% from the previous year. None of the fish appeared to be suffering from the disease nor to have excessive number of organisms on the gills. Columnaris was not recovered from any fish from any other site during this winter period. No titres have been observed in white fish.

Reactor Effluent Monitoring

Temperature units to hatching, defined as the sum of the daily water temperatures above 32 F from fertilization to hatch, are inversely related to temperature, i.e., warmer the water, smaller the temperature units. Among the salmonids, temperature units are clearly species dependent, but it may also tend to be a racial characteristic. Three lots of eggs from local fall run chinooks, incubated 2 F above river temperatures in 0, 2, and 4% effluent, hatched at 1,028 temperature units. Egg lots incubated at 4 F above river temperatures in 2 and 4% effluent hatched earlier than lots in colder water but temperature units of 1,012 were similar. The local chinook stock appears to require about 1,000 temperature units to hatch, compared with about 900 observed in past studies on Puget Sound chinook stock.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Effects of Water Temperature on Fish

Tissues from trout acclimated to 5 C were sampled for electron microscopy and histochemical studies to compare subcellular elements of fish acclimated to different temperatures. In addition, samples were taken from exercised and unexercised trout to study how muscular activity and temperature affect the distribution of glycogen in the cells of various tissues.

Phosphorous - Zinc

After an oral dose of 9 $\mu\text{Ci Zn}^{65}$ and 33 $\mu\text{Ci P}^{32}$, whitefish retained about 1% of each isotope at nine days post administration. Highest concentrations were found in the gastrointestinal tract which contained 0.75 $\mu\text{Ci P}^{32}/\text{g}$ and 0.073 $\mu\text{Ci Zn}^{65}/\text{g}$. Lowest concentrations were found in the muscle which contained 0.002 $\mu\text{Ci P}^{32}/\text{g}$ and 0.003 $\mu\text{Ci Zn}^{65}/\text{g}$.

Strontium

One pig born to a sow that received 625 $\mu\text{Ci Sr}^{90}$ daily during gestation died at two years of age. This animal received extrauterine Sr^{90} from the sow's milk and from a food pellet for one month following weaning. Examination of peripheral blood obtained 12 hours before death revealed a white blood cell count of $>100,000/\text{mm}^3$ and a precipitous drop in the erythroid components and platelets. Histopathological examination of the tissues is not completed.

Rats were conditioned on a semi-purified diet containing 2% calcium and 0.5% phosphorous diet before injection with Ca^{45} and Sr^{85} . Some of the rats were injected one week after conditioning and others at 41 days to determine the excretory patterns at different intervals after isotope deposition.

Cerium

Studies in progress to test the effectiveness of various agents for removing cerium have shown that the chelating agent, deferoxamine (DFA), does combine with cerium in vitro but has no effect on Ce^{144} in vivo even when used in combination with DTPA. The surface active agent, Pluronic F-68, also was ineffective for increasing cerium elimination.

Cesium

Two ewes being fed Cs^{137} daily (1 $\mu\text{Ci}/100$ g feed) have received 300 and 765 μCi over a period of 71 and 68 days, respectively. Body burdens of these animals reached approximately nine times the average daily Cs^{137} intake 16 days after feeding began and have remained relatively constant since then. A third ewe was sacrificed on the fifty-seventh day of gestation after receiving a total of 290 $\mu\text{Ci Cs}^{137}$. Her body burden on the day of sacrifice was thirteen times the average daily intake of Cs^{137} . The placental fluid

and embryonic tissue contained 0.27% of the total body burden, while the genital structure contained 0.14%.

Plutonium

Rats were subcutaneously injected with either $\text{Pu}^{239}\text{O}_2$ or Pu^{239} nitrate and then treated with dimethyl sulfoxide (DMSO) alone and with DMSO containing DTPA to examine their effects on Pu^{239} translocation. No results are available yet.

Inhalation Studies

Long-chain fatty acid synthesis was demonstrated in the mitochondrial fraction of dog lung tissue homogenates. The results were similar to those obtained for rabbit and rat lung. In other experiments it was found that rat lung mitochondria are capable of esterifying fatty acids and degradation studies further confirmed that synthesis of long-chain fatty acids is by elongation of pre-existing fatty acids.

Inhalation studies of strontium fluoride, oxide, carbonate, phosphate and titanate indicate that only the titanate remains in the lung for any appreciable time.

The effectiveness of thiazolidine-4-carboxylic acid in speeding the clearance of I^{131} from rat thyroid was examined further. The optimal time for administration appears to be about three hours after administration of I^{131} . The optimum dose is between 25 and 50 mg/kg which indicates that the chemotherapeutic index is between 4 and 6. Although in these tests the drug only removed about one-third of the I^{131} in the thyroid, the reproducibility of the positive results encourage further tests to increase its effectiveness.

Of six dogs exposed to Fe^{59}O_3 aerosols three showed relatively high deposition in the head. The highest counts were obtained in the areas defined by a vertical plane passing through the laryngeal and pharyngeal cavities. Clearance of Fe^{59} from these head regions was much slower than from the rest of the gastrointestinal tract. This finding may relate to some of the peculiarities noted in earlier studies of pulmonary clearance.

Co-carcinogens

Gross observations of liver tumor induction in rats, maintained on DMAB containing diet, showed that the incidence was increased about two-fold if $\text{Ce}^{144}\text{-Pr}^{144}$ was injected during the first month after the initiation of DMAB ingestion. No significant increase was seen if cerium was administered later than one month after DMAB feeding began.

Secondary Disease Studies

Total body exposure of mice to 500 R is followed by a depression of hemolysin plaques/ 10^6 spleen cells, hemolysin plaques/spleen, spleen index, and serum

hemolysin titers. This decrease in the primary antibody response occurs regardless of variations in the time between antigenic stimulation and irradiation. Timing is an important factor at lower radiation doses. Exposures to 250 R, at one day before or at three days after antigenic stimulation caused a significant decrease in the primary antibody response, but exposure one day after stimulation produced only a decrease in serum hemolysin titers.

Gastrointestinal Radiation Injury

Rats were prepared with biliary fistulae four days prior to injection with tritiated thymidine. They were killed at 1, 8, 24, 40, 46, 52, and 58 hours after injection to measure the influence of bile on crypt cell transit time. Grain counts and DNA isolation procedures on segments of the intestine are to be done by R. J. M. Fry and W. R. Kisielewski of Argonne National Laboratory.

Plant Nutrition

Application of pyranine fluorescent dye to geranium leaves and removal of excess dye showed specific sites of retention. Further work is in progress to determine the significance of this with respect to uptake of materials by leaf surfaces.

Preliminary tests of the effect of surface on retention of particulates showed that smooth surfaces such as parafilm or paper retain particles less than 10 microns, while leaf surfaces retain particles up to 20-25 microns in diameter. The pubescent leaves of geranium retained the largest particles tested - 25 microns, and none were observed to be dislodged by a vigorous airstream. These tests were conducted using ZnS particles and comparing photographs of these particles taken before and after treatment with the air stream. Moistening of surfaces increased retention of particles and doubled the retention of ^{131}I gas.

Leaching studies of soils taken from greasewood areas where there is high salinity and from normally calcareous soil areas showed little difference in the movement of cesium-137 through the soil column when 300 inches (50 years equivalent of rainfall) was passed through the seven-inch soil columns. In all cases the first half-inch of soil retained 95% or more of the initial activity adsorbed at the surface of the column.

Microbiology

Addition of methionine to minimal media will cause E. coli to increase its relative rate of synthesis of RNA by a factor of about 30%. There is no corresponding increase in rate of growth. Using experiments in which cultures are moved on to or off of supplemented media indicates the necessity of examining the qualitative nature of the RNA stimulated by presence of methionine.

Related experiments are being conducted using Neurospora crassa and varying the conditions under which the mold is grown.

Radiation Effects on Insects

Special source irradiators have been developed by which flour beetles will be chronically irradiated with Co^{60} at dose rates of 200, 50, and 22 rads per day and with the radiation being done at two different temperatures. These cultures were established and monthly census of populations will be carried out over a prolonged period of time.

Terrestrial Ecology

Statewide fallout collections were completed for the year's series in the Cascades-Hanford-Blue Mountain areas. Data lost in the fire are being reconstituted through the IEM, and analyses are continuing. Compilation of the data may not be complete for several months.

Wintering waterfowl populations declined steadily during the month from 44,000 birds at the beginning to about 8,000 at the end. The greatest loss was associated with the exodus of about 30,000 mallards during the first week of February. The Canada goose population reached a seasonal high of 15,000 birds during this same period. Pairing of geese occurred at the usual time near the first of the month, increasing from 7 pairs on the 2nd of February to 57 pairs on the 24th of February.

Arrangements were made by which Dr. Wyatt Cone, of the Prosser Irrigation Station, could study insect populations occurring in the natural area surrounding Rattlesnake Springs.

Alaskan Studies

Corrected values for whole-body counts obtained at Anaktuvuk Pass on January 21-22 were as follows:

<u>Age Group</u>	<u>No. of Persons</u>	<u>Cs¹³⁷ body burden (nCi)</u>		
		<u>Mean</u>	<u>S.D.</u>	<u>Maximum</u>
Children (< 14 years)	30	88	+	41
Minors (14-20 years)	12	390	+	170
Adults (> 21 years)	41	660	+	240
Control adults	25	640	+	190
				970

These values are about 15% lower than those obtained during November, about one-half those of last July, and 50% greater than one year ago. The decrease from November is surprising in view of the availability and utilization of fresh caribou meat throughout this winter, which was expected to stabilize or increase Cs^{137} body burdens at this time.

The NaI (Th) crystal component of the shadow-shield counter suffered further cracking, evidently due to physical shock, and is now usable only at Anaktuvuk

Pass. This may affect the subsequent program of whole-body counting of people at other villages.

Sampling of caribou and other animals at Anaktuvuk Pass continued. Cesium-137 concentrations in caribou flesh during November were slightly greater than observed one year ago.

Columbia River Limnology

Approximately 100 white fish were collected at Priest Rapids Dam on January 30. The mean concentration of Zn^{65} in the eyes and gastrointestinal tract was 530 and 820 pCi/g wet weight, respectively. Sixty-six percent of the eye samples and 72% of the gastrointestinal tract samples were above background limits for Zn^{65} . Eighty-one percent of the samples collected a year ago had above background levels of Zn^{65} . Results of total beta, Zn^{65} and P^{32} analysis of muscle tissue are not completed.

Wm. J. Bair
BIOLOGY DEPARTMENT

WJB:ba

APPLIED MATHEMATICS DEPARTMENT
MONTHLY REPORT - FEBRUARY, 1965ACTIVITIES FOR OTHER HAPO COMPONENTSN-Reactor Department

A set of single-sample plans was developed for a specific application by Fuels Engineering.

The Quality Control Engineering Unit has outlined and proposed a 12-lecture course on the concepts and procedures of statistical quality control to be given to N-Fuels engineers and technicians. The proposed outline was reviewed in light of the request that Statistics personnel participate in the lectures.

A previously used plan for sampling steam generator tube bundles during production outages was reviewed. This plan is being updated and expanded to provide an overall sampling procedure commensurate with the current goals.

Irradiation Processing Department

A discussion of statistical methods appropriate to 3-way and 4-way crossed experimental layouts was written for the Product Engineering Unit of the Production Fuels Section.

Data reduction and analysis procedure was begun on data obtained according to a 2 x 3 crossed layout relative to quench and anneal effects in terms of irradiation-induced distortion of fuel elements.

Technical assistance in the installation, adjustment, and conduction of acceptance tests for ground detection devices on the 105 Building reactor coolant pressure monitoring systems has been completed at B and C areas. This brings to a successful conclusion the installation of this device on all reactors for which it was scheduled.

Chemical Processing Department

Programming of the mathematical model for the proposed chemical counter current crystallization process is progressing as scheduled.

ACTIVITIES WITHIN THE LABORATORIES2000 Program

A solution was obtained and an EDPM program written to evaluate the transient heat transfer and the maximal temperatures experienced in a source-laden slab which radiates heat into two media at different temperatures. Initial computer

runs have been good, but improvements are being made to the program to increase its efficiency and versatility.

4000 Program

Particle-Packing Studies

A table of particle sizes and proportionate mixes for a .505-inch I.D. cylindrical fuel element was computed which lists three almost equally desirable alternatives from a density standpoint. Studies were also made on modes of packing which would have the property of uniformly distributed voids.

Fundamental Ultrasonic Studies

The attenuation properties and the direction of motion for a spectral element of a plane wave pulse propagating in a semi-infinite medium have been evaluated in terms of material parameters and the geometric orientation of the plane wave.

The ratios of incident to rebounding potential amplitudes of spectral elements have been evaluated by solving the appropriate stress-displacement equations for liquid-liquid, liquid-solid, solid-liquid, and solid-solid interfaces.

A lecture on the mathematical model used for evaluating these motions was given and well received by those in attendance.

General

An analysis of Pu burn-up (PRTR) data was continued. Error evaluations from prior analyses were employed in the function used to compute the cross sectional ratio, in an attempt to estimate the error propagated to the ratio.

An experimental design was proposed for obtaining estimates of the variability and chemical reactivity of each of ten types of graphite.

A flow chart of the PRTR fuel fabrication system was completed. A design for an experiment relative to the nupac and vipac subsystems was proposed.

5000 Program

Actinide Element Research

Work continued on the statistical calculations subroutine for nonlinear search program to be used to obtain least-squares estimates and joint confidence regions for the diffusion constants of an equilibrium model describing the extraction of sulfuric acid by Toa-Benzene.

Statistical Methods in Radiochemistry

Work continued on the power function calculations for the uniformly most power similar test to detect radioactive decay in the presence of random background.

Conditional power curves have been graphed as a function of the number of counts observed and the number of half-lives of observation time. Unconditional power curves for both fixed and random time counting procedures are now being calculated as appropriate combinations of the conditional values.

6000 Program

Computer programs were written to handle data from perfusion experiments and from chelating agent tests on the rate of removal of Cs^{134} .

Model parameters were estimated from capture-recapture data from various beetle populations.

An analysis of spectrophotometric data was initiated. The problem consists of obtaining estimates of equilibrium and absorptive parameters (with confidence intervals) in a nonlinear regression model with error in both variables.

A statistical analysis was begun to investigate the effect of diet and time on the Cs^{137} body burden of villagers in Northern Alaska.

Other

An analysis was begun on data generated by the Nuclear Materials and Propulsion Operation, General Electric Company, Cincinnati. The study concerns the effect of neutron radiation on certain properties of alloys. A prediction model is being sought to express time-to-rupture as a function of temperature, stress and neutron dose.

An experimental design was constructed for the calibration of a constant temperature fluid heater which maintains temperature by the simultaneous use of six induction furnaces. The main purpose of the design is to determine whether or not the effect of the six furnaces is additive over the heat range of interest.

Manager
APPLIED MATHEMATICS DEPARTMENT

CA Bennett:lg

RADIATION PROTECTION DEPARTMENT
REPORT FOR FEBRUARY 1965

Organization

Betty B. Haller transferred to Metallurgy Research effective February 26. Sandra S. Young accepted a position as secretary with Dosimetry Studies and Evaluation Section effective February 23, 1965.

Laboratory Experience

There were two incidents involving four employees which required special bioassay sampling for plutonium analysis. In addition to the incidents involving plutonium, there were three incidents involving four persons that required evaluation for possible intake of other radioisotopes.

The badge film of a Coolant Systems Development employee working at the 1706-KE facility showed 1.8 R for January. A formal investigation of the handling of the employee's film badge dosimeter produced evidence that it was subjected to heat while at home. An experiment to verify the heat effect on the badge film while in the employee's home is in progress.

Environmental Evaluation

The average concentration of fallout materials in the air in the vicinity of the Hanford project during the four week period ending February 19, 1965 was 0.2 pCi β /m³, the same as the average for the past two months.

The estimated dose to the GI tract due to drinking water from the Richland municipal water system was about 4.4 mrem during the months of January and February. This was slightly less than the dose estimated for similar months of 1964. The average concentration of P³² in the flesh of whitefish caught at Hanford was 46 pCi/gm. This was less than half that found during January 1964, but quite similar to 1963 experience. The rate of transport of P³² in the river as measured at Richland dropped from an average of 40 Ci/d during December to about 25 Ci/d during January. The decrease was not indicated by IPD-GE measurements of rate of release which did not change between December and January. If the value of 25 Ci/d is a true value, it represents a new low for this time of year.

The U. S. Testing Company began analyzing environmental samples on a limited basis on February 23. Samples which are being delivered to them are: air sample scrubbers and filters, water samples requiring total beta only, one large volume water sample per week for detailed isotopic analysis, beef thyroids,

milk, and oysters. This limited delivery will be continued for several more weeks until U. S. Testing has completed calibration. The Radiological Analysis group will continue to analyze other off-site samples so that our schedule may continue at the reduced level established in December.

Locations for new control plots outside of the 200 Area exclusion fences were investigated and laid out. Control plots on the Wahluke Slope were redefined with stakes and signs. Surveys at control plots showed no detectable contamination.

Aerial monitoring flights to Spokane and down the Columbia River to Astoria were made.

Nuclear Safeguards

With the assistance of Physics Laboratory and Programming personnel, a research proposal was prepared and submitted to the U. S. Army Electronics Command, Fort Monmouth, New Jersey. The objective of the proposed research is to develop an improved land fallout prediction system based on the dynamics of the fallout environment from time of burst to fallout deposition. The proposed project would require about 90 man-months.

New studies are underway of the effect of various postulated production reactor accidents on the Wahluke Slope, assuming that the entire area were developed for farming. The economic parameters have been determined, analyses based upon them have been made, and conclusions are being formulated.

Work is continuing on the high-power density core analyses for PRTR. Preliminary data and subsequent analysis indicate that a portion of the center of several fuel rods will be molten during operation. Studies are under way to determine the tolerable limit of fuel rod partial melting. It has been proposed that a boron chemical shim (boric acid dissolved in the moderator) replace or augment the present mechanical shims during high-power density core operation. Preliminary tests of relatively dilute boron shims are planned. In support of these tests an analysis was made of the nuclear effects of a hot primary coolant blowdown through a pressure tube rupture into the borated moderator (20 ppm boron shim), at initial moderator levels of 8 feet, 6.5 feet and 4 feet. In all cases the combined result of clean hot primary coolant entering the moderator, void formation, and rise in moderator level was a negative reactivity insertion.

Further improvements were made in the computer program for whole body dose from a radioactive cloud, permitting the extension of the exposure time interval from 8 hours to 24 hours. Whole body doses from a postulated ground level release of activity from PRTR were calculated. The computed results are being analyzed.

A brief survey was made of the nuclear safety of the Dynapak operation in the 308 Building. This is the first processing operation which has come to our attention which could possibly cause a nuclear detonation. The equipment as presently set up is nuclearly safe, since it cannot contain a critical mass at maximum density and with optimum conditions of fast neutron reflection by the metal parts of the machine.

Dosimetry Studies

A program was initiated to calibrate the whole body counters for lung burdens of various radionuclides using a REMCAL phantom. It is planned that the calibration will include Pu^{239} , Co^{60} , Fe^{59} , Sc^{46} , Ru^{103} and $\text{Zr}^{95}\text{-Nb}^{95}$. These radionuclides were chosen since they are the ones most commonly found in radiation incident related measurements.

Calibration of the Co^{60} radiation facility for the Biology Department was begun. Standard R-meter chambers related directly to the U. S. Bureau of Standards calibrated R-meters are being used for the study which was about 80 percent completed in February.

The Van de Graaff electron accelerator was used to provide a high beta dose to study pasteurization of specially packaged peeled potatoes for General Electric's Irradiation Processing Department.

A special dosimetry problem requiring the measurement of Co^{60} gamma radiations at a wide variety of positions within a flour beetle population was solved for the Biology Department. Seventy lithium phosphate glass rod dosimeters were provided, calibrated and evaluated to give the needed dosimetry information.

Environmental Studies

A study was made to examine dietary correlations that may be made using results of the Hanford Whole Body Counter and diet questionnaires. The example chosen was the Cs^{137} body burden resulting from milk consumption. The data taken during nine random months were chosen as a sample. A small but highly significant correlation was found between milk consumption and Cs^{137} body burden for the 979 sets of results examined. It was found that an average of 26.3% of the Cs^{137} body burden was obtained from milk. The effective half-life of Cs^{137} in the total body calculated from these data is 67.3 ± 22.3 days, which agrees well with the ICRP value of 70 days.

The RLOO-AEC requested comments on a proposal by GE-NRD to dispose of chemical decontamination wastes directly to the river following a forthcoming primary heat exchanger decontamination. In reply it was stated that, although no

applicable radiological or chemical limits for the river would be expected to be exceeded with the proposed disposal, river disposal of wastes from this routine, if infrequent, job represented an undesirable waste disposal practice. Disposal to a temporary crib was recommended.

Efforts in the development of the "Nuclear Facilities Monitoring Guide" were directed toward completing a section which quantitatively describes the significance of gaseous iodine release and the basis for calculating permissible stack releases. More time has been spent on this one aspect of monitoring than any other because iodine releases cause the major problems in routine power reactor operation. This is one of the areas in which the Guide can make an obvious contribution.

A study is in progress of the effects of reactor effluents on Columbia River water quality. Compilation and reduction of routine operating and meteorological data continued. One dye test for travel time measurement was made from N Area to 300 Area during the month. Additional temperature measurements were made in reactor effluent plumes and near the shoreline below K and G Areas where warm water seepage is occurring.

Applied Reactor Engineering, GE-IPD, has undertaken a study of inlet water temperatures at 181-D. During recent low river flow periods with N Reactor in operation, temperature increases in the raw water level exceeded any previously experienced on plant. A letter has been received requesting the assistance of J. P. Corley in this evaluation. A consultation with the IPD representative was held; the extent of future assistance is uncertain.

A special study of inlet water temperature at the 181-N river pump house has been in progress. An additional dye test for B Reactor effluent intake at 181-N was made at an intermediate river flow rate. The results confirmed an estimate based on earlier data.

Radiological Development

Design of a new thermoluminescent reader was completed. The design provides for improved stability within the heating and light integrated circuits. Also, larger quantities of thermoluminescent materials may be read with the new equipment. This results in a simple means to increase the sensitivity and lower the detection limit of the thermoluminescent dosimeters. Ample space and design considerations will permit installation of an optical system so that specific light wave lengths may be examined independent of the total luminescent glow. Specific wave length study may provide a lower background for the system by eliminating non-radiation caused transitions.

Radiological Engineering consultation was provided for the design of a 10,000

square foot addition to the Fuels Recycle Pilot Plant (FRPP). This addition would include facilities for a plutonium laboratory, reactor fuels rupture and decontamination laboratory and a high-level analytical chemistry cell. The design is proceeding on the basis that absolute separation of the currently designed functional areas of the facility and the proposed facilities is to be maintained. In addition, Radiological Engineering assistance in the design of gas decontamination equipment for the FRPP metallurgical cells was begun. The major problem is to devise a system for the collection of the volatile fission products that may escape from a ruptured fuel element. A vacuum chamber of a relatively few liters in volume may prove adequate for the collection and storage of these materials at reasonable cost.

An improved and wider time range system for pulsing the electron Van de Graaff for special dosimetry physics and chemistry studies is being designed and installed. The addition of this system to the Van de Graaff accelerator when coupled with the capabilities of the flash x-ray equipment will provide a precisely controlled irradiation capability from about 0.1 μ sec to many hours.

A new 1000 gram plutonium fluoride neutron source was obtained. Calibration equipment modifications to accommodate the use of this source were provided. The new source is doubly contained with an inner can of monel material and an outer can of aluminum. The monel inner container was chosen to provide maximum inertness to fluorine. An air space of approximately 1/8 inch is provided between the two containers so that x-ray examination can be used to detect any possible source swelling of the inner container. The higher density monel metal has significantly reduced the gamma radiation dose rate from this source. Also, the recent purification of the plutonium used has resulted in a source with a minimum of Am^{241} . Gamma radiation dose rates from this source will increase as the Am^{241} content increases.

The use of silver phosphate glass rod gamma dosimeters for in-reactor gamma dose measurements at levels to about 10^6 R/hr was discussed with Nucleonics Instrumentation. An interference of 10^{12} n/cm²-sec would necessitate shielding the neutron flux or use of a glass dosimeter with relatively low neutron sensitivity. Some 23 w/o boron shielding was provided for testing. The use of quartz glass dosimeters was recommended as an alternate.

Outside Consultations

Several meetings were held with Dr. Mancuso, Research Professor, University of Pittsburgh, concerning the availability of exposure records for use in the AEC epidemiological feasibility study being conducted by him. At the request of Dr. Mancuso, occupational exposure histories were prepared for 75 individuals for use with certain medical statistics derived from occupational health records.

The RLOO-AEC requested that they be supplied with (1) estimates of increased environmental survey costs and (2) the cost of evaluating radiation exposure to people if a contamination event occurred in the Wahluke Slope area, if the assumption were made that the Wahluke Slope area were extensively farmed. It appears that adequate surveillance could be accomplished for within \$10,000 per year. The cost of evaluating radiation exposure to people from a contaminating event is estimated to be within \$5000. This estimate includes those incidents where it is necessary to define only the magnitude of the exposure and extent of area involved.

Hal Bernard and William Regan from the Environmental and Sanitary Engineering Branch, DRD, AEC Headquarters, met with Environmental Studies & Evaluation Section personnel several times during the period February 1 to 3, to discuss work in progress on the "Nuclear Facilities Monitoring Guide."

Consultation service was supplied to GE-CPD personnel concerning open pond aqueous waste disposal practices and radiation problems associated with planned excavations in the immediate area surrounding the old production buildings, 221-B and 224-B.

Approximately two days were spent at the U. S. Testing Company facilities tuning up the badge processing equipment, the film x-ray identification coder and the mechanized film densitometer. Essentially all difficulties encountered originated from dust or dirt in the equipment and the fluctuating voltage in the electrical power supply to the U. S. Testing building. Neither of these conditions was encountered in the 300 Area facilities.

Staff Activities

R. F. Foster and R. B. Hall visited the Division of Water Quality and Pollution Control, U. S. Public Health Service, Department of Health, Education and Welfare, Cincinnati, Ohio, February 16 and 17. The purpose was to discuss the methods used and results obtained in describing the effect of the Hanford Project on the content of radioactive materials in the Columbia River.

H. E. Hanthorn and L. J. Nitteberg presented an oral report on the status and problems of the high-power density core safeguards analysis at an information meeting for R. Grube of Washington-AEC.

J. K. Soldat presented a two-hour lecture entitled "Evaluation of Radiological Conditions Through Environmental Analysis" to about one dozen personnel of the Special Analysis group of the analytical laboratory of Battelle-Northwest.

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J. P. Corley was assigned to the Industrial Wastes Committee of the Pacific Northwest Pollution Control Association.

R. O. Budd, J. P. Corley and R. B. Hall received notice from the American Board of Health Physics of certification as Health Physicists.

Manager
RADIATION PROTECTION DEPARTMENT

AR Keene:np

REACTOR DEVELOPMENT - 04 PROGRAMPLUTONIUM RECYCLE PROGRAMPlutonium Recycle Test ReactorOperation

Reactor output for February was 1719 MWD for an experimental time efficiency of 91% and a critical efficiency of 90%. There were five operating periods during the month, two of which were terminated manually, two were terminated by scrams, and one continued through month-end. A summary of the fuel irradiation program as of February 28, 1965, follows:

	<u>Al-Pu</u>		<u>UO₂</u>		<u>UO₂-PuO₂</u>		<u>Other</u>		<u>Program Totals</u>	
	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>	<u>No.</u>	<u>MWD</u>
In-Core	0		6	1864.8	74	15894.2			80	17759.0
Maximum				403.0		451.4				
Average				310.8		214.8				
In Basin	7	572.5	27	3076.1	61	9248.4			95	12897.0
Buried							1	7.3	1	7.3
Chem. Proces.	68	5465.8	35	1965.8	—	—	—	—	103	7431.6
Prog. Totals	75	6038.3	68	6906.7	135	25142.6	1	7.3	279	38094.9

(Note: MWD/Element x 20 = MWD/TU for UO₂ and UO₂-PuO₂)

Heavy water loss and indicated helium loss for the month were 455 pounds and 109,509 scf., respectively.

Equipment Experience

The main maintenance jobs during the month's outages were primary system leak repairs and diesel generator wiring repairs.

Preventive maintenance utilized 150 manhours or 3.6% of available manhours of assigned craftsmen.

Improvement Work Status (Significant Items)Work Completed

Hot Crud Sampler Installation
Gamma Scanner Shutter

Work Partially Completed

Corrosion Loop Installation
Air Lock Door Operators
Modification to PRTR Warehouse 3718-C
Supplemental Emergency Water Addition
PRTR Steam Utilization
Additional Fuel Storage and Examination Facility
Creep Test Facility
Flux Wire Scanning System
D₂O Cleanup Facility
Pneumatic Irradiation Facility

Design Work Completed

"B" Cell Instrument Tubing Penetration

Design Work Partially Completed

Instrument Power Supply
PRTR Experimental and Building Facility Addition
PRTR Increased Power Level

Process Engineering and Reactor Physics

Four of the 58-inch UO₂-2 w/o PuO₂ elements are under irradiation. The maximum exposure as of February 28, 1965, was 77.4 MWD (2335 MWD/TU). The power generation rate for these four elements continues to be in the 1000 - 1350 KW range.

Appendix VI, HW-84544, Calculated Reactivity Effect of Displacement of Borated D₂O with Clean D₂O in the PRTR Calandria, was issued during the month.

The reactivity effect of injecting uniformly dispersed voids into the moderator of PRTR with the present core configuration has been calculated, with 20 ppm natural boron and with no boron in the moderator. Addition of the boron reduced the magnitude of the effect by about 30% but it remained strongly negative.

Initial machine runs to determine thermal shield temperature distribution indicate isothermal conditions for all practical purposes. On this basis, shield curvatures previously computed appear to be quite conservative in regard to expected bending. While considerably more work can be done, higher power levels do not appear to offer any mechanical difficulties from thermal expansion of the thermal shield.

A bulletin was posted in prominent locations describing an example of minor stress corrosion cracking which occurred on a top ring header spare jumper connection. This is the first in a planned series of bulletins to familiarize personnel with typical examples of stress corrosion and how it can occur at PRTR.

Experimental Reactor Services

The status of the various test elements at the end of February 1965 is shown below. Those elements discharged prior to February 1, 1965, have been deleted from this table.

Test No.	Channel Location	Element Number	Description	Date Initial Charge	Date Discharged	Approximate Accumulated MWD
14	1956	5097	Swaged-0.48%	4/2/62	--	276.1
14	1758	5099	Vipac-0.48%	5/8/62		285.1
37	1550	1097	UO ₂ -Physics	5/12/62		314.0
48	1156	5150	0.48% (1/2"x1/2" Pads)	8/1/62		288.0
54	1542	5116	0.48% (clip-on pads)	5/8/62		293.8
54	1554	5118	0.48% (clip-on pads)	5/8/62		451.4
61	1249	5186	0.48%-Physics	5/28/63		315.8
61	1445	5192	0.48%-Physics	6/13/63		297.0
67	1047	5117	0.48% (Repaired wire)	10/20/63		256.3
72	1253	5253	1% (Zr coupons)	9/1/64		154.1
85	1855	5230	Vipac - 1%	1/30/64		218.5
108	1954	6501	EBWR Cluster	1/30/65		12.7

Fuel Element Rupture Testing Facility

The irradiation of the EBWR cluster in the FERTF was completed with the reactor shutdown of February 24. The test element accumulated about 552 hours operating time and about 10.6 MWD in the loop during FERTF Test #3. The power level of the fuel element was found to be about 480 KW.

The three inch length slit was extended to about 6-1/2" on UO₂ fuel element 1067, and the element was charged for the reactor startup of February 25.

Chemical Processing of Spent PRTR Fuels

CPD has postponed the next chemical processing campaign until July 1965 because of other AEC commitments.

TECHNICAL SHOPS

Total productive time for the period was 27,719 hours. This includes 17,845 hours performed in the Technical Shops, 6,083 hours assigned to J. A. Jones Company, 3,568 hours assigned to off-site vendors, and 223 hours assigned to other shops. Total shop backlog is 21,653 hours, of which 90 per cent is required in the current month with the remaining hours distributed over a three-month period. Overtime worked during the month totaled 2,164 hours or 10.7 per cent of the total available hours. Distribution of time was as follows:

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	<u>Man Hours</u>	<u>% of Total</u>
N Reactor Department	2,844	10.3
Irradiation Processing Department	5,227	18.2
Chemical Processing Department	343	1.2
Battelle-Northwest Laboratory	19,305	69.6

LABORATORY MAINTENANCE

Total productive time for February was 22,032 hours of 23,228 potentially available. Of the total productive time, 97% was expended in support of Battelle-Northwest departments, with the remaining 3% directed toward providing service for other Hanford organizations. Craft overtime worked during February was 1.8% of total available hours. Distribution of time (in hours) was as follows:

A. Shop Work		1596
B. Maintenance		7871
1. Preventive Maintenance	1754	
2. Emergency or Unscheduled Maintenance	1518	
3. Normal Scheduled Maintenance	4199	
C. R&D Assistance		12,665

WD Richmond

Manager

Test Reactor and Engineering Services

WD Richmond:JGZ:pr

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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.

<u>INVENTOR</u>	<u>TITLE OF INVENTION OR DISCOVERY</u>
A. E. Albrethsen	Uranium Dioxide-Lead Radiation Shielding
W. A. Haller, R. W. Perkins, and J. M. Nielsen	Gamma Spectrometry Utilizing Beta Anticoincidence Counting for Cesium-137 Measurement in Fission Burnup Analysis and Manganese-54 Measurement as a Fast Neutron Flux Monitor
G. H. Strong	Air Cooled Pressurized Motor Pump
G. H. Strong	Underwater Breathing Augmentation Device, or "Rebreather:
G. H. Strong	Totally Opaque Blind
L. T. Pedersen	Controlled Cycle Engine

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