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PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT
APRIL 1965

Compiled by
Department Managers

May 14, 1965

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PACIFIC NORTHWEST LABORATORY
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RESEARCH AND DEVELOPMENT

1. Reactor and Materials Technology

Production Reactor Support. The thermal stability of additive uranium compositions appears to be a function of the volume of small second phase particles produced by the additives silicon, iron, and aluminum.

Examination of a fluted fuel element (N-Reactor single tube size) after 2500 MWd/ton exposure indicates that the design performed as intended in accommodating volume expansions of up to 11% locally. The unexpectedly large localized swelling is anomalous and associated with end effects.

Fatigue endurance limits have been determined for several sizes of N-Reactor fuel supports.

Irradiated Al-Li target rods with axial holes filled with boron and boron carbide have been fabricated for reactivity tests of the effectiveness of buried poison when the specimens are heated to rupture. A test element with 50 wt% Cu-50 wt% Cd buried poison is being prepared.

Elevated temperature reactions between cadmium alloys and zirconium are being investigated.

The kinetics for oxidation of Zircaloy-2 at 900, 1000, and 1100 C in 25 torr water vapor has been expressed by means of a simple rate equation.

Anodization techniques have been developed for the measurement of oxide thickness on K-Reactor Zircaloy process tubes.

Based on five N-Reactor (predefected) fuel element tests in an out-of-reactor loop, fuel elements with high concentrations of iron and aluminum in the core displayed better failure patterns than the standard elements.

High chloride concentrations, formerly noted in crevices between tubes and tube sheet of the N-Reactor steam generator, have decreased with time. Apparently the chlorides originally present are being leached out by the secondary coolant.

Fourteen boiling burnout points were obtained in the study of heat transfer conditions with a coproducer configuration in N-Reactor. In general, the results of the experiments agree quite well with those run previously using a similar test section which had both surfaces of the annulus heated.

Long-term irradiations of N-Reactor graphite in the GETR have now been carried out to a maximum dose of 4.8×10^{21} neutrons/cm² ($E > 0.18$ MeV) at temperatures of 425 to 825 C. All samples are continuing to contract, with the greatest contraction occurring at the lowest temperature. Because of the unavailability of Irradiation Units, these tests will be discontinued at the end of this fiscal year. The possibility of continuing the irradiations in the C-Reactor at Savannah River during the high-flux operation is being investigated.

Plutonium Recycle Program. Agglomeration techniques for mixed oxide fuels are being evaluated to produce a new, cheaper feed material for fuel element fabrication. Five different starting materials and processes have been studied to date.

The maximum burnup on a UO_2 -2 wt% PuO_2 element in the PRTR is currently 3030 MWd/ton_U with an average of 1775 MWd/ton_U for the seven elements in the reactor. The first 2 wt% PuO_2 element, containing three rods of salt cycle fuel material, was discharged from the PRTR rupture loop and examined. The element is ready for recharging in the reactor.

All prototype components for Phase I PRTR boron shim test were installed and out-of-reactor tests were performed. Phase I tests are scheduled for the May outage.

Based upon two experiments, the crack propagation strength of PRTR Zircaloy-2 pressure tubes is about 8% greater for fast neutron exposures of 1.2×10^{21} nvt than for fast neutron exposures of 1×10^{20} to 6×10^{20} nvt. Specimens having the lower exposures had crack propagation strengths which were no different than for unirradiated specimens.

Initial measurement of crack propagation of properties of Zr-2.5 wt% Nb pressure tubes reveals it to be more resistant to crack propagation than cold worked Zircaloy-2.

Plutonium Ceramics Research. Postirradiation examinations are being made of high and low density stainless steel-20 vol% PuO_2 cermet irradiated to 1.6% burnup of the fissile atoms. The high density cermet operated with a centerline temperature of 1300 C. There was some evidence of sintering in the central region but no evidence of a reaction between the stainless steel and the PuO_2 . The low density cermet operated with a partially molten core and produced structural changes analogous to those occurring in irradiated UO_2 .

Six hundred grams of $\delta\text{-Pu}_2\text{O}_3$ were prepared by carbon reduction of PuO_2 . No PuO was found on the surface arc melted buttons.

Preparation and characterization of a working stock of plutonium nitride is in progress. Preliminary work on PuN -20 wt% SS cermet has shown the necessity for storing the nitride in a very pure atmosphere. Techniques for metallographic examination of nitride-containing materials are being developed.

Isothermal oxygen disassociation pressures over americium suboxides are being measured. Plutonium suboxides are being prepared for similar measurements.

Ceramics (Uranium) Fuel Research. The pneumatic impaction process is currently being investigated as a means of densifying BeO , BeO-UO_2 , and BeO-PuO_2 .

Mechanically mixed and impacted UO_2 -20 wt% PuO_2 is being prepared for radiation evaluation in EBR-II in cooperation with Argonne National Laboratory. A process has been developed that yields a high density, low gas content, near stoichiometric, fuel material.

The calculated burnup on the last of three capsules containing UN-20 wt% PuN is 12.8×10^{20} fissions/ cm^3 . X-ray diffraction analysis of the reaction zone in the stainless steel cladding from the first capsule has revealed no foreign phases.

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Single crystals of UO_2 were prepared, characterized, and shipped to two sites in Japan for basic studies.

A controlled pressure-temperature capsule continues to operate satisfactorily in-reactor at 1000 psi and 625 C.

Swelling Studies. Density measurements on uranium specimens recovered from the swelling capsule controlled at 575 C and 1000 psi indicate a reduction in the amount of swelling typically observed in low pressure capsules by factors of 5 to 15.

Metallographic examination of uranium specimens irradiated at 575 C and 1000 psi to 0.15 at. % BU reveals that the crystallographically aligned microtearing typical of uranium irradiated at this temperature at low pressure has been suppressed by the 1000 psi pressure.

Irradiation Damage to Reactor Metals. Several nickel-base alloys irradiated to exposures of 1×10^{21} nvt at temperatures to 740 C have been tested at room temperature, 300 C, and 650 C. Results from the room temperature tests indicate that while the strength values increase with irradiation, the ductility of the alloys tested remains good. Data from the 650 C tests show that in general a marked decrease in ductility is found in most alloys, especially at the higher exposures.

To determine the effects of irradiation in a liquid metal environment, a temperature controlled capsule has been designed for insertion in the ETR. A bench model of this capsule has been fabricated and a safety circuit and temperature control system installed.

Three in-reactor creep tests of stainless steel and Zr-2.5 wt% Nb alloy are currently in progress.

Continuing efforts have been made to measure the crack velocity in a specimen undergoing catastrophic failure. Test equipment is now being set up to measure the crack velocity by the use of eddy currents.

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Helium stored in the recovery balloon of the helium purification system for the gas loop was purified in two passes from 3700 ppm total contaminants to no detectable contaminants except 6 ppm moisture.

Thermal conductivity measurements on Hastelloy X Feltmetal insulation for the ATR gas loop hot piping are nearing completion.

Thermal conductivity tests on the foil insulation for the model gas loop were completed. A failed bearing in the frequency changer has been repaired and is back in operation.

Graphite Studies. The reaction rate of carbon dioxide with water vapor is greatly enhanced by the presence of argon in a gamma radiation field. Energy absorbed from gamma radiation is utilized with almost 100% efficiency over a wide range of partial pressures of carbon monoxide and water vapor. It had been shown in earlier tests that helium also sensitizes this reaction.

The oxidation rate of 7% boronated graphite (used as the shield in Fermi Reactor) in helium containing 1% oxygen decreases with time. This is believed to be the result of formation of B_2O_3 , which acts as a physical barrier to further oxidation. When the B_2O_3 is leached out with water, the oxidation rate is again increased.

Metallic Fuel Development. Irradiation of two sets of Th-U fuel elements continued uneventfully to integrated exposures of 880 and 11,100 MWd/ton, respectively.

Submicron uranium carbide particles dispersed in uranium are apparently effective in restraining fission gas swelling, but extensive grain boundary tearing was observed in the material.

Advanced Reactor Concept Studies. Experimental data on gas generation and transport in mononitride fuels have been reviewed. Fission gas release from nitrides appears comparable to that from carbides.

Work on the plutonium and U^{233} fueling of a Fast Compact Reactor has been completed. The final report and two reports on computer programs used in the study are being prepared.

Critical Flow of Steam-Water Mixtures. Additional data were obtained of upstream burnout in a 12 ft long electrically heated test section at 1500 psig with a flow rate of 7×10^6 lb/hr-ft². It was found that the data agreed well with previous data and that it made little difference if the heated portion of the test section included the upstream half, the downstream half, or the entire test section.

Fast Flux Test Facility Studies. The plans for the FFTF preliminary design study were reviewed with AEC Headquarters and accepted. Contracts were initiated on various aspects of the study with six off-site firms and consultants. There is need for early initiation of irradiation experiments to prove out the suitability of PuO₂ cermet as driver fuel for the FFTF.

Radiation Effects on Metals. Annealing rhodium irradiated to 7×10^{18} nvt at 800 and 1000 C for 2 hr results in coarsening of the spot defect size and a significant reduction in spot density, as revealed by transmission electron microscopy.

A capsule containing single-crystal molybdenum length-change and X-ray samples with several levels of carbon as impurity has been opened after an exposure of 1×10^{20} nvt ($E > 1$ MeV). Preliminary measurements show that length increases, after this exposure, are greater for those samples having lower carbon content.

A technique which permits a plutonium sample in the pole figure goniometer to be raised in temperature sufficiently high to effect the alpha to beta transformation has been developed and is being used for establishing crystallographic relationships between the alpha and beta phases of plutonium.

An explanation for the formation of disks of columnar alpha grains of plutonium during transformation has been advanced.

Customer Work. The second run of the HTLTR mockup was completed. The goal temperature of 700 C was maintained for 115 hr after which the unit was cooled and examined. One heater element was removed, and no

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malfunction nor corrosion were discernible. No further core corrosion was detectable compared with the condition at the end of the first run.

Moisture in HTLTR atmosphere was shown to corrode B_4C -graphite control material at 1100 C in two environmental tests. Additional tests are in progress.

EBWR capsules have acquired burnups of 4.7×10^{20} fissions/cm³. Thermal cycling studies of EBWR physics rods with aluminum cores have shown both growth and shrinkage of the rods.

The fabrication of 23 irradiation samples of Pu^{240} -Al and Pu^{241} -Al specimens for Phillips Petroleum Company was begun. Fabrication of plutonium and thorium bearing physics test rods for IPD has been started.

Sixty-eight Saxton rods were fabricated to 86-88% theoretical density. Fabrication will be completed and shipment made by May 5, 1965. Thermal cycling studies on fuel elements made from Saxton tubing rejected by ultrasonic testing has shown some propagation of these defects.

2. Physics and Instruments

Physics calculational models used in analyzing the N-Reactor are being developed and evaluated. A two-dimensional model is being compared with N-Reactor startup data on power distribution. Also, diffusion theory is being compared with transport theory.

Calculations are under way for deducing k_{∞} and thermal utilization from PCTR data taken on the coproduct lattice. The cases under study are wet, dry, and dry without target. Preliminary analyses and preparations are being made for other N-Reactor studies, including: Supercells of Pu-Al drivers and ThO_2 targets; control rod enhancement by replacing B_4C by B^{10} ; and a comparison of water to masonite as used in the exponential pile to simulate H_2O .

Critical experiments were performed with PuO_2 -polystyrene compacts that provide further data for evaluating the effects of neutron interaction

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on criticality, and criticality in complex geometry (core in the shape of a cross).

A series of experiments were begun utilizing PuO_2 -polystyrene assemblies to evaluate the neutron isolation properties of materials such as polyethylene, borated polyethylene, polypropylene and compressed wood; the data are of interest in the design of annular type vessels and for Class I type shipping containers.

A series of neutron multiplication experiments were completed with enriched uranium tubes (1.36 in. ID, 2.35 in. OD) of 1.95 wt% U^{235} in light water. Three lattices were measured. The experiments provide data needed for nuclear safety guidance in handling enriched N-Reactor fuels.

The use of the triple-axis neutron spectrometer for crystal structure analysis was studied and showed very adequate counting rates and resolution on test samples.

The measurements of slow-neutron scattering from Zr-H planned with the present sample have been completed. The results of scattering with excitation of the first three optical levels have been obtained from the data and compared with that expected from a simple harmonic oscillator.

Multiphonon calculations to aid in the interpretation of the Zr-H results have been delayed due to deficiencies and errors found in the Harwell computer programs LEAP and ADDEL T.

A successful measurement series of 3 to 15 MeV total cross sections was carried out for Li^7 , Li^7 , Ca^{42} , Ca^{44} , Tc^{99} , W^{182} , and W^{186} . The first use of an ORTEC ion source in the Van de Graaff and new electronic instrumentation in the detector system each gave greatly improved performance.

The measurement of reactivity worth of control and safety rods have been completed in the PRCF fueled with 19-rod Pu-Al clusters moderated with D_2O . Also, reactivity worths of a boron arrow were obtained from flux transient data, and worth of cadmium as a function of position in the

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core was measured. Flux measurements including lutetium activations from which spectral indices are determined were measured. The moderator level coefficient was measured to be about 1.7 times the two-zone D_2O core of Pu-Al and UO_2 elements.

Preparations for the 2 wt% PuO_2-UO_2 light water experiments planned for the Critical Approach Facility are complete and the experiments will start early in May. These experiments will use three plutonium isotopic compositions, and a range of moderator-to-fuel ratios.

Measurements in the PCSTR on the 6 1/2 in. graphite lattice, fueled with 19-rod clusters containing 0.9 wt% low exposure PuO_2 in depleted UO_2 , are complete. The multiplication of the lattice is very close to, but is less than, one.

Requested information was provided to AEC Headquarters on accelerating the schedule of the proposed Phoenix fuel experiment in the MTR calling for power tests in the last half of FY-1967. No decision has been reached by the AEC on this experiment. Use of the APPR as an alternate facility is being examined. The Phoenix experiments in the PCSTR are continuing with the 27 at. % Pu^{240} fuel in a cell having a H/Pu ratio of 200.

Conclusions from a study of methods for calculating effective resonance integrals for Pu^{240} indicate that the NRIA approximation and the integral method used in computer code ZUT give essentially identical results and agree with the experimental data of Nichols.

A rough draft of a document describing preanalysis of planned experiments with the plutonium demonstration experiment in the EBWR has been written. This report describes the critical loadings and reactivity changes which are expected during the startup experiments and at various stages of burnup of the core.

Several modifications have been made to code HRG recently. A change in the treatment of unresolved resonance should improve the accuracy of the calculation of their effects. Other changes make the code more flexible and useful.

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An improved method for updating the Basic Library code has been developed. The new method reduces the machine time involved in updating by a factor of 7 to 9.

The initial part of the second run of the HTLTR mockup has been successfully completed. This consisted of a heatup to 700 C, where the temperature was held for 115 hr. The prototype control rod, safety blade, and heater element showed no visible deterioration from the tests. A small gas cleanup system was effective in reducing the water in the nitrogen gas to 1200 ppm--a factor of 3 to 4 lower than the level without the cleanup system.

Fuel element development for the HTLTR using TD nickel cladding is under way. Graphite is also being considered as a driver fuel cladding material.

Physics studies continued on the Fast Flux Test Facility. Reflector control of the FFTF appears to be marginal and analysis of other control schemes are being carried out. Postshutdown heating data were provided for coolant system design analysis. A study of alternate fuels and structural materials in the reactor are under way. Four sets of cross section data used in fast reactor calculations are being compared.

The current MELEAGER burnup code was checked with experimental data for the buildup of Np^{237} , Pu^{238} , Am^{241} , Cm^{242} , and Cm^{244} as well as the plutonium fuel isotopes Pu^{239} , Pu^{240} , Pu^{241} , and Pu^{242} . Data were compared from Yankee rods having exposures from 13,000 to 31,000 MWd/ton and Vallecitos rods having exposures from 2000 to 16,000 MWd/ton. The quantity of the isotopes produced checked within 1% for all isotopes, except U^{235} for which the initial concentration is uncertain.

A total in-core exposure of 2×10^{21} neutrons cm^{-2} has now been received by the two regenerating ion chamber neutron detectors installed in KW-Reactor. Performance further confirms regenerating concept.

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Attempts to detect hydriding in irradiated IPD process tubes by thermoelectric techniques have proven unsuccessful in the field so far. Continuing efforts are investigating this and eddy current techniques.

A continuous wave laser beam has been successfully used to detect a 10 Mc ultrasonic signal in water. The beat frequency between the Doppler shifted light and original light beam was recorded.

The calibration of the plutonium counter for dogs has been completed. The calibration obtained using a dog phantom was in very good agreement with the calibration obtained by counting dogs which later died and were analyzed.

Another trip to Alaska to measure Cs^{137} in natives was made. Eskimos at Anaktuvuk were measured with the shadow counter which has been located there for the past 1 1/2 yr, and the natives at Arctic Village and Ambler were measured with the recently developed simplified whole body counting method.

The combination of an Einzel lens and a gap lens was found to provide a better focus of a helium ion beam for the positive ion accelerator. This reduces the interference of molecular hydrogen in the separation of the doubly charged helium ion beam.

Additional measurements of the turbulent fluctuations in wind velocity in the lower atmosphere were made during neutral and unstable atmospheric density conditions. Analog computer analysis for power spectral density estimates and cospectral estimates were completed for the stable atmosphere case.

3. Chemistry

Mechanical attrition failed to increase the rate of evolution of polonium from bismuth oxide.

Recovery of polonium was relatively poor in a minimixer settler test of dibutyl-carbitol solvent extraction separation of polonium from bismuth. Batch contact studies indicate the rate of extracting and stripping polonium is low.

[REDACTED]

Spray calcination runs with commercial thorium nitrate and material from Purex reprocessing were completed.

Previous data had shown that excessive erosion of the dissolver floor could be encountered when the model thoria dissolver contained an appreciable amount of abrasive solids. It has been established that the velocity of the impinging stream from the agitating devices is by far the most important variable and that the penetration rate into the metal is approximately a seventh power function of the jet velocity.

A process was developed for recovery of strontium and rare earths from a process concentrate containing relatively high concentrations of sodium and lead and lesser concentrations of aluminum and fluoride.

The loading of zeolite with Sr^{90} for a long term tests of welded capsules was completed. Elution of Sr^{90} from zeolite after 1 yr residence was readily accomplished although complete details on the completeness of removal are lacking.

A full-scale glass mockup of the redesigned slag and crucible dissolver was completed and placed in operation. Circulation studies confirmed the capability of the new configuration to maintain or restore circulation in the presence of a large amount (~15 liters) of residual, settled fines.

Simulated Nuclear Fuels Services' first- and second-cycle waste with added phosphoric acid was successfully converted to a "glass" in a spray calciner-melter. In a second engineering-scale test, similar waste, but without added phosphate, was pot calcined in prototype equipment designed for later radioactive service.

Leaching studies on radioactive glasses have shown a higher leaching rate for samples stored in moist air than in water or dry air.

No observable corrosion occurred in an AISI 304-L SS pot used in a laboratory-scale rising liquid level pot calcination and melt production run. Pot temperature was 900 C throughout most of the run. Feed used had the same composition as that used in a recent ORNL run in which severe pot corrosion was experienced.

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It has been postulated that the concentration of Pu^{236} and Pu^{238} in high exposure plutonium might force remote plutonium fuel fabrication methods. A comparison between the permissible level of Pu^{236} and the concentration found in a Yankee fuel element (24,600 MWd) has shown that this is not likely to cause a problem; the effect of the anticipated Pu^{238} content requires further evaluation.

In the studies of oxidation of plutonium the fraction of plutonium airborne and transported from small metal specimens oxidizing at 400 C was found to be very small, of the order of 10^{-5} to $10^{-6}\%$.

The adsorption and elution behavior of Rb^+ and Na^+ with Linde AW-500 zeolite was studied to evaluate possible effects on cesium loading and the purity of cesium if eluted from a storage canister or from a process purification column of AW-500 zeolite.

The particle size distribution of a sample of Linde AW-500 zeolite following use in rail transit of a cesium shipment was compared with that of a sample of the same zeolite taken before shipment. The physical stability appears to be satisfactory.

Preliminary indications are that Ce^{+3} and Pm^{+3} may differ sufficiently in kinetics that they may be separated without the use of oxidants in the D2EHPA solvent extraction system.

In the salt cycle system Np has been found to codeposit with UO_2 in the partition electrodeposition step. Americium tends to remain in the melt under either UO_2 electrodeposition or electrocodeposition conditions.

The use of "back emf" measurements for salt cycle process control has been shown to be limited. However, the method provides an indication of initial salt conditions and can be used as a measure of impending PuO_2 precipitation.

Pressure tests of the three-part containment vessel for the Containment Systems Experiment were completed satisfactorily. Design of the Containment Systems appurtenances was completed and construction was started as soon as the contractor for the containment systems vessel left the site.

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Use of monodisperse particles and improved techniques of measuring wall deposition are permitting better definition of the effect of air flow, particle size, and tube diameter on the transport of particles.

A quartz light bulb as a possible source for photometry has worked successfully for over 65 days immersed in molten halide salts at temperatures of 600 to 750 C.

Multidimensional gamma ray spectrometry showed the presence of Al^{26} in stony meteorites and, for the first time, in iron meteorites. With the same technique, Ru^{103} , Ru^{106} , Ce^{141} , and Ce^{144} in 5-month-old fission product mixtures were measured without chemical separation. Ba^{140} was also measured.

Trace metal concentrations in liver of Alaskan mammals have been compared to those found in local domestic animals. Higher concentrations prevail in the Alaskan samples.

Fe^{55} has been measured in the blood and food samples and body burdens estimated for man and animals.

Effective half-life and fractional absorption of technetium by humans was found to be different than previously understood.

Equipment for purification of promethium is essentially complete and installation in the hot cell has started.

Microspheres of samarium oxide have been successfully made using a plasma torch.

Thermal conductivity of SmF_3 was determined over the range 20 to 400 C. Compatibility testing of Sm_2O_3 and Nd_2O_3 with a variety of metals continues.

A promising highly sensitive method for measuring dose rate as a function of shielding has been developed.

A method for determination of tungsten-bonded oxygen in UO_2 tungsten cermets has been developed, as has a method for determination of total uranium in the same system.

[REDACTED]

The ability of the pneumatic impaction machine to fabricate complex isotope heat source geometries was illustrated by the preparation of a capsule, 2.5 in. diam by 2.8 in. length, and containing five 5/16 in. ID tubes. The tubes run the axial length of the capsule and could be used to heat a working fluid.

In the 100-F Area reactor effluent disposal to trench test radio-ruclide analysis data indicated that water moves at a rate of about 8 ft/hr through the soil in this area and that decontamination factors for Cr^{51} and Zn^{65} were 2.6 and 118, respectively, for travel through about 700 ft of soil. The total beta decontamination factor was 21.

Large volume ground-water samples, obtained from three wells located from 2 to 7 miles southeast of 200 East Area, were given special analytical processing to define the isotopes contributing to routine gross beta activity determinations.

Core barrel samples of soil taken immediately adjacent to 200-W burial gardens contained no detectable contamination.

Several significant changes were made to increase the utility of an existing GENORO hydrology computer program.

Associated with the fission product simulation for the Containment Systems Experiment the suitability of the cooled quartz tubing method for melting and retaining clad UO_2 specimens for long periods was proved by a 2-hr run. Another test demonstrated the capability of melting UO_2 at distances up to 20 ft from the high frequency generator.

4. Biology

No columnaris organisms were isolated from over 200 fish sampled during April from four sites along the Columbia and none were isolated from fingerling salmon scheduled for release at the McNary spawning channels. In some years, the fingerlings were heavily infested.

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Two additional swine (over 3 yr old) in the 125 $\mu\text{Ci Sr}^{90}$ /day group developed clinical cases of leukemia (for a total of 10 suspected and/or confirmed in this group out of 26 over 2 yr old) as diagnosed from the peripheral blood. Both of these animals received Sr^{90} in utero and by ingestion after birth. One neoplasm appears to be a lymphocytic leukemia, the other a myelogenous leukemia.

The 2 kg fetus of a ewe sacrificed 113 days after breeding contained 0.86% of the total Cs^{137} burden of the dam or about one-tenth of the average daily Cs^{137} intake of the dam.

Femurs of rats sacrificed 4 mo after exposure to uranium ore dust (pitchblende) contained 5 to 10 times, and livers 25 to 75 times more Th^{230} than U^{238} . Lungs contained about nine times more Th^{230} than U^{238} . All of the data from the study indicate that uranium contained in the inhaled ore particles was selectively translocated from the lung and excreted, whereas thorium was selectively retained.

Eight rats, each given one "insoluble" 1.0 mm, 0.87 mCi Sr^{85} particle by gastric intubation, showed no gross radiation pathology. The particles were excreted after 1, 2, and 5 days and showed a 50% decrease in surface radioactivity.

A lower population of nesting pairs of geese and considerable predation on Locke Island, site of about 35% of the nesting in past years, indicates an all-time low in production of young.

Anaktuvuk Pass residents were counted during April and Cs^{137} body burdens were nearly the same as during January. Average values in 25 control adults were 650 ± 160 nCi.

5. Radiation Protection

Forty-one fishermen were examined in the Mobile Whole Body Laboratory as part of a special off-site study. The fishermen were examined during 3 days of counting at locations adjacent to Casey's Pond and in Columbia Park. Only one fisherman had results appreciably different than the Hanford employee average.

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The effluent from the 100-B Reactor was marked with dye on two occasions in order to measure the time required for the effluent to reach downstream points under low river flow conditions. A few more runs of this type (three from the old reactors and four from N-Reactor) will be carried out after the freshet season in order to complete a series that provides the correlation between travel time and river flow.

Thermolumiscent dosimeters (TLD's) and silicon diode neutron dosimeters were exposed to reactor bursts of about 7×10^{16} fissions. The irradiations were made utilizing the Oak Ridge Health Physics Research Reactor located at the Oak Ridge National Laboratory. Gamma and neutron dose measurements were made in air and on a burro. The TLD and silicon diode results for the neutron dose measurements were in good agreement.

TECHNICAL AND OTHER SERVICES

The average concentration of P^{32} in the flesh of whitefish caught between Ringold and Richland during the month of March was about 60 pCi/g. This was substantially less than the 440 pCi/g measured last year and the range of 100 to 500 pCi/g experienced during the past several years.

Work is continuing on the PRTR high power density core hazards investigation. Accident analysis associated with the proposed PRTR boron shim pilot tests was presented to the Pacific Northwest Laboratories Safeguards Council in a document coauthored by R. J. Smith, L. J. Nitteberg, and T. F. Demmitt. The document describes the boron shim test system, physics parameters, and possible hazards to reactor operation during the tests at <20 ppm boron. The proposed test and report were reviewed by the Safeguards Council.

A decision was made between Battelle-Northwest, U. S. Testing Company, and the AEC to transfer the responsibility for make-up, delivery, and accountability of film dosimeters to U. S. Testing Company.

An extensive set of single-stage sampling plans was developed for use in procurement of reactor hardware.

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Debugging of the EDPM program for the mathematical model of the proposed chemical counter current crystallization process is well underway. Numerical problems have forced the use of some double precision routines.

The EDPM program TREND being written for the process control computer application to a C-column is now approximately two-thirds complete. TREND is that portion of the control program which contains the statistical analysis of the sampled data.

A study was initiated to develop an appropriate multistage experimental design for investigating the contours of the response surface associated with certain waste solidification variables.

An analysis was carried out on experimental data obtained in accordance with a design (1/9 replicate of a 3^6 factorial in blocks of nine) previously recommended. This experimental effort has resulted in the identification of a desired optimal dissolution solvent.

Collaboration with Physical Measurements Unit on authoring an expository paper on stress wave transmission and attenuation in visco-elastic media is now complete.

The analysis was continued on experimental data relative to the Nupac and Vipac subsystems. The second-stage experiment is currently being designed.

A modified version of GEM was completed using weights which reflect instrument instability as well as counting statistic variation. A preliminary evaluation indicates that the modification will extend the range of the GEM program to give valid standard deviation estimates of nuclide decay rates in the greater than 10,000 count range.

An EDPM program to calculate both stream lines and equipotential lines in a two-dimensional ground water flow system has been written, debugged, and placed in service.

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SUPPORTING FUNCTIONS

Plutonium Recycle Test Reactor output for April was 1225 MWd for both an experimental time efficiency and a critical efficiency of 66%. There was one operating period during the month which was terminated manually to begin a scheduled outage. This operating period, which started at 0719 on March 23, 1965 and ended at 0045 on April 18, 1965 for a total run time of 617 hr and 26 min, established a new longest continuous run record for the PRTR. A summary of the fuel irradiation program as of April 30, 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			3	1 218.5	76	17 666.5			79	18 885.0
Maximum				431.0		479.8				
Average				406.1		232.4				
In Basin	7	572.5	30	3 806.1	64	10 201.2			101	14 579.9
Buried							1	7.3	1	7.3
Chemical Processing	68	5 465.8	35	1 965.8					103	7 431.6
Program Totals	75	6 038.3	68	6 990.5	140	27 867.7	1	7.3	284	40 903.8

(Note: MWd/Element x 20 $\approx$ MWd/ton_U for UO₂ and UO₂-PuO₂)

Heavy water loss and indicated helium loss for the month were 1162 lb and 132,082 scf, respectively.

A total of 2086 craft manhours were utilized during the April reactor outage as follows:

Repair	847
Modification and Improvement	562
Operations Assistance	314
Preventive Maintenance	363

Fuel Element Rupture Testing Facility Test #6, Test Irradiation of a Salt Cycle Fuel Element, was completed during April. Examination of the element showed areas of white corrosion product at the ends of the fuel regions and a brown deposit on the lower bracket surfaces and on the first

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inch of the cladding surfaces. No damage was apparent and the element was recharged under PRTR Test #118 to continue the irradiation in the main coolant loop.

Document BNWL-40, "PRTR Fuel Element Rupture Test Facility Capabilities and Project History," by P. C. Walkup, was issued in April.

The chemical processing campaign for spent PRTR fuels again has been cancelled.

A total of 22,745 hr of shop work in Technical Shops Section was produced with a backlog at the end of the period totaling \$206,225 (22,914 hr). To meet emergency requests for service, it was necessary to work 1579 overtime hours or 5.9% of the total hours worked in Technical Shops.

Total productive time in the Laboratory Maintenance Section was 23,400 hr of 24,900 potentially available. Craft overtime worked during April was 2.8% of the total available hours. Manpower utilization (in hours) was as follows:

A. Shop Work		1700
B. Maintenance		7800
1. Preventive Maintenance	2100	
2. Emergency or Unscheduled Maintenance	1200	
3. Normal Scheduled Maintenance	4500	
C. R&D Assistance		13900

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REACTOR & MATERIALS TECHNOLOGY DEPARTMENT MONTHLY REPORT

APRIL 1965

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - O2 PROGRAM

1. Metallic Fuel Development

Alternate Uranium Composition. Operating conditions for driver fuel elements of a single tube design and extended goal exposure require development of a fuel material more swelling resistant than the current 150 Fe - 100 Si N-fuel composition. Previous monthly reports have described evaluation tests that are in progress for materials fabricated to single tube geometry and both beta and gamma heat treated. The alloys being studied are shown in the following table with the calculated volume of second phase compound present in each. Data obtained during the past month on recrystallization and hot hardness of the 350-350-800 alloy are also shown. In the March report it was observed that the materials would be expected to fall into three groups if the volume of small second phase particles are an indication of behavior. It is seen that the hot hardness values do have this relationship and that the recrystallization and hardness recovery data obtained thus far also show a correlation. Stress cycling tests at 400 and 500 C of all the materials are being completed to 5000 cycles.

Studies are in progress to evaluate the irradiation performance of uranium base alloys in terms of composition, fuel fabrication history, corrosion behavior, and irradiation history. An irradiation test of 120 specimens of 17 uranium base alloys containing small additions of Zr, Mo, Nb, Al, Si, Fe, P, and C in various conditions of heat treatment is being conducted in NaK capsules. Calculations based on thermocouple temperature measurements and neutron flux distributions along the NaK capsule columns show that maximum uranium temperatures range from 625 C at the center of the capsule columns down to approximately 500 C at the ends. The temperature differences are a result of an unexpected peak in the reactor flux. A goal exposure of 3000-4000 Mwd/ton is planned for these tests.

Fluted Fuel Element Evaluation. The examination of an irradiated N-single tube fuel element from an ETR irradiation test was started. This fuel element is a companion piece to one which failed during its 13th cycle of irradiation. The elements had received an exposure of approximately 2500 Mwd/ton. The shape of the fuel outer surface was intended to permit some swelling without creating high tensile stress in the outer clad. This was possible because the fluted shape permitted outward deflection of the clad, with only bending strains, up to

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EVALUATION OF URANIUM BASE FUEL MATERIALS

Alloy	Composition			Volume Percent Compound	Hot Hardness Kg/mm ² *				Recrystallization Temperature**		Hardness Recovery Temperature***	
	Fe	Si	Al		300 C		400 C		Beta	Gamma	Beta	Gamma
	ppm	ppm	ppm		Beta	Gamma	Beta	Gamma				
1	150	100	-	0.62	94.6	93.1	60.3	60.3	475 C	480 C	445 C	445 C
2	250	350	-	1.81	98.5	101.7	70.8	75.1	505 C	505 C	465 C	470 C
3	250	250	250	1.69	94.8	96.9	71.7	75.0	505 C	505 C	455 C	465 C
4	350	-	800	1.74	105.0	101.2	68.8	74.6	500 C	525 C	450 C	465 C
5	350	350	800	2.99	129.0	123.5	85.9	90.6	-	-	-	-

*Average of tests run at 400, 600, and 800 pound loads, 4 hours loading, 10 mm WC ball.

**Temperature for 50% recrystallization in 20-hour anneals of 21.4% cold-worked material.

***Temperature for 50% decrease in 20-hour anneals of 21.4% cold-worked material.

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a volume expansion of 11%. The shape also imposed less clad restraint on the uranium and thereby lessened buckling forces on the inner clad.

Examination of the irradiated fuel indicates that the design performed as intended in regard to accommodating swelling; however, unexpectedly large amounts of swelling occurred in some portions of the fuel. Near each end of the element sufficient swelling had occurred to fully utilize the capabilities of the fluted shape. The cause of the high localized swelling near the ends of the element has not been determined.

N-Reactor Fuel Support Development. Fatigue curves were nominally completed for 1/4, 3/16, and 1/8-inch wide tube-tube inner supports. Endurance limits expressed as reciprocating load for the three sizes of supports are approximately 280, 170, and 70 pounds, respectively. Fatigue data are currently being developed for the target fuel support assembly.

Target Element Development. A study to determine the effects of a high temperature reactor excursion on Zr-2 clad Al-Li alloy target elements is continuing.

Examination has been completed on an irradiated Al-Li target element double canned in aluminum and Zr-2 that failed during an ex-reactor heating test. This test did not follow the pattern of previous runs in that failure was observed at a lower temperature and two areas of failure were observed. Metallographic examination has established that the core in the region of both failures had melted, and both ruptures had occurred where target support tabs had apparently been torn off before the heating test.

Two samples have been provided to evaluate the possible use of a high cross section poison in the center of a target rod that would be dispersed when a target element is overheated. Irradiated target rods of Al-Li alloy were drilled to provide 0.125 inch diameter axial holes which were then filled with boron and boron carbide and recanned in Zr-2. Reactivity measurements before and after ex-reactor heating to rupture will establish whether or not dispersion of the buried poison will compensate for the loss of Li-6. An additional test element using a 50% Cu - 50% Cd alloy as buried poison is being prepared.

Elevated Temperature Reactions Between Cadmium Alloys and Zirconium. Previous studies on elevated temperature reactions between various poisons and Li-Al target core materials have shown a possible reaction between the cadmium alloys and the zirconium cladding. In order to determine if any reaction exists, several zirconium cans containing pure cadmium and several containing a Cd - 50% Cu alloy were welded closed and will be heated to various times and temperatures. Half the cans were in the bright-etched condition and half were autoclaved to also determine the effect of the oxide film on the reaction.

Compaction of Lithium Aluminate Powder. Lithium aluminate powders have been compacted to 72% theoretical density by cold pressing and 83% theoretical

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density by hot pressing at 500 C and 800 C. It is planned to compact these materials to higher densities by pneumatic impaction at temperatures up to 1350 C.

2. Corrosion and Water Quality Studies

High Temperature Oxidation of Zircaloy-2. Interest has recently been expressed in the high temperature oxidation of Zircaloy-2 by water vapor. Zircaloy-2 coupons (1 x 0.5 x 0.03 inch) have been exposed to a 25 torr water vapor atmosphere at temperatures of 900, 1000, and 1100 C. The kinetics of reaction at each temperature have been expressed by a parabolic fit (i.e., the weight gain expression is $[\Delta W/A] = kt^{1/2}$). Assuming parabolic kinetics, the rate expression for reaction at these higher temperatures is

$$k \text{ (mg}^2\text{dm}^{-4}\text{min}^{-1}\text{)} = 1.14 \times 10^{12} \exp (-46,000/RT).$$

At each of these temperatures it was noticed that breakaway did not occur and that the scale formed was adherent. The thin samples did deform during testing; the 1100 C sample was brittle, while the 900 and 1000 C samples were ductile at the end of the exposure.

Corrosion Evaluation of Hydrided K-Reactor Zircaloy Process Tubes.

It was found that the oxide thickness on K-Reactor Zircaloy-2 process tubes was not thick enough to be measured by metallographic techniques. Anodizing methods for Zircaloy-2 were then investigated as analytical tools for oxide thickness measurements.

In 1% NaOH solution a direct relationship exists between the anodizing voltage and oxide thickness on Zircaloy (0.049 mg/dm²/volt). Anodization of Zircaloy will not occur until a sufficient potential is applied to overcome the electrical barrier formed by an existing oxide film. Therefore, oxide thicknesses on Zircaloy could be measured by determining the potential required to overcome the electrical barrier formed by the existing oxide and converting this value to equivalent oxide thickness.

To date, results have been obtained by this method on Process Tube 3075. Two pieces of this tube have been measured--one from the heavily hydrided portion and one from the in-flux portion. The respective oxide layers were 0.25 mg/dm² and 0.5 mg/dm².

The oxide layer of 0.25 mg/dm² for the hydrided portion of the tube was measured on the outside diameter. It was not possible to anodize the hydride layer on the inside diameter.

N-Reactor Fuel Element Rupture Tests. The rupture rate and severity following a fuel element cladding defect is being evaluated in the Irradiated Rupture Prototype (IRP), a shielded high temperature loop.

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Special size KER-irradiated N-Reactor prototype fuel elements are pre-defected in the Radiometallurgy Facilities and placed in a shielded test section which can be attached to the loop. After the water temperature in the loop is at the approximate surface temperature of typical fuel elements in the reactor, the shielded test section is valved in and the water contacts the defected fuel element. This water temperature is maintained until the fuel element releases core material, as determined by activity monitors. The water temperature is then lowered at the N-Reactor shutdown rate, about 3 C per minute, following a 5-minute hold at 300 C. The amount of core material release is followed during the test by activity monitors and is determined after the test by weight measurements. Other pertinent information is obtained by visual inspection and measurements before and after the test and by continuous activity measurements in different portions of the loop.

Five tests have been performed (during this reporting period) with KSE-5 elements (single-tube, thick-walled, 1.6% enriched to evaluate swelling behavior at high uranium temperatures) with a single 0.025" hole in each. The tests were performed with fuel elements containing two different core additions to compare in-reactor swelling rates. These were the standard N-Reactor fuel composition with 120 to 140 ppm of both iron and silicon added to the uranium and the second type was with 400 ppm iron and 800 ppm aluminum added to the uranium core.

The first two IRP tests used elements with essentially the same exposure (~1300 Mwd/ton) and temperatures (470 C maximum fuel temp) but each with the different core material. Both of the elements failed in a similar manner. Each had an incubation period (time at temperature before activity release) of 145 to 168 minutes and preliminary weight losses of 4 grams before cleaning. The rupture consisted of a single raised and torn blister about 5/8 to 3/4" in diameter.

The next two elements were exposed to 1670 Mwd/ton and had a maximum fuel temperature of 580 C. The element with the iron and aluminum core additions had a 255-minute incubation period and a 5-gram weight loss. The rupture again consisted of a single raised and torn blister about 3/4" in diameter. The standard N-Reactor core addition element ruptured quite severely. After an 81-minute incubation period, the activity buildup in the loop filter was quite rapid and reached 65 R/hr at the end of the test. Preliminary uncleaned weight losses for the element was 133 grams. The rupture had extended for 6" in length and about 2/3 of the way around the circumference of the element for the 6" length. This ruptured area was covered with many raised and torn areas. Considerable difficulty was experienced in removing the fuel element from the shielded test section due to the swelling.

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The fifth fuel element exposed for 2650 Mwd/ton and a maximum fuel temperature of 566 C, and with iron and aluminum additions has not failed after a 200-minute incubation period. This test is continuing.

Reprecipitation and Deposition From Decontaminating Solutions. Two proprietary reagents were tested to determine the changes in appearance and amount of redeposition from decontaminating solutions held at operating temperature for extended periods of time. Samples of solutions containing coupons of carbon steel and stainless steel were heated to operating temperature for 66 hours.

A solution of a proprietary oxalic acid solution (Turco 4521), at a concentration of 60 g/l and operating temperature of 75 C, darkened with time. Precipitation was noticed first after 66 hours and increased in amount until the end of the test.

A proprietary sulfamic acid solution (Turco 4306-D), at a concentration of 90 g/l and a temperature of 61 C, also darkened with time. No precipitation was observed while the solution was heated, but upon cooling, a precipitate formed in the bottom of the sample bottle. The inhibitor had not given any indication of separation in the heated solution but was observed to be floating in a cooled sample drawn after 42 hours at temperature.

Evaluation of Film Removal from N Steam Generators. Two steam generators at N-Reactor had been decontaminated using a 2-step procedure alkaline permanganate, followed by Turco 4306-D, a proprietary sulfamic acid solution. Poor technique resulted in excessive dilution of the alkaline permanganate, and the films and contamination were only partially removed from the stainless steel tubes. A visual examination of several pieces of tubing removed from the steam generators revealed that partial cleaning and film removal had been obtained. The film was a reddish-brown color instead of the normal black color. In areas near the tube sheet, the film had been removed. In areas further from the tube sheet (between Baffles 2 and 3), the film was partially stripped and still further from the tube sheet (between Baffles 4 and 5), the film appeared to have only changed color from black to reddish-brown.

Fretting Tests on N-Reactor Fuel Elements. Two special N-Reactor fuel elements, assembled with inner (buggy) springs having spring constants 40-70% of that normally used were exposed in the N-Reactor test section of TF-7. The purpose of the test was to determine if relative movement between the spring and outer fuel element would result in fretting corrosion of the fuel element cladding. Temperature and flow rate during the 125-hour test were 277 C and 32 gpm, respectively. Amplitude (peak-to-peak) and frequency of vibration (resulting from externally applied vibration) at the approximate midpoint of the test section process tube were 0.001-0.002 inch and 54-68 cps, respectively. Visual examination after exposure revealed no fretting corrosion.

Chloride Concentrations in Crevices in N-Reactor Steam Generators.

An extensive tube examination program is being conducted to monitor the severe intergranular attack previously found in the N-Reactor steam generators and to determine if new types of corrosion are occurring. As part of this program, analyses for chloride were performed on the OD of 20 tube ends removed from the tube sheets during the March 1965 outage and the data were compared with data obtained from previous analyses. Present chloride concentrations are 0.03 to 0.13 micrograms/cm² now, as compared with concentrations of 0.95 to 1.9 micrograms/cm² in 1963. There were no obvious differences in chloride levels among the six steam generators sampled or between leaking or nonleaking tubes. X-ray fluorescent measurements indicated the chlorides were uniformly distributed over the tube surfaces. The chlorides were distributed primarily on the outer surfaces in 1963, but are not occluded deep in the corrosion product films.

It is concluded that the chlorides that were originally present from the manufacturing process are being leached out and that chlorides from the water are not concentrating in the crevice areas. This is fortunate because if the chlorides had been concentrating in the crevice area, this could possibly lead to chloride stress-corrosion cracking. Whether or not the chlorides found were a factor in the intergranular attack in the crevice area is not known.

Hydride Formation in Zircaloy Tubes. Severe hydriding has occurred in a number of the Zircaloy process tubes in the K-Reactors and may require replacement of many tubes unless the causes of this hydride formation are identified and ways are found to repair existing damage and prevent further hydride formation. Factors which may contribute to this hydride formation include: (1) damage to thin Zircaloy oxide layer from scraping by discharging fuel, (2) deposition of aluminum on bare Zircaloy surface with subsequent corrosion reaction to produce hydrogen, (3) fretting damage of Zircaloy oxide from vibratory contact with aluminum spacers, and (4) slight possibility of fluoride decontaminant effects on the Zircaloy surface. Present investigations to determine the cause of this Zircaloy hydriding problem include fretting tests, exposure and analyses of pretreated coupons under typical reactor conditions, and corrosion probe development.

Fretting may be the major or a contributory cause of the attack. The spacers are relatively light in weight and under heavy load. External vibration of the tube, if present, may cause true fretting with destruction of the protective ZrO₂ film. This could then lead to accelerated attack of the bare metal by the coolant or even to galvanic attack. Fretting is being evaluated in 1706 KEL Laboratory in a special test facility designed to simulate the small movement under heavy load. The samples are mounted in a heated tank, maintained at reactor effluent temperatures, and relative movement between the aluminum spacers and the Zr-2 process is achieved by externally applied vibration. The

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test is scheduled to continue for 18 weeks or nine operating periods. The hydride concentration of sections of the Zr-2 process tube sample will be determined after each operating period.

Pretreated coupons are being exposed to pH 6.6 process water in out-of-reactor facility TF-20 at 120 C. First samples, discharged after 360 hours, are being examined visually and analyzed for hydride. Similar pretreated coupons were installed in KE-Reactor tubes for exposure to standard process water and to water at different concentrations of pH, dichromate, and peroxide. Zircaloy coupons were discharged from KE single-pass tube 2952 after 14 months of exposure in 130 C process water. These samples and other Zircaloy samples exposed 3 to 12 months in other single-pass facilities will be analyzed for hydriding.

CO₂ for pH Control. Testing of pH adjustment to 6.6 by CO₂ addition is being performed in the 1706 KE in-reactor twin-tube facilities to evaluate effluent radioactivity and corrosion effects. Laboratory data have shown reduction in aluminum and carbon steel corrosion under these conditions. Savings in chemical costs might be possible by use of stack gas CO₂ to replace sulfuric acid for pH adjustment of reactor cooling water. Coupon corrosion samples and corrosion probes were installed in 1706-KE facilities SP-3, 4, X & Y for exposure in CO₂ treated water. Exposure periods will be one and three months, respectively.

3. Gas-Atmosphere Studies

Burnout Monitoring. Since the early part of January 1965, F-Reactor has operated on Production Test IP-725, which permits a 50 C increase in graphite temperature above the 735 C limit. On the basis of acceptable rates from oxidation monitors discharged February 3 and March 30, 1965, approval was obtained to raise the temperature limit another 50 C. At the next reactor startup, probably in mid-April, this supplemental production test will become effective.

Profiles of the burnout rates showed the absence of any O₂. The only peak appeared near the stack centerline where the temperature and the flux were a maximum. The peaks for the two tests were ~1% per 1000 operating days (%/KOD) in Channel 1067 and ~2%/KOD and ~3%/KOD for the first and second tests, respectively, in Channel 3580. In both tests monitors which were protected from the reactor gas atmosphere by a 3/8-inch thick wall of graphite oxidized at the same rate as exposed monitors, indicating a homogeneous reaction.

Monitors in Channels 1889 and 1960 at C-Reactor from January 25, 1964 to March 7, 1965, had rates below 0.6%/KOD. There were no indications

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of the presence of O₂, and the protected and exposed monitors had equivalent burnout rates.

Graphite-Zirconium Compatibility Loop. Tests Nos. 10, 11, and 12 have been completed in the Graphite-Zirconium Compatibility Loop at a maximum temperature of 820 C.

Test No. 10 operated for 455 hours at a water-vapor pressure of 4500-5000 ppm. The buildup of gaseous products was followed by gas chromatography. Within 48 hours of startup the H₂ and the CO had exceeded 1%. A mass spectrometer analysis taken at that time showed 1.43% H₂, 1.01% CO, 0.26% CO₂, and O₂, N₂, and CH₄ all less than 0.01%. These results, together with a few analyses made during Test No. 9, suggest that the water-gas shift reaction:

is being catalyzed. Appearance of the bar, especially where it contacts the Hastelloy-X, indicates that the Hastelloy-X may be catalyzing this reaction.

In Test No. 11 the water concentration was less than 80 ppm and hydrogen 1000 ppm. This test was designed to see if hydriding would occur under conditions where small lab experiments predicted hydriding. Hydriding was observed and will be reported elsewhere. Graphite oxidation was negligible ($<10^{-7}$ g/g/hr) as expected.

In Test No. 12 the carbon dioxide concentration was approximately 2% and water less than 50 ppm. Oxidation rates of 5×10^{-7} to 5×10^{-6} hr⁻¹ were observed for the surface samples. These rates are comparable to those obtained in Test No. 10 for the high-water-vapor concentrations. The carbon dioxide oxidation was more uniform over the interior samples than the water-vapor oxidation.

Gas-Graphite Reactions. The thermal reactivity of N-Reactor graphite (TSX) with dry air at atmospheric pressure was determined in the temperature range from 625 to 727 C. The samples were oxidized from 10 to 50% weight loss. The rate can be expressed by the equation,

$$R \text{ (g/g/hr)} = (2.29 \pm 0.04) \times 10^{10} e^{\frac{-51.8 \pm 1.9}{RT}}$$

The standard deviations for the pre-exponential factor and the activation energy are shown in the equation. The data were collected from 11 different samples and at seven different temperatures for a total of 21 individual determinations. The value for the activation energy agrees well with the value of 49.9 kcal/mole previously reported for the temperature range 497 to 630 C.

A comparison of earlier work and these data suggests that the reaction between TSX graphite and O₂ in the temperature range from 497 to 727 C

occurs uniformly throughout the whole sample, i.e., the reaction is in Zone I where the rate is independent of the surface-to-volume (S/V) ratio. The sample geometry in the earlier work was a parallelepiped with dimensions from 2 x 8 x 20 mm and a S/V ratio of 1.3. The geometry in this latest work was solid cylinders with dimensions of 0.43 dia. x 4" long and 0.43" x 1" long with S/V ratios of 9.9 and 11.4, respectively.

At 630 C the reaction order with respect to the O₂ concentration was previously found to be approximately 0.5 over the concentration range from 0.2 to 1%. Using the value for the reaction order and extrapolating the reaction rate to the concentration of O₂ in air, excellent agreement was obtained with the reaction rate measured in this latest work.

4. Process Tube Development

Irradiated Tube Testing Facility. A report entitled "Irradiated Burst Test Facility," BNWL-6, describing the facility and its testing capabilities, was issued this month. This test facility, which became operational in March 1964, has operated favorably since.

K-Reactor Welded Zircaloy-2 Process Tubing. A series of burst tests at room temperature and elevated temperature was made on sections of welded Zircaloy-2 tubing for the Hanford K-Reactors. In general, the strength of the tubing compares favorably with extruded and drawn tubing. At room temperature, failure is in the weld zone and at 300 C the greatest strain is in the welded heat affected zone, but failure is opposite the weld zone. Report BNWC-51 describing the tests has been issued.

5. Thermal Hydraulics Studies

N-Reactor Boiling Burnout Studies. Fourteen boiling burnout points were obtained in the study of heat transfer conditions of co-producer fuel elements for N-Reactor. The experiments were conducted in the heat transfer laboratory with an electrically heated model having the geometry of the co-producer inner annulus; i.e., a horizontal coolant annulus with a heated outer wall and an unheated inner wall. The annulus width was 0.2765 inch with an inside diameter of the heated outer wall of 1.775 inch and an outside diameter of the unheated inner rod of 1.222 inch. The purpose of the present experiments is to investigate possible effects of a heated surface versus an unheated opposing surface on burnout behavior. Future experiments, with models having dimensions more closely matching the co-producer fuel, will investigate burnout behavior in the entire co-producer fuel and the effects of dimensional changes.

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The fourteen burnout points were obtained at 1500 psig with mass velocities of 0.5, 1, 2, 3, 4, and 5 million lb/hr-sq ft and with inlet temperatures of 590, 560, and 530 F (310, 295, 275 C). Outlet conditions at burnout ranged from about 25 F (15 C) subcooling to 10% steam quality. Thirteen of the burnouts were shown by abnormal temperature rises indicated by thermocouples used as burnout detectors. These burnouts occurred at or near the downstream end of the 24-inch long heated section. The fourteenth point resulted in physical failure of the model at a point about 3 to 10 inches from the downstream end. The failure, a 1-1/4-inch long crack in a bulged, discolored area, occurred about two inches downstream of a set of ceramic supports separating the inner and outer tubes.

In general, the results of these experiments seem to agree quite well with those for a two-heated-surface model run previously. A preliminary examination of the data indicates that burnout heat fluxes were about the same for the two models, at comparable enthalpies and mass velocities. The enthalpy range covered in these experiments, however, was rather limited. Plans were made to repair the model and conduct experiments which will extend the comparisons to greater subcooling.

Hydraulic Tests. Visual flow tests were conducted in the Hydraulics Laboratory to determine the vibration response of an N-Reactor target element. The element is the innermost element of the N-Reactor fuel column and is supported in the uranium fuel piece by nine buggy-spring bumpers along the length and periphery of the target element. The initial design value for the spring constant was to be 0.002 inch deflection per pound force but, because of etching, the value was reduced one-half. With such a low spring constant it was considered possible that the assembly might reach resonant frequency at the flows which would be encountered in the target element channel.

Tests performed in the laboratory with the 1.360 inch OD element (with rails) supported in a 1.345 inch ID pyrex tube indicated that excessive element vibrations would not be encountered up to flow rates of 35,000 lbm/hr (in-reactor mass flow rate = 33,600 lbm/hr).

The relative frequency and amplitude response of tube and element were measured using a strobe-light, reed tachometer, and grid microscope. High speed movies to frame speeds of 8500 frames per second also were obtained. Visual observation with the above equipment indicates 3 to 4 mils amplitude and 2100 cycles/min of tube and element vibration at 15,000 lbm/hr and 6 to 3 mils amplitude and 3200-3500 cycles/min at 35,000 lbm/hr. The latter frequency represents the approximate pump frequency.

Relative deflection between tube and element was difficult to measure and was not visually detected due to its small amplitude and the variable frequency response of the system.

6. Graphite Studies

N-Reactor Irradiations. The two capsules currently in the GETR, H-5-5 and H-6-5, are operating satisfactorily. The faulty connections made in the terminal box during the installation of H-6-5 have been corrected, and all thermocouples in both capsules are functioning properly. All sample temperatures are in agreement with previous capsule history.

The report, HW-81250 A, "The H-4, H-5, H-6 Irradiation Experiments: Irradiation of N-Reactor Graphite--Interim Report No. 1," on the data obtained from the first nine capsules in the program, H-4-1 through H-6-3, has been issued. Results are presented for samples irradiated to a maximum neutron exposure of 4.8×10^{21} nvt, $E > 0.18$ Mev. The TSX samples all showed continued contraction with increased neutron exposure, and to the exposures reported, the TSX graphite exhibited nearly the same contraction behavior as the reference CSF graphite in both the transverse and parallel orientations.

The N-Reactor graphite tests in the GETR will be terminated by June 30, 1965. Investigations are under way to determine the feasibility of irradiating these samples in the C-Area Reactor at the Savannah River Plant during the high-flux operation there. The fast flux ($E > 0.18$ Mev) is of particular interest for the graphite irradiations, and possible methods for determination of this flux by both calculational and measurements techniques are being examined.

7. NRD Shielding Studies

Shield Cask Code. The shield-cask code is presently being debugged. The code has been on the computer with a sample problem (a shield cask of rectangular geometry) and has output in the correct format; however, problems still exist in the calculational routines. The sample problem has an existing hand calculated solution. When the code yields a satisfactory solution to the sample problem, the code will be further checked by solving a cylindrical shield cask design problem for which a hand solution also exists. During this period a special isotope (isotopes other than isotopes produced from direct fission yield and subsequent decay) library was compiled for use with the code. Also completed were the buildup factor and mass attenuation coefficient libraries.

B. WEAPONS - 03 PROGRAM

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

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR Fuel Element Performance. Two discolored areas on the cladding surface of the defect rod (DI-20) from PRTR FE-5237 were identified as zirconium hydride by radiometallurgical examination. The element failed in January 1965, after reaching an exposure of 1×10^{20} fissions/cc (3420 Mwd/ton fuel). The fuel contained 40 ppm oil, but this rod was reported to have had three discolored spots on the cladding after autoclaving. It has not been determined if the hydriding originated on the internal or external cladding surface.

The first 2 wt% PuO_2 element, containing three rods of salt cycle fuel material, was discharged from the PRTR rupture loop and examined. One band on the downstream end was twisted and was replaced. The element is ready for recharging into the reactor.

A salt cycle fuel rod was successfully irradiated in the MTR and is presently awaiting postirradiation examination.

Fuel Element Refurbishing. Two PRTR irradiated fuel elements were repaired in the PRTR basin. One element had a broken wire wrap which was successfully repaired. The other element had a number of broken and missing bands, and this element was rebanded with three new bands.

Fuel Recycle Pilot Plant. Design and procurement of remotized fuel element refabrication equipment, for initial recycle operation, is estimated to be 20% complete. Purchase requisitions have been issued on all capital equipment to be procured for this purpose from FY-1965 equipment funds. Installation is scheduled to begin as soon as FRPP (324 Building) construction progress will allow.

Spheroidization Studies. Agglomeration techniques for UO_2 and UO_2 -plus-additives are being evaluated to produce feed materials for vibratory compaction of fuel elements. Five different starting materials and processes have been studied to date.

1. Low density UO_2 fines and nuggets were agglomerated into spheres by ball milling using large (2-15 mm diameter) UO_2 chunks for the mixing action.
2. High density UO_2 fines and nuggets were agglomerated by the technique used in (1) above.
3. Calcium oxide and micronized UO_2 were combined and ball milled using large UO_2 chunks for the mixing action. The calcium oxide appeared to accelerate the agglomeration action.

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4. Micronized molybdenum and micronized UO_2 were combined using technique (3) above; however, the resulting agglomerates were rough. Subsequent rolling treatment on vibrating emery cloth produced hard, smooth agglomerates.
5. (a) Larger (2-6 mesh) spheroidized particles have been produced by adding a binder to micronized UO_2 forming large granules which were then rolled by hand in trays to generate spheroids.
(b) The same treatment as (a) except the starting material was a -300 mesh arc fused UO_2 and after the granules were formed, the green chunks were vibrated in a tray of micronized UO_2 .

All of the above material requires a sintering treatment as the final densification step. Final densities ranging from 74 to 98% of theoretical density have been produced. Particle sizes varying from +2 to -325 mesh have been made by one or more of the above processes.

PRCF Ultrasonic Crack Testing. Eight samples of 20 PRCF fuels were ultrasonically tested for cracks in the welded portion of the fuel rod at the clamped end. One small crack was detected in one of the samples. The remaining fuel rods in that batch were tested and another small crack was detected. Due to tight time schedules, test standards could not be fabricated for the PRCF fuel geometry so the exact size of the cracks is questionable. It is felt, however, that the cracks are insignificant for PRCF use and none of the fuel was rejected.

Fuel Cycle Analysis. A test case for the cost of impacted, vibrationally-compacted mixed oxide fuel elements has been successfully programmed through the Fuel Element Fabrication Code. The numbers submitted for the building cost, overhead costs, etc., are highly tentative. When more accurate values for the reject rates and overhead charges are substituted, the calculated costs will be more realistic. For purposes of comparison, a case is being prepared for a pellet fuel element.

PRTR High Power Density Core Fuel Elements. The maximum burnup on a UO_2 -2 wt% PuO_2 HPD element in PRTR is 3030 Mwd/tony with an average of 1775 Mwd/tony for the seven HPD elements presently in the core. These burnups have been accumulated at maximum rod powers of 19 kw/ft which produce fuel temperatures as high as 2600 C at the mid-plane of the elements.

Postirradiation examination of the center rod and rods from the six-rod ring from HPD FE-6000 (maximum rod power 19 kw/ft) indicates that the power generation was appreciably lower in these inner rods than in the outer rods from the 12-rod ring. This was also indicated by the lower gas release from the inner rods. No central void was formed at the mid-plane of the inner rods from the element and only a small tear-drop void

formed in the bottom of the rod from the six-rod ring. These fuel structural effects were expected since calculations indicate that the rod power generation rate in the six-rod ring and the central rod is only 55 and 42%, respectively, of the rod power generation rate in the 12-rod ring.

Preliminary calculations indicate that a plenum will be necessary to accommodate the gases released during high temperature operation of HPD elements to prevent excessive internal pressures.

Fuel Element Rupture Test Facility. There are plans to irradiate a HPD element in the FERTF at a maximum rod power of 22.5 kw/ft (515,000 Btu/hr-ft²). This power will yield fuel temperatures above melting with as much as 30% of the cross-sectional area molten.

Corrosion and Water Quality Studies

Boron Recovery From Used Resin. An effective (but not necessarily optimum) regeneration technique was developed to recover H₃BO₃ from the anion resin that will be used to remove H₃BO₃ at PRTR. The procedure utilizes Na₂SO₄ or H₂SO₄ as the regenerant and provides quantitative recovery of H₃BO₃ in 10 bed-volumes of regenerant. Recovery of H₃BO₃ from the regenerant can be effected by sulfate precipitation. An alternate technique utilizing NH₄OH as regenerant will be studied to attempt simplification of recovery from the regenerant. A simple, effective regeneration-recovery process will definitely be required if B¹⁰-enriched H₃BO₃ is used as the chemical shim additive at PRTR.

H₃BO₃ Ion Exchange Studies. A laboratory model of the PRTR moderator and H₃BO₃ removal ion exchange system was assembled and operated under conditions similar to those proposed for the Phase 1 tests. Two tests were performed to demonstrate the H₃BO₃ control capability. The system was then disassembled and the boron concentration profile through the ion exchangers was determined. Results demonstrated that the system design is entirely satisfactory for proposed Phase 1 operations.

Additional work is in progress to determine resin boron capacity values under dynamic conditions and to evaluate the factors controlling boron elution in the resin beds. Results from these studies will be used to define the performance of the H₃BO₃ removal exchangers and the performance of the borated IX-3 exchanger during Phase 2 operations.

H₃BO₃ Prototype. Installation of all prototype components in the PRTR was completed. Design Acceptance Tests (using P-10) are currently in progress and should be completed by the end of April.

Localized Corrosion in Shim Wells. A pilot plant corrosion test is in progress in the EDU Shim Test Loop to determine the corrosiveness of 140 F, 50 ppm H₃BO₃ to a PRTR shim rod and shim well. The shim rod is in

the lower position so that it is in contact with the shim well walls. A flow of 3 gal/hr is maintained past the shim rod. Examination of the shim rod after two and four weeks of exposure revealed very little generalized corrosion; however, numerous white deposits resembling aluminum corrosion product were observed on the shim well walls after four weeks. Since these deposits resemble the deposits which covered pitted areas during previous testing, the shim well will be removed for destructive examination. Corrosion measurements on coupons of aluminum and stainless steel after two weeks of exposure showed negligible penetrations.

High Radiation Levels in PRTR Primary System. PRTR primary system radiation levels have been increasing. Shutdown radiation data show that the buildup rate accelerated in August or September 1964, after being essentially constant since the PRTR decontamination. This accelerated buildup rate has continued into the Spring of 1965. Work has been started on defining the variables affecting PRTR radioactivity buildup rates. The following summarizes program progress during the past month:

- (a) Crud Levels. A crud probe installation has been used to obtain a primary system sample for radiochemical studies. Preliminary results show the presence of Co-60, Fe-59, and Sb-124. Other species are present, but they have not yet been identified. Since a quantitative study of coolant borne crud levels is dependent upon knowledge of flow, a flow measuring device is to be installed on the crud probe equipment at PRTR in late April.
- (b) Radioactive Films. Two nozzle cap coupons are being discharged to provide data on the identity of radioactive species retained in PRTR primary system films. Supplementary data are being obtained from PRTR components, but these sources are inadequate for all but qualitative measurements. Sufficient specimens for a detailed study of adsorbed activity and loose crud levels will not be available until the By-Pass Flow Loop becomes operational. Completion and operation of this facility has been postponed until June 1965.
- (c) Components. Inspection of various PRTR components continues. Several fiberglass filters from various PRTR streams have become available during the past month. Five appeared to be predominantly loaded with ion exchange resin. One was coated with a heavy black deposit which has been identified as organic material.

PRTR Crevice Corrosion Tests. Out-of-reactor tests to study corrosion in heated crevices, with geometries similar to those existing on the heat transfer surface of PRTR fuel elements, continued during the month. Microscopic examination of sections from the heated Test Assembly No. 2

(described in the March report) verified that the crevice geometry resulting from the addition of "cluster" bands to the Zircaloy-2 clad heating units is conducive to accelerated, localized corrosion.

Test Assembly No. 2 had been exposed in lithiated water at pH 10 for 60 days with a surface heat flux of 550,000 Btu/hr-ft² surface. From a preliminary examination, it appeared that corrosion was even more extensive than that associated with an initial specimen exposed to the same heat flux for 70 days. Sectioning showed that this was not the case; maximum heating-unit surface penetration, where "cluster" bands had contacted the heated length of the initial and second specimens, was approximately 0.007- and 0.004-inch, respectively. Microscopic examination of sections from Test Assembly No. 2 also revealed heating unit penetration where wire wraps had contacted the heated length. In some locations the penetration associated with wire wrap contact areas, up to 0.002 inch deep, was completely filled with gray-white corrosion product deposit. Corrosion of the "cluster" band was limited to crevice areas where the band contacted the heated length of the heating unit.

PRTR Corrosion Monitoring. Continuous monitoring of corrosion in various PRTR systems was proposed at the time the reactor was being built. This monitoring was to detect corrosion changes and provide a continuous history of corrosion in the shield, reflector, and moderator system. Provisions were made during construction for later installation of probes. Accordingly, four probes were installed in 1961: bottom shield inlet and outlet, reflector, and moderator. Two of these probes apparently did not provide any initial readings. One probe provided an initial reading but is no longer operable and one probe of 6061 aluminum is still operable and indicates an average corrosion rate of 0.2 mil per year in the reflector system. Replacement of the other three probes is now scheduled for the present outage and additional probe installation is planned for the May outage.

PRTR Pressure Tubes

Zr-Nb Pressure Tubing. Brittle fracture tests at room temperature on the 30% CW tubing were started this month. Tests employing a 1-1/2-inch long slot were completed. In the first test with a 1-1/2-inch long slot milled to a depth of 80% of the original tube wall thickness, failure occurred at an internal pressure of 5460 psi, and there was no crack propagation. This specimen was subsequently resealed and repressurized. Failure occurred at an internal pressure of 7060 psi. Crack propagation was substantially less than for Zr-2 in a cold worked condition.

An irradiated specimen having cold worked microstructure was tested for crack propagation resistance. The specimen failed at a nominal stress of 26,000 psi at room temperature. The specimen's average exposure was

1.2×10^{21} nvt. Another similar specimen had failed at a nominal stress of 24,000 psi, but it had an average exposure of 6×10^{20} . The 100% increase in neutron exposure is indicated to have produced an 8% increase in crack propagation stress. This is the second instance of an indicated increase in crack propagation stress accompanied by an increase in fast neutron exposure. In both cases the stress increase was of the order of 8% and the exposure increase was about 100%.

Reactor Components Development

Fretting Corrosion Investigation. Additional in-reactor tests have been conducted to evaluate zirconium concentration in the primary coolant system. As in the first test, the data show the primary system zirconium content as the steam pressure was increased from 335 psig to 365 psig and maintained for an extended time period. Primary bulk and channel 1653 coolant samples were periodically drawn for chemical analysis. There appears to be a definite increase in zirconium concentration with time, averaging better than twice as high in Channel 1653.

Inlet jumper vibration data from Channels 1653 and 1354 were taken during the test and the applicable traces analyzed. These data indicate that the amplitude of the Channel 1653 inlet jumper increased approximately 10%. When the steam generator pressure was increased to 365 psig, Channel 1354 initially decreased approximately 12.5% after which the amplitude increased slightly to reflect an approximate 10% decrease based on the amplitude of vibration at the start of the test. These amplitude changes are approximately one-half as much as occurred previously under similar test conditions. Similarly, the increases in zirconium content are considerably less than noted previously. Comparison of these data with those taken during the prior primary coolant pressure (temperature) effect investigations indicate that the ratio of zirconium concentration at a pressure of 365 psig, to that at 335 psig, varies directly as the ratio of inlet jumper motion of channel 1653 at 365 psig to that at 335 psig.

Fuel Re-use Study. Construction of the liquid sodium loop for the fuel re-use experiments is complete. Two hundred pounds of sodium have been charged into the loop storage tank and shakedown runs of the loop are in progress. The over-all Liquid Metal Facility has some remaining work to be done on peripheral equipment before it is complete.

Those systems of the sodium loop that it is possible to operate without circulating sodium have performed without difficulty.

Gasket Study. A test program for improving the performance of the spiral-wound asbestos-filled gasket seals on the PRTR pressure tube assemblies has been formulated (BNWC-30), and testing is under way. Low chloride asbestos, stress relieved stainless steel winding, and substitution of Inconel for stainless steel are being investigated as means of minimizing or eliminating stress corrosion cracking; thermal and autoclave treatments are being investigated as a means of avoiding the apparent gasket "relaxation."

The gasket testing station duplicates the PRTR nozzle cap, tube-nozzle and inlet jumper-tube closures. Five simulated PRTR thermal cycles have been completed on two sets of Inconel wound, and one set of stainless steel wound, asbestos-filled gaskets. On the inlet closure, there was no leakage from any of the gaskets until the 5th cycle when one of the Inconel gaskets leaked 2 ml/hr; on the tube-nozzle closure, the Inconel gaskets had no leakage, but the stainless steel gasket consistently leaked 0.5-3 ml/hr; and on the cap closure, the Inconel leaked 2-7 ml/hr, while the stainless steel essentially did not leak.

The holding torque of the bolts for each fixture was determined after the third thermal cycle. The torque on the bolts at the nozzle cap and tube-nozzle closures were the same as made at installation. However, the torque on the bolts at the inlet-tube closure of all three test stations had decreased from the original 45 ft-lbs to values between 26-45 ft-lbs with an average of 33 ft-lbs.

Design Analysis

Plutonium Utilization. Further examination of ZODIAC-2 input data for the calculation of Yankee Core 1 burnup has disclosed several minor errors; these have been corrected. In addition, the saturable fission products Xe-135 and Sm-149 were included to obtain a better estimate of the neutron spectrum during burnup. However, the net result of the changes made in the calculation was not sufficient to bring the reactivity loss with burnup into agreement with measured values. A search for additional discrepancies has been resumed.

Thermal Hydraulic Studies

Tests were run in the Hydraulics Laboratory to determine reasons why graphite has been lost from a crud sampler used to analyze for particulate matter in the PRTR flow system. The sampler consists of a porous basket containing 10-15 grams of powdered spectroscopic graphite which filters a 0.15 to 2.0 gpm stream of the process water operating at 1050 psig and 520 F.

During several of the sampling periods at the reactor, the graphite has been lost, apparently having been washed into the primary system. The

nature of the graphite loss was not determined; i.e., whether the graphite was being washed around leaky sealing gaskets and into the stream, through the porous stainless steel basket bottom, or whether it was lost through voids in the weld around the periphery of the porous bottom was unknown. Tests performed in the 189-D laboratory indicate that a gasket failure would be evidenced by visible indications around the sample holder and could be readily detected.

Tests run over a two-hour period at a pressure differential of 90 psi (reactor conditions) between the upstream and downstream portion of the basket showed no apparent graphite loss (by weight) at 10 C coolant temperature. At a pressure differential of 500 psi over a one-hour period, 0.9 gram was washed through the basket. The latter pressure differential, of course, represents a much more severe condition than would occur during operation of the reactor sampling system.

It is possible that, during the initial valving process in which the upstream valve is opened, the free expansion of the 520 F water at 1050 psig into the basket holder (which is initially at atmospheric conditions) produces a high velocity steam jet that breaks up the fine particles of graphite. Also, the rapid expansion of the water in the previously saturated graphite particles may, due to rapid heat addition and steam formation, contribute to the disintegration of the spectroscopic graphite.

Another possible reason for the loss of graphite could be a mistake in valving at the reactor since the wrong sequence could wash the graphite out of the basket.

2. Plutonium Ceramic Research

PuO₂-Stainless Steel Cermet Irradiations. Postirradiation examinations are being made of high and low density stainless steel-20 vol% PuO₂ cermets. The cermets were irradiated in the ETR at a calculated specific power of 3.86 kw/inch to 1.6% burnup of the fissile atoms. Only a few preliminary results of these examinations are available.

The high density (97.3% TD) cermet pellets operated with a calculated centerline temperature of 1300 C. No evidence of reaction between the PuO₂ and the stainless steel could be detected in the polished specimen. There are indications that some sintering occurred in the central portion of the pellets. In general, the high density cermet appears to be sound. A total of 6.07 cc's of gas with an activity of 50 mrad/hr was obtained from the capsule containing the high density pellets.

The low density cermet pellets operated with a partially molten core during irradiation. Calculated centerline temperature for these pellets was 1415 C. The four pellets in the capsule fused together. The structural changes which occurred in the low density cermets are analogous to those in irradiated UO_2 . A central void was formed which extended through three of the four pellets in the capsule. The region surrounding the central void contained a very dense matrix, free of voids and somewhat depleted in PuO_2 . The peripheral region of the pellet was only slightly changed from the as-fabricated condition but did contain evidence of sintering. There are no indications of gross dimensional changes. A total of 5.18 cc's of gas with an activity of 500 mrad/hr was obtained from the capsule containing low density pellets.

Measurements on cermet density and off-gas compaction are under way. A more detailed metallographic examination of the irradiated specimens is also being made--with particular attention to changes in the PuO_2 and reactions between the PuO_2 and the stainless steel matrix.

Production of High Purity $\beta\text{-Pu}_2\text{O}_3$. Six hundred grams of $\beta\text{-Pu}_2\text{O}_3$ were prepared by carbon reduction of PuO_2 . Sintered pellets of the reduced material were arc melted in a purified argon atmosphere to remove volatile impurities and to improve the density of the material. The density of a representative sample, after melting, was 11.36 gm/cm^3 (99.2% TD). The arc melted buttons contained blue and greyish-yellow tarnish films which were determined to be PuO_2 with a lattice parameter of $4.923 \pm 0.007 \text{ \AA}$ from x-ray diffraction patterns of the button surfaces. Several buttons with metallic black surface films contained some $\alpha\text{-Pu}_2\text{O}_3$, but no PuO was found on the surface of these samples.

The prepared $\beta\text{-Pu}_2\text{O}_3$ is being analyzed for gas content, semi-quantitative impurity content, oxygen-plutonium ratio and percent plutonium content.

Plutonium Nitride. Synthesis of a stock of PuN for use in fabrication, property measurements, and irradiation experiments is continuing. Several procedures have been tested and an optimum procedure was chosen for use with presently available equipment. X-ray diffraction indicates oxygen contents running generally less than 0.4%. Lattice parameters show good consistency between batches. An apparatus to provide analysis for nitrogen by a modified Kjeldahl method is being developed.

PuN -SS Irradiation. PuN -20 wt% SS (type 316) pellets were pressed to 70% TD in preparation for a PuN -SS irradiation compatibility study. These pellets were left overnight in an oxygen contaminated nitrogen hood atmosphere during which time the PuN -SS pellets oxidized yielding

approximately an 8% growth in length and 2-4% growth in diameter. The dimensional changes were sufficiently large to preclude their placement in the stainless steel can intended as the primary container for the irradiation study. This rate of oxidation indicates that extraordinary care must be taken and some hood modification made to assure meaningful studies of plutonium nitrides or uranium-plutonium nitrides.

Plutonium Ceramics Metallography Facilities. Techniques for the metallographic preparation of PuN and nitride-containing specimens in general are being developed. The hood facility is operating on nitrogen and contains less than 1/2% free oxygen in two sections of the hood. The section nearest the air lock contains one to two percent O₂.

High Temperature Dilatometer. A plastic water line inside the vacuum enclosure ruptured causing burnout of the tantalum heating element. The system has been completely cleaned, and a heating element of a new design was installed. The unit is now back in operation with water filters installed to prevent plugging and over-pressure of the plastic tubes. The glove box facilities for the dilatometer are about 75% complete.

It has been found that the ThO₂ sample holders being used are dimensionally unstable at elevated temperatures (about 1300 C) and are therefore unsuitable for high temperature thermal expansion work. An attempt to stabilize the ThO₂ with a high temperature (1750 C) long time (65-hr) soak in a hydrogen furnace resulted in cracking of the tubes. It is felt that the cracking did not result from thermal shock.

Phase Studies. Oxygen dissociation pressures in the Am-O system have been measured at 1014 C. The oxygen pressure in the system was fixed by externally controlling the decomposition of a cupric-cuprous oxide mixture. Equilibrium over the reduced americium oxides was approached from both low and high temperatures and occurred rapidly for O/Am ratios greater than 1.95. The dissociation pressures varied linearly between O/Am 1.985 and 1.966 and extended from 14.3 to 0.96 mm O₂ at these compositions, respectively. In the vicinity of AmO_{1.90}, approximately seven hours were required to attain equilibrium with an associated oxygen pressure of 6×10^{-3} mm Hg.

Some doubt exists as to phase relations in the PuO system below PuO_{1.75}. A recent British diagram, obtained by high temperature x-ray diffraction, is in disagreement with the earlier Hanford diagram in that a eutectoid is not shown at PuO_{1.70}. This construction is inconsistent with thermal expansion and electrical resistivity

results obtained at Hanford, Harwell, and Mol. In our earlier diagram, phase fields below $\text{PuO}_{1.70}$ were not rigorously investigated due to sample oxidation. Currently, DTA and high temperature diffraction techniques using extremely dry atmospheres are being developed to examine this portion of the diagram. Samples with O/Pu ratios below 1.8 have been prepared by reacting PuO_2 and $\text{PuH}_{2.7}$, both prepared from high purity alpha plutonium metal, at 1650 C in vacuum. A sample of composition $\text{PuO}_{1.555}$ showed three phases: $\alpha + \beta \text{ Pu}_2\text{O}_3$ plus a FCC phase at room temperature. Additional compositions are presently under examination.

The Plutonium-Boron System. Nine arc-cast samples of Pu-B compositions between 50 and 96 at.% B have been annealed at 1600 C and slow cooled over a period of approximately 16 hours. X-ray diffraction analysis of these samples is in progress. Compatibility tests were conducted prior to annealing. Boron nitride displayed satisfactory compatibility with Pu-B compositions, but reactions of varying severity were observed with the following refractory metals and oxides: W, Ta, Mo, Al_2O_3 , ThO_2 .

3. Ceramic (Uranium) Fuel Research

Heating Characteristics of Pneumatic Impaction Cans. The first attempt to establish the heating and cooling characteristics of pneumatic impaction cans use in the fabrication of cermet was partially successful. The cans consisted of double walls of molybdenum separated by MgO insulators. Four W-5% Re - W-26% Re thermocouples were positioned half-way up the can: (1) between the outer molybdenum can and the MgO liner, (2) between the MgO liner and the inner molybdenum can, (3) 0.25-inch from the inner molybdenum can, and (4) in the center of the can. The inner molybdenum can was filled with 0.25-inch thick molybdenum discs.

The thermocoupled can was placed in the induction heating coil and heated while the thermocouples recorded the temperatures. An optical pyrometer was used to measure the outer surface temperature of the can.

Only two of the four thermocouples, Nos. (1) and (4), operated satisfactorily. The results showed that the temperature in the center of the can lagged the outer surface temperature by about 500 seconds (6 minutes). The temperature lag between the outer surface temperature and thermocouple No. (1) [on the inner surface of the outer molybdenum can] was only about 150 seconds (2.5 minutes).

When the outer surface temperature reached about 1400 C, a longitudinal break occurred in the can, and it was necessary to discontinue heating. Coincident with this break, thermocouple No. (1) ceased to function, probably because of oxidation. The can was left in the induction coil

for about five minutes during which the central temperature dropped only 50 C and was then transferred in 15 seconds to the cold impaction die. While in the die, the central can temperature (thermocouple No. (4)) dropped at the rate of about 2 C per second. The results of this experiment indicate that:

1. Cans should be heated for at least 15 minutes after the external surface reaches the desired temperature. A heating time of 30 minutes is recommended.
2. Total heater-die transfer time plus die residence time prior to impaction should be recorded for each shot. Central can temperatures should be corrected by the observed temperature loss-rate of 2 C/second.

Relationships which were not determined because of the break in the can, and which are sufficiently important to justify a second attempt, include:

1. Relative temperatures on either side of the MgO insulators.
2. Relative temperature loss-rates near outer can surface compared to the center of the can when the can is in the die.
3. Actual time to reach temperature equilibrium throughout the can during heating.
4. Behavior at higher temperatures (to 1800 C).

Knowledge of these factors will permit accurate estimates of the impaction temperatures and of the can temperature uniformity.

Pneumatic Impaction of Pure Tungsten. In a study of the basic mechanism of pneumatic impaction, plates of pure tungsten, both 5 μ powder and -150 mesh spheres, were prepared by pneumatic impaction at 1400 C. The plates showed some laminations and cracks but will be satisfactory for obtaining density, metallography, and bend strength specimens. Plates of the same materials will be produced by pneumatic impaction at 1600 C and 1800 C. The aim of this work is to establish the optimum temperature for bonding tungsten by pneumatic impaction. Subsequent experiments will be concerned with varying the impaction pressures and using post-impaction heat treatments.

Pneumatic Impaction of Beryllium Powder. Fine (-17 micron) beryllium powder has been obtained. It will be loaded into a pneumatic impaction can and impacted at about 1000 C. A cylinder about 2 inches in diameter by 2 inches high will be made so that both vertical and horizontal tensile specimens can be machined. It is hoped that pneumatic impaction

can be used to produce beryllium bodies having a random small grain structure which will exhibit tri-axial ductility. Most commercial methods currently being used produce preferred orientation of the grains and uni-axial or bi-axial ductility characteristics.

Pneumatic Impaction of BeO and BeO-containing Fuels. The pneumatic impaction process is currently being investigated as a means for densifying BeO, BeO-UO₂, and BeO-PuO₂. Initial effort in this area is devoted to the pneumatic impaction of BeO and BeO-UO₂, to be followed by an evaluation of the latter's fission product retention capabilities. Other property measurements, such as strength and internal friction, will be determined for both materials where appropriate and necessary to evaluate fabrication characteristics. As information is acquired, emphasis will be shifted to fabrication studies of BeO-PuO₂ fuels for reactor concepts such as the EBOR (Experimental Beryllium Oxide Reactor).

Two pounds of Japanese NCK-type, CF-grade BeO were supplied by General Atomic for pneumatic impaction studies. This material reportedly sinters to high density at low temperatures (1450 C). As a result, the grain structure remains small, so that relative strength, structural integrity, and fission product retention may be enhanced. However, the initial submicron size BeO particles agglomerate easily, and tend to sinter inhomogeneously, so that there are regions of extremely high density (almost transparent) and regions of lower density which permit the fission gases to escape. The pneumatic impaction process may alleviate the agglomeration problem, while yielding dense bodies of small grain size.

In an initial test the BeO powder (as-received) was pressed into the cavity (0.75" OD by 1.5" long) of a mild steel die, and the assembly inserted into a 2" OD stainless steel impaction container. In turn, this container was placed inside a 2.5" OD stainless steel can, the annular area containing a MgO thermal insulation sleeve. The assembly was then evacuated, heated to 1200 C, and impacted at ~300,000 psi. Post-examination of the impacted BeO pellet revealed extensive laminations throughout the pellet length. A dark gray (~0.1-inch thick) zone, circumferentially surrounding the BeO, presumably was a reaction product of the BeO and mild steel die. A surface of the pellet was polished and etched for periods up to 1.5 hours in a cold, 50% HF solution, but microstructural features were difficult to distinguish. The latter may be due to the presence of some transparency, since the BeO pellet appeared extremely dense.

The laminar cracking of the above BeO pellet is thought to be a result of differences in the thermal expansion of BeO and mild steel and/or springback of the more ductile mild steel after impaction. To alleviate this possibility, current effort is being made to impact BeO pellets in an aluminum- or magnesium-oxide matrix. These oxides have more desirable thermal expansion characteristics and are expected to act as a buffer during springback.

Sodium Bonding Facility. Equipment is being designed and fabricated for use in an experimental sodium bonding facility. This includes a sodium extruder, sodium outgas furnace, and first stage bonding furnace under construction with a final stage bonding furnace under design. A preliminary equipment layout in the existing hood designated for the sodium loading facility has been completed. Detailed drawings will be made to incorporate the necessary hood changes.

UO₂-PuO₂ Fast Fuel Development. Approximately 8 Kg of UO₂-20 wt% PuO₂ fast reactor fuel is being prepared for irradiation evaluation in EBR-II. The high density mixed oxide is being prepared by ball-milling the powders to obtain a homogeneous mixture of small particles followed by pneumatic impaction to obtain a high density feed material suitable for vibrational compaction. High temperature vacuum outgassing (900 C for 4 hours at 10⁻⁴ mm of Hg) of the ball-milled powders effectively removes adsorbed gases which cause low impaction densities with the high surface area powders. The vacuum outgassing treatment also reduces stoichiometry to avoid excess oxygen in the fuel material. Analytical results on the first 1 Kg batch of UO₂-20 wt% PuO₂ processed by this technique were O/U ratio 2.007, gas content 0.036 cc/gm, impacted density 98-99% TD.

Effects of Atmosphere on the Electrical Resistivity of UO₂. A single crystal UO₂ specimen (1/4-inch diameter x 1-1/4-inch long) which had been heated in argon containing a slight oxidizing impurity was reheated to a constant temperature of 860 C in a reducing atmosphere (argon - 8% hydrogen). The electrical resistivity was observed to increase with time from an initial 1.6 ohm-cm to a saturated value of 8.0 ohm-cm in approximately 8 hours. A plot of log resistivity versus time showed a linear relationship.

Reproducible resistivity versus temperature data can be obtained to 1600 C when the measurements are made in the reducing atmosphere.

UN-20 wt% PuN Irradiation. The third and last irradiation capsule containing UN-20 wt% PuN pellets (GEH-14-618) has arrived at the Radiometallurgy Laboratory for postirradiation examination. The calculated burnup on the fuel in this capsule is 12.8 x 10²⁰ fissions/cm³.

X-ray diffraction analysis of the affected zone in the 304L stainless steel cladding used in a previous capsule (GEH-14-616) has shown no detectable amounts of any foreign phases. Further methods of analyses are being carried out.

Electron Microscopy of Irradiated UO₂. Specimens of sintered UO₂ used for long-term low temperature irradiation studies have previously been examined by electron fractography-replica methods and changes induced

by the irradiation correlated with physical property changes. To provide further microstructural information, polished surface electron microscopy of the same specimens is in progress.

Radiation levels from the mounted specimens are now low enough to permit processing them in the Ceramics Research Laboratory, where an excellent polish was achieved. Electron microscope replicas are in preparation. Microhardness determinations are being repeated on the repolished specimens.

Metallic Impurities in Mixed Oxide Fuel. Representative samples of typical lots of mixed oxides, used as fuel for PRTR, EBWR, and Saxton rods, were analyzed for metallic impurities such as Pb, Sn, and Zn. In general, impurity levels were about the same in all cases.

Some PRTR samples were representative of fuel elements now in the reactor that have accumulated exposures greater than 3000 Mwd/tony. Impurity levels (ppm) were: Sn 20-30, Pb 1-5, and Zn less than 50, which is the reportable lower limit of the emission spectrographic method.

Materials and Information Exchange. Twenty single crystal specimens of UO₂ were prepared, characterized, and shipped to Japan for the Central Research Laboratory, Toshiba, and the Research Laboratory, Atomic Fuel Corporation, for research studies and information exchange.

Requests for UO₂ crystals were received from the Watson Research Center, IBM Corporation, and the National Magnet Laboratory at M.I.T.

4. Basic Swelling Studies

Irradiation Program. A controlled pressure-temperature capsule containing six high purity uranium, two U + Fe-Al, and two U + Fe-Si specimens has continued to operate satisfactorily in-reactor at 1000 psi and 625 C. During reactor shutdown, this capsule is controlled at 1000 psi and 400 C.

Two controlled pressure-temperature capsules which will operate in-reactor at 500 psi and 450 C and 575 C, respectively, were assembled. One was shipped to the reactor for charging, and the other is being stored for future irradiation.

Postirradiation Examination. The density measurements made on uranium specimens removed from a swelling capsule irradiated at 575 C and 1000 psi to 0.15 at.% B.U. reveal that this pressure reduced the amount of swelling in high purity uranium by factors of 5 to 15 over that typically observed in uranium irradiated at low pressure. U + Fe-Al and U + Fe-Si alloy specimens exhibited reductions by factors

of 2 to 5. The effectiveness of this 1000 psi external, hydrostatic pressure is significantly less at 575 C than was the case for 450 C. At 450 C the reduction for high purity uranium was by factors of 50 to 100.

Preliminary metallographic examination corroborates the density measurements. The as-extruded high purity uranium specimen shows extensive grain boundary porosity (grain boundary tears were present in one end of the sample) and a few large pores scattered throughout the matrix. None of the crystallographically aligned microtearing typical of low pressure irradiation at 575 C was present. The 1000 psi pressure is effective in suppressing the aligned tearing at 575 C but not the grain boundary porosity and tearing. This behavior substantiates the concept that defect migration and agglomeration processes are responsible for swelling in this temperature range.

5. Irradiation Damage to Reactor Metals

Alloy Selection

Several nickel-base alloys are being studied to determine the effects of irradiation upon their mechanical properties. Specimens of Inconel 600, Inconel 625, Inconel 702, Inconel 718, Incoloy 800, Hastelloy N, and Hastelloy X-280 have been irradiated at 50, 280, 580, and 740 C and are currently being tested in tension in the temperature range of room temperature to 650 C.

Specimens of Hastelloy N irradiated at 280 C to an exposure of 1.03×10^{20} nvt ($E > 1$ Mev) and tested at room temperature and 650 C show a marked effect of test temperature. For the case of the room temperature tests, a 70% increase (to 74,750 psi) in yield strength and a corresponding 27% decrease (to 49.5%) in uniform elongation was found. The corresponding values at the 650 C test temperature, 44,600 psi and 6.1%, respectively, show that a marked decrease in yield strength and uniform elongation occurs at the higher test temperature. Similar results have been found for other solution hardenable nickel-base alloys.

Specimens of Inconel 625 have been irradiated at 280 C and tested at 650 C. These specimens were irradiated at exposures to 1.46×10^{21} nvt ($E > 1$ Mev) in the solution treated and aged condition. Results of the tests indicate little change in strength at exposures to 5.68×10^{20} nvt. At an exposure of 1.46×10^{21} , the data indicate a reduction in both strength and uniform elongation as compared to the lower exposure data.

Specimens of Inconel 718 irradiated at exposures of 1.03×10^{20} , 1.57×10^{20} , and 3.2×10^{20} nvt ($E > 1$ Mev) at 280 C were tested at 650 C. Data from these tests show an increase in yield strength from 87,440 psi at the lowest exposure to 138,610 psi at the highest exposure. The corresponding percent uniform elongation values were 5.21% and 0.85%, indicating a decrease in ductility. Similar effects were found for specimens irradiated at 50 C to exposures of 1.8×10^{20} and 1.0×10^{21} nvt and tested at 650 C.

To determine the effects of irradiation in a liquid metal environment, a temperature controlled capsule has been designed for insertion in the ETR. A bench model of this capsule has been fabricated and a safety circuit and temperature control system installed. Final assembly of the equipment is now being completed with actual operation of the capsule scheduled late this year.

In-Reactor Measurements of Mechanical Properties

Three in-reactor creep tests are currently in progress. Tests on annealed 304 stainless steel are being conducted at 750 C, 10,000 psi stress and 525 C, 30,000 psi stress. The 650 C test has accumulated 1200 hours of test time and exhibits a rate of 4.4×10^{-6} /hr, though this rate has not yet been established as a minimum. The 525 C in-reactor creep test and its control are in the primary stage of creep and minimum creep rates are yet to be determined. The third test, sponsored jointly by the AEC and the AECL, is being conducted on heat treated Zr-2.5 wt% Nb. Heat treatment consisted of heating specimens 15 minutes at 870 C, water quenching, and aging at 500 C for 24 hours, followed by air cooling. The in-reactor test was just recently started and a second stage rate cannot be established.

Another in-reactor creep capsule was recovered and sent to Radio-metallurgy for specimen removal. The creep test was terminated after specimen failure at test conditions of 650 C, 20,000 psi. This will be the first of the in-reactor creep specimens recovered at which test conditions were unchanged for the test duration.

Irradiation Effects in Structural Materials

Continuing efforts have been made to measure the crack velocity in a specimen undergoing catastrophic failure. These have been preliminary efforts with the major goal being the determination of the ability of a material to arrest a moving crack. To date the utilization of the Doppler effect to ascertain the crack velocity has proven somewhat unsuccessful due to the insensitivity of high-speed recording equipment. However, a new approach based on eddy current techniques is being studied, and equipment is presently being assembled for a trial run.

Irradiation Effects in Nickel-Base Alloys

The purpose of this program is to determine the effects of modified microstructures on the reaction to subsequent irradiation of nickel-base alloys. These structural modifications are made by pre-irradiation thermal and/or mechanical treatments and are screened by tensile and stress-rupture tests and metallographic examination on both control and irradiated material.

Twenty-four Hastelloy X-280 specimens irradiated at 280 C to a goal exposure of approximately 2×10^{20} nvt were discharged at the conclusion of ETR Cycle 70. These specimens, representing a variety of preirradiation treatments, will be returned to BNW for tensile and creep-rupture testing. Three quadrants (18 specimens) containing Hastelloy X-280 and Inconel X-750 will be irradiated during ETR Cycle 71. These specimens have been heat treated to give a range of microstructures. Metallographic samples of Inconel X-750, representing five different heat treatments, have revealed that the desired range of precipitate aging (from underage to average) was obtained.

Components for three creep-rupture units are being fabricated from nickel alloy material for testing in air to 925 C. Studies are being made to incorporate a completely automatic, hydraulic loading system as a further refinement of the prototype creep-rupture unit.

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.

Capsules containing Ferrovac E and Battelle zone-refined iron have been sent to the reactor for irradiation to goal exposures of 5×10^{19} nvt and 1×10^{20} nvt. Iron specimens irradiated to 5×10^{18} nvt and annealed at 350 C for three hours have been sectioned, thinned, and examined in a JEM microscope. Distortion of the image by the magnetic field of the specimens is less severe than in the Philips microscope and should give better resolution. No spots were observed, however, in agreement with observations performed on the Philips.

A specimen holder for measuring resistivity ratios of single crystals has been designed and built. Troubleshooting of the high sensitivity resistivity apparatus is in progress and should be completed in the near future. A specimen holder for wire hairpin samples has also been designed and built and will be used to measure cold work recovery detectable by resistivity changes.

ATR Gas Loop Studies

Foil Insulation. The thermal conductivity tests on the foil insulation were completed, and the data are being analyzed. Preliminary checks showed that when a low temperature test point was run after the loop had been at high temperature the apparent conductivity of the foil was much higher than would be expected. Several test points were run which verified this indication. Gaps apparently formed at the joints in the insulation upon cooling down from a higher temperature, unless the test section was vibrated. The vibration would allow the insulation to settle into place and reduced the leakage of gas at the joint. A summary of results will appear in DCK-I-79.

Pipe Insulation. A topical report on thermal properties of Hastelloy-X felted metal insulation was prepared and is now ready for publication.

Model Loop Operation. During the month a bearing failure occurred in the eddy-current-variable speed drive in the frequency changer assembly. The bearing apparently had been run with some of the cooling water in the housing. The water washed the lubricant out of the bearing and caused the failure. One man-week was required to repair the unit.

Bypass Purification System. The bypass purification system on the model gas loop was improved by enlarging the low temperature charcoal and molecular sieve adsorber beds. This is the first step in making the system a complete cleanup system. A room temperature molecular sieve bed will be added when the parts arrive.

Helium Purification System. The high pressure helium purification system was placed in service following completion of repairs to the compressor. Helium was recovered from the 2500 cubic foot plastic balloon with the following results:

ppm	H ₂	O ₂	N ₂	CH ₄	CO	H ₂ O
Purity from Balloon	*	640	3000	33	*	8
Purity after 1st cleanup	3.5	1.3	1.9	*	*	7.5
Purity after 2nd cleanup	*	*	*	*	*	6.0

*No detectable amount from chromatograph, sensitivity is about 0.1 ppm.

Heater Design and Piping Modifications. All major components and material for fabrication of the concentric two-stage helium gas heater for the model loop have been ordered. Bids on the revised ceramic items have been returned with a saving of 80% of the original bids on these items.

A computer program was written to determine optimum insulation thickness, piping sizes and cooling water flow for the piping from the regenerative heat exchanger to the new heater and from the heater to the test section. The program also investigated the effects of cooling water flowing concurrent or counter current with the gas flow.

Dissimilar Metal Welds. The construction of the ATR gas loop will require the welding of thick wall sections of Hastelloy X-280 to Haynes 25 and AISI 316 stainless steel. The results were previously reported (1) on the welding procedure and properties of these weld combinations utilizing Hastelloy-W filler metal. Hastelloy-W has a long successful history in joining dissimilar high temperature alloys, but the results of short time elevated temperature tensile tests of wrought Hastelloy-W indicate a sharp loss in ductility near 1200 F (650 C). The results of tensile tests on wrought Hastelloy-X indicate a similar but less severe loss in ductility at 1200 F (650 C), and Hastelloy-X filler has also been successfully used in joining dissimilar metals. The effect of aging on the ductility of Hastelloy-X and Hastelloy-W weld metals aged at temperatures in the low ductility temperature range of 1200 F to 1600 F (650 to 820 C) has not been previously investigated. The aging is being done in inert gas at 1000 F (540 C) and 1200 F (650 C) for the AISI 316 stainless steel joints and at 1400 F (760 C) and 1600 F (850 C) for the Haynes 25 joints.

The relative ductilities of the welded joints are being determined from bend tests of 0.625-inch thick longitudinal weld specimens. The ductility is measured as the maximum elongation in the outer fibers of the material where cracking or failure was initiated. The bend testing of unaged, 500-hr aged, and 1000-hr aged specimens is complete. The tensile properties of aged Haynes 25 and Hastelloy X-280 base metals, and Hastelloy-X and Hastelloy-W weld metals are also being determined. The tensile testing of the 500-hr aged material is complete and specimens of 1000-hr aged material are being machined. Weld specimens for aging periods greater than 1000 hours are currently being aged. All of the unaged, 500-hr aged, and 1000-hr aged bend specimens initially failed in the weld metal.

(1) R. L. Knecht, Dissimilar Metal Welds of Certain Superalloys and Stainless Steels, HW-78575, August 1963.

As expected with aging, a loss in ductility and increase in yield strength was observed. The aging is considerably more sluggish in Hastelloy-W than in Hastelloy-X. Although both weld metals with a 500-hr age reach similar minimum room temperature ductilities, the tensile tests at the 1000 F (540 C) and 1200 F (650 C) aging temperatures indicate that Hastelloy-W has a more useful amount of ductility remaining than Hastelloy-X.

The effect of a 500-hr age at 1000 F (540 C) on the yield stress and ductility of Hastelloy X-280 is quite small and in good agreement with the results of a 100-hr age at 1000 F (540 C) reported by Levy and Wheeler.⁽¹⁾ The effects of a 500-hr age at temperatures greater than 1000 F (540 C) on yield stress and ductility of Hastelloy X-280 are relatively small when tested at the aging temperature and quite pronounced when tested at room temperature.

The above results of the tests on the weld metals and base metals aged 500 hours suggest that Hastelloy-W has better aging characteristics than Hastelloy-X when used as filler wire for joining Hastelloy X-280 to AISI 316 stainless steel and Haynes 25. This suggestion is based on considerations of the weld metal yield stress being larger than the base metals yield stresses, the relative sluggishness of weld metal aging, and the amount of useful ductility remaining in the weld metal and the base metals after an aging treatment. The tensile testing of the weld and base metals aged 1000 hours and greater than 1000 hours is continuing.

(1) Quarterly Progress Report - Metallurgy Research Operation - January, February, March, 1964, edited by J. J. Cadwell, HW-86269, April 15, 1964.

6. Nuclear Graphite Studies

High Temperature Graphite Irradiations. The second capsule, GEH-13-11, in the series of high temperature irradiations of graphite has completed two cycles of irradiation. Just prior to the current shutdown, two of the 12 thermocouples failed. Analysis of the cause of failure is presently under way.

EGCR Graphite Irradiation. The 10th capsule, H-3-10, in the series of long-term irradiations of EGCR graphite successfully completed the first cycle of irradiation in the GETR. All thermocouples functioned properly and sample temperatures agreed with those of previous capsules.

Effects of Additives on Graphite Contraction. Samples from the Contract DDR-136 were discharged after an initial exposure of 1625 Mwd/At_K at a temperature of 650 ± 50 C.

The samples were prepared in three sets: (1) Needle coke, pitch binder with the following additives: 5% Al, 5% Al₂O₃, 5% Al₄C₃, 4.5% Fe₂O₃-0.5% SiO₂, 4.5% Fe₂O₃-0.5% Al₂O₃, 4.5% Fe₂O₃-0.5% Co₂O₃. (2) Fluid coke, pitch binder with the following additives: 5% Ni₂O₃, 5% Fe₂O₃. This set also includes two purification schedules in which the purifying gas was introduced at either 2300 or 2900 C. (3) Needle coke, phenolformaldehyde resin binder with the following additives: 5% Fe₂O₃ and 5% Ni₂O₃.

In Set (1) at this low initial exposure, the sample with the Al additive contracted the greatest; other samples were very closely grouped. In Set (2) the Ni₂O₃ sample contracted most and the Fe₂O₃ sample least when compared to the standard without additive for this set. The temperature of purification did not appear to make any difference. In Set (3) the Ni₂O₃ sample contracted slightly more than the Fe₂O₃ and standard samples.

The coefficient of thermal expansion did not show any marked changes, within each set, either in the unirradiated or first-exposure data.

Argon-Sensitized Radiolysis of Carbon Monoxide - Water-Vapor Mixtures. The helium sensitized radiolysis of carbon monoxide and water-vapor mixtures was reported previously. The data showed that all of the energy absorbed in the helium was effective in causing the carbon monoxide - water-vapor reaction even when the gas contained more than 99% helium.

Since argon has a lower ionization potential than helium, it is of interest to investigate the efficiency of energy transfer from the argon to carbon monoxide and water vapor. We have therefore studied

the radiolysis of carbon monoxide - water vapor mixtures in varying quantities of argon. The results are summarized below.

Sample No.	% A	% CO	% H ₂ O	G(H ₂) Molecules/100 ev	G(CO ₂) Molecules/100 ev	Total Pressure, torr
35-R-6	24.8	38.2	37.0	3.14	5.72	704
36-R-6	55.8	21.7	22.4	3.20	4.42	704
37-R-6	75.4	12.6	12.0	3.63	5.98	706
38-R-6	85.7	6.97	7.36	3.49	4.51	717
39-R-6	90.2	4.95	4.88	3.86	5.98	706
40-R-6	93.9	2.94	3.17	3.91	5.01	582

The data indicate that energy absorbed in the argon is also effective in causing reaction in all gaseous mixtures studied. Further work is currently in progress to extend the data to mixtures containing as much as 99.5% argon.

Oxidation of Boronated Graphite: As reported earlier, oxidation of a graphite sample containing a nominal 7 wt% boron (grey material, resulted in a rapidly decreasing oxidation rate, presumably as a result of the formation of B₂O₃, which serves as a physical barrier to further oxidation. To determine the effect on the oxidation rate of removing this B₂O₃, the same sample was placed in a container of distilled water, a vacuum applied to the container to remove air from the pores of the sample, and then the vacuum released to force water into the partially evacuated pores. Next, the water was boiled to dissolve as much B₂O₃ as possible, the sample removed from the water and out-gassed to determine the weight of B₂O₃ lost, and the oxidation rate was determined. After the rate measurements, the entire procedure was repeated. In all cases oxidation was carried out at 730 C with helium containing 1% O₂ at a flow rate of 150 cm³/min. The results of these treatments are presented in the following table:

Run	Wt. of B ₂ O ₃ Removed, g	% of Initial B Removed	% Oxidn. of Sample	Initial Oxidn. rate, hr ⁻¹	Final Oxidn. rate, hr ⁻¹	$-\Delta n_c$ $\Delta n_{B_2O_3}$
1	0	0	--	--	-2.53×10^{-4}	12.6
2	0.0133	10.1	6.07	-1.51×10^{-2}	-3.32×10^{-3}	7.18
3	0.0151	21.4	10.1	-2.54×10^{-2}	-6.54×10^{-3}	3.66
4	0.0156	33.2	15.1	-2.30×10^{-2}	-7.36×10^{-3}	6.87
5	0.0286	55.0	22.4	--	--	--

The values in columns 2 through 5 refer to the start of the run. A comparison of the initial oxidation rates of column 5 to the final oxidation rate of the preceding runs, column 6, demonstrates that B₂O₃ removal effects an appreciable rate increase. In contrast, when the initial and final rates for a given run are compared, it is seen that a significant decrease in rate occurs with the formation of B₂O₃. The ratio of the moles of carbon lost to the moles of B₂O₃ formed, column 7, gives an indication of the relative rates of graphite and boron oxidation. The general decrease in this ratio may indicate an increasing accessibility of the boron to oxidation as a consequence of graphite removal, or a more effective inhibition of graphite oxidation by the remaining boron. Unfortunately, the fragile sample was accidentally broken after the leaching treatment of Run 5.

Pore-size-distribution measurements have been performed on all four types of boronated graphite but without the success ordinarily found with other graphites. For an unknown reason, the specific surface areas obtained by summing the pore areas are larger than the specific areas found by the BET method, the discrepancy being greatest for the samples containing a nominal 7 wt% boron. This may be seen in the following table.

<u>Sample</u>	<u>Boron Content</u>	BET Surface Area,	BJH Surface Area,
		<u>m²/g</u>	<u>m²/g</u>
D-80-T	5% (black)	0.46	0.48
E-78-T	5% (grey)	0.15	0.14
H-79-T	7% (black)	0.30	0.34
J-70-T	7% (grey)	0.20	0.25, 0.24

Qualitatively the pore-size-distributions are similar to those found for other graphites, with a peak in the 20-25 Å pore-radius range and usually a secondary peak or peaks at larger radii. The significant difference in the surface areas of the black and grey materials results from the greater number or pores of all sizes in the micro-pore range for the black boronated graphite.

7. Metallic Fuel Development

Irradiation of Thorium-Uranium Fuel Elements. The irradiation of six tubular Zircaloy-2 clad thorium-2.5 wt% uranium-1.0 wt% zirconium fuel elements continued successfully in the ETR-P7 and M-3 loops. Current irradiation conditions are:

Loop	Core Temp., °C	Surface Heat Flux BTU/hr-ft ²	Integrated Exp. Fissions/cm ³ (Mwd/ton)	Fuel Volume Increase %ΔV/V
P-7	425	5.1×10^5	3.9×10^{20} (11,100)	1.0
M-3	520	7.1×10^5	2.1×10^{19} (880)	-

Irradiation of Metallic Uranium with Submicron Uranium Carbide Dispersions. The irradiation performance of uranium containing a submicron dispersion of uranium carbide is being evaluated.

Density measurements have been completed on four fuel rods, three with 0.5μ diameter UC particle dispersions, and one with UC particles in the size range 2-5μ, which were irradiated to 0.3% burnup at a maximum fuel temperature of 500 C. The density decrease of the fine UC containing fuel rods, all of which suffered cladding failures, ranged from 8.8% to 13.4% whereas the decrease for the fuel rod with larger UC particles was only 3.4%.

Metallographic examination of the uranium in the rod with the larger UC particles shows considerable porosity of about 1μ size with some small amount in the 10μ range. Most of this porosity is spherical. In the fuel with the submicron UC dispersion the uranium exhibits extensive grain boundary tearing porosity. The difference in the as-irradiated densities is readily confirmed by the large difference in size and amount of porosity observed.

Postirradiation annealing of declad samples at 625 C for 10 hours has produced approximately twice the density decrease in the uranium with the 2-5μ particles as in the uranium with the submicron UC. The density changes in these rods follow the same trend as was observed in similar samples examined earlier.

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

Work was started on installing the pressure tube and preheater in the heat transfer laboratory for running the fog-cooled experiments. This work requires the removal of the existing preheater, installation of a new preheater to generate 100% steam, installation of the AECL flow nozzle, and installation of the connecting piping. Simultaneously, assembly of the 19 rods for the electrically heated test section was continued. This test section consists of 6.5-ft long rods spaced 0.040 inch apart in a 3.25-inch pressure tube. The test section is designed for a maximum heat flux of 680,000 Btu/hr-ft².

The tests have been designed to cover both fog (wet steam) and sub-cooled water inlet conditions. For fog-cooled conditions the tests will compare the differences in boiling burnout heat fluxes when using the AECL nozzle. This nozzle can be adjusted to provide a homogeneous fog, or a separated fog flow, at the test section inlet. The separated fog flow will have the water concentrated at the interior of the bundle. The tests will also determine the effects on boiling burnout of a smooth versus a roughened pressure tube wall. The rough wall is created by 0.050-inch high circumferential ridges every 3.25-inch along the length of the tube.

9. Advanced Reactor Concept Studies

Nitride Fuel Cycle Studies. Document BNCC-43, "Economic Considerations for a Nitride Fuel Cycle Study," was issued during the month. This document reviews and evaluates the fuel cycle costs reported in a Combustion Engineering, Inc., study of a carbide-fueled 1000 Mwe fast breeder reactor⁽¹⁾, with comparisons to the expected costs of a nitride fuel cycle. These comparisons, included in the previous monthly report, indicated that the costs of nitride fuel conversion and fabrication may be critical to the cycle, and that any development of such a cycle should include efforts toward development of an economical process for conversion of fuel materials to the nitrides.

Cross section data for the 18-group library to be used in the nitride fueled fast reactor study have been transferred to cards in HFN input format. This was accomplished using a new version of HRG developed by Theoretical Physics which now provides for punching microscopic cross sections in addition to the macroscopic punch option, formerly available. A detailed examination of the library data is being carried out to be certain that best available values are utilized before generating the HFN library tape.

Gas generation and transport in mononitride fuels has been briefly reviewed. Although there appears to be only limited experimental data on fission gas release from nitrides currently available, similar data for carbides are more abundant. Both uranium monocarbide and mononitride have similar crystal structure (FCC) and lattice dimensions that are nearly the same. On this basis one might expect the diffusion properties of noble gases in nitrides to be somewhat similar. Interestingly, diffusion measurements of xenon in UC and UN yield essentially the same activation energy of about 60 k Cal/mole. Calculation of gas generation rates in the fast neutron spectrum from all sources in the fuel was initiated.

(1) Liquid Metal Fast Breeder Reactor Design Study, Combustion Engineering, Inc., CEND-200, Vols I and II.

10. Critical Flow of Steam-Water Mixtures

The high pressure critical flow apparatus was readied for critical flow experiments using a "saturated" blowdown of heated water. This method differs from previous tests which used high pressure air to maintain the pressure in the water storage tank at a constant pressure.

The first experiment used the test section employed earlier and reported in HW-80535. This test section was in the form of a pipe 0.500-inch ID and 10 inches long. Pressure taps were provided at close intervals along the length of the pipe to enable the determination of axial pressure gradients during the blowdown of the 1000-gal vessel. The purpose of this experiment was to provide data to allow a comparison of the results of transient experimentation with those of the steady state method used earlier. The water in the vessel was heated to 535 F (280 C) (saturation pressure = 920 psia) and the experiment initiated by the impulse opening of the main discharge valve, thus allowing the heated water to pass through the test section to the atmosphere. During the discharge of the water in the vessel (which took approximately 12 minutes), the pressures at the various positions in the test section and in the tank were measured by transducers and recorded on an oscillograph. Flow rates were measured both by drop of water level in the vessel and by an orifice flowmeter. The test was successful, and the resulting data were prepared for further processing in the computer center.

11. Upstream Boiling Burnout

Additional experiments were run to investigate conditions at which boiling burnout takes place at locations upstream of the outlet end of uniformly heated test sections. The test section used consisted of a 12-ft long electrically heated tube with flow through a 0.435-inch hole. The test section was constructed so that it was possible to heat the entire 12-ft length, the upstream 6-ft alone, or only the downstream 6-ft length.

Extensive tests were made at 1500 psig with a flow rate of 7×10^6 lb/hr-ft². The following are the most significant observations of these tests:

- a. Burnout was observed at locations ranging from the downstream end to 14 inches from the upstream end of the heated section. In 15 out of the 24 tests, burnout occurred at a location away from the downstream end.
- b. Although the inlet enthalpy varied from 400 to 700 Btu/lb and the outlet enthalpy from 630 to 850 Btu/lb, the

enthalpy at the point of burnout fell in a much narrower band of 630 to 730 Btu/lb (3-12% quality). As the inlet (or outlet) enthalpy was raised, the location of burnout moved further upstream, thus producing only a small increase in burnout enthalpy.

- c. With all three modes of heating, the burnout heat flux decreased with increasing enthalpy at the lower enthalpies where downstream burnout occurred. At an outlet enthalpy of about 675 Btu/lb (11% quality) the burnout started to shift to upstream locations and the burnout heat flux reached a minimum of about 1.05×10^6 Btu/hr-ft². With 12 feet heated, the burnout heat flux remained at this value on further increases of enthalpy. However, with the downstream 6 feet heated, the burnout heat flux rose again with increasing enthalpy to 1.6×10^6 Btu/hr-ft² for 730 Btu/lb at the upstream burnout location.
- d. In those runs where downstream burnout was obtained (i.e., those at lower inlet enthalpies), the burnout heat flux was 10-30% higher (at the same outlet enthalpy) with the upstream 6 feet heated than with the entire 12 feet heated. With the downstream 6 feet heated, the results were intermediate.
- e. In two identical runs with 12 feet heated, burnout occurred upstream of the center. In two other runs with the upstream 6 feet heated and the same inlet enthalpy (600 Btu/lb), burnout was observed at the same location and at the same heat flux.
- f. Three runs were repeated to check reproducibility. Agreement was excellent with respect to both burnout heat flux and position.
- g. The results, with 12 feet heated, agreed well with previous data obtained with a 12-ft test section. The present test section differs from the old only in having axial inlet and outlet flow and having 18-inch unheated lengths at the inlet and outlet rather than radial inlet and outlet flow at the ends of the heated section. (All these burnouts occurred at least 4 feet from the start of the heated section.)
- h. Measurements of pressure fluctuations in the test section were made with a high speed recorder to determine whether such fluctuations were related to upstream burnout, as seems to have been the case in low pressure tests. The magnitude of

the pressure fluctuations was comparatively small, on the order of 10 psi, and preliminary examination of the data revealed no relation to burnout.

Although any firm conclusions must await more complete analysis of the test data and the completion of additional tests, two conclusions seem to be indicated by the results of these and previous tests of upstream burnout at high pressure:

- a. The graph of burnout heat flux versus local enthalpy (for the diameter, pressure, and flow used in these tests) passes through a local minimum.
- b. The burnout heat flux is higher at locations near the entrance to the test section or the start of the heated length than when the same local enthalpy exists further downstream.

12. Plutonium and U-233 Fueling of a Fast Compact Reactor

Work on this program has been completed except for completion of the final reports. The final report of the study is approximately 60% complete in rough draft form. Rough draft documentation of the core thermal hydraulic code, SWISSCHEESE, was completed. A document describing the gas transport property computer program, MIX, is being prepared for issuance in final form.

13. Fast Flux Test Facility Design Studies

Important progress items during the month were (a) review and acceptance of the FY-1965 study plan by AEC Headquarters, (b) extensive travel by FFTF personnel to obtain current data on sodium technology applications in fast and thermal reactor service, (c) initiation of use of six off-site firms and consultants, and (d) need for early initiation of irradiation experiments to prove out the suitability of PuO_2 cermet as driver fuel for the FFTF.

A draft of the Progress Report, Fast Flux Test Facility Studies (April 1964-April 1965), was prepared and is currently under management review.

A \$20,000 Procurement Directive (RL00-65-1) was placed with Atomics International for assistance in conceptual design of the FFTF materials handling system.

A \$20,000 Procurement Directive (RL00-6502) was placed with Battelle Memorial Institute, Columbus, Ohio, for assistance in conceptual design studies.

Consultant Agreement BCA-48 was placed with Gardner L. Locke, with a maximum limit of \$5,000.

At month-end procurement activities were in progress to obtain \$15,000 of service from Argonne National Laboratory and \$30,000 of service from Atomic Power Development Associates. Arrangements are being made to obtain, under an existing USAEC-COO contract, about seven man-weeks (\$7,000) of service from Mr. J. DeFelice, President of Nuclear Technology Corporation.

With the cooperation of the financial representative, a special cost accounting and weekly cost reporting system was set up. Costs are coupled directly to the individual principal work assignments in the program. This is the initiation of a procedure which will be expanded to a full PERT-cost system.

Preliminary estimates of the PACE-funded design costs for FFTF were made as follows:

\$ 600,000 - FY-1966	Advance Engineering
\$1,000,000 - FY-1967	\$500,000 Nuclear Engineering \$500,000 Title I Design
\$2,200,000 - FY-1968	\$500,000 Nuclear Engineering \$1,700,000 Title II Design.

Revised procedures for construction budget reviews were issued by the AEC January 5, 1965, but did not arrive at BNW until early April. This was too late to incorporate the effect of the new procedures into the current \$75,000,000 FFTF cost estimate.

Review of the procedures indicated that the FFTF estimates for contingency (16.7%) and escalation (8.5%) may be low. The revised procedures would indicate 18% contingency and 17-1/2% escalation and would result in increasing the FFTF cost estimate from \$75,000,000 to \$82,500,000.

The FFTF design estimate (26.3%) is consistent with the range of 12-32% costs shown in the AEC procedures and is slightly (but justifiably) higher than the recommended 20% for complex projects.

Existing data from research performed for Fermi on irradiation effects in boronated graphite have been compiled in memorandum form for FFTF use. These data include mechanical, thermal, and physical property changes resulting from irradiation in Hanford facilities at 370 and 480 C to exposures of 5×10^{21} n/cm²-sec (thermal).

Several studies are in progress on various aspects of the reactor cooling systems. Included are studies for establishing (1) the number

of parallel loops, (2) the method of dumping reactor heat, and (3) a method for process control of the primary, secondary, and tertiary cooling systems. In addition, a machine program is being prepared for analyses of hydraulic and thermal transients caused by simple loop perturbations, loss of pump power, and scrams. This program is needed before the method of process control can be firmed up and before a satisfactory emergency cooling concept can be developed.

A review of the relative merits of Na or NaK for the secondary coolant system has been completed. While NaK is liquid at room temperature, there are a number of more important factors which appear to be in favor of sodium, and a decision has been made to use Na as the secondary coolant.

An exhaustive study is being made of candidate driver fuel materials, in particular the cermet type, as requested by the USAEC at the FFTF meeting in Germantown, Maryland, in April 1965. A large amount of unclassified work has been conducted in the USA on cermet fuel development, with particular emphasis on the UO_2 -stainless steel type. Experimental work on UO_2 - and $(\text{U},\text{Pu})\text{O}_2$ -stainless steel cermet pins and plates was carried on in the U.K. up until about two years ago when the work was discontinued because of poor performance of the high ceramic loading (to 50 vol%) type fuel and a breeding penalty.

A metal fuel of interest to the FFTF is Nb-20 wt% U alloy. Fuel performance can be termed highly promising on the basis that most of the density decreases observed (2.5-3.0% of total atom burnup) are required to accommodate the fission products. Bare, unrestrained pins irradiated in NaK, with maximum fuel temperatures to 1040 C, and to total atom burnups to 0.9% or U atom burnups to 9.7% showed no evidence of poor performance or interaction with the coolant.

14. Nuclear Rocket Fuels Studies

Studies conducted for AEC-SNPO on tungsten- UO_2 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.

Molybdenum foils irradiated to 10^{20} nvt have been annealed at 575 C and examined by transmission electron microscopy. The only significant difference between this microstructure and those at lower exposures was the "smoothing out" of the dislocation lines and the appearance of more circular type loops. At these temperatures, only those processes requiring relatively short range dislocation motion (e.g., climb) are occurring.

Investigations of the nature of the loops formed in molybdenum irradiated to 10^{20} nvt and annealed at 750 C or higher are being continued. On relatively few measurements has it been possible to obtain ideal diffracting conditions. Nevertheless, some vacancy type loops have been identified. Interstitial loops are present in much greater number. At the present time, it is not clear as to why there should be so few vacancy loops. Apparently, most of the free vacancies and small vacancy clusters have been annihilated at an early stage and only the largest vacancy clusters are stable enough to form loops.

The role of dislocations as sinks for the point defects created during irradiation is currently being studied by irradiating pre-strained samples. Transmission electron microscopy shows no depletion of the spot defects around dislocations in the irradiated samples which had been pre-strained up to 20%. An actual count of the number of spot defects revealed no difference between pre-strained and annealed samples. In this case, it is most likely that the migration distance of the point defects at the irradiation temperature is so short that any denuded zone about a dislocation would be of the same magnitude as the distance between the spot defects. These pre-strained and irradiated samples will be subsequently deformed in order to study the effect of irradiation on dislocation sources. In particular, it will be noted whether the dislocations formed during the pre-strain serve as sources after irradiation or whether new sources such as grain boundaries are activated.

Rhenium samples irradiated to 7×10^{18} nvt have been further annealed for two hours at 800 C and at 1000 C. At 800 C, there is a slight coarsening of the spots and a decrease in the spot density from 4×10^{16} to $1.5 \times 10^{16}/\text{cc}$. Upon annealing at 1000 C, there is no further coarsening but a large decrease in the spot density, from 4×10^{16} to $1 \times 10^{15}/\text{cc}$. This behavior is very similar to the annealing behavior of the irradiated JM molybdenum in which there was some coarsening and then disappearance of the spots but no loop growth. Rhenium samples are currently being irradiated to doses of 1.5 and 3×10^{19} nvt. It is hoped that larger spots and loops will result and that the character of the defects can be determined.

Low temperature annealing experiments on irradiated (1×10^{19} nvt, $E > 1$ Mev) molybdenum foils have shown that there is no measurable

decrease in lattice parameter in high purity foils annealed at 200 C. There is a small but measurable decrease, 0.003 to 0.005%, for foils containing 100-200 ppm and 400-500 ppm carbon.

Annealed single crystals of molybdenum have been prepared with pairs of faces cut parallel to crystal planes having the same spacing but different crystal directions. Pairs chosen are (330), (411); (550), (710); (550), (345); and (600), (442). Eight crystals in all have been prepared. These crystals will be used in x-ray diffraction experiments to determine the existence of crystallographically-oriented defect configurations. After complete pre-irradiation characterization, they will be submitted for irradiation.

A capsule containing single-crystal length-change and x-ray specimens has been opened after reaching a goal exposure of 1×10^{20} nvt ($E > 1$ Mev). Examination of the specimens is not complete, but preliminary measurements show the magnitude of the length increases to be greater the lower the carbon content of the crystal.

X-ray diffraction determinations of lattice parameters are in progress.

The temperature-dependence of the upper and lower yield stresses in irradiated and unirradiated molybdenum has been found to contradict the dependence predicted by existing theories of yielding based on the concept of the breaking away of a dislocation from a solute atom atmosphere. This lends further support to the conclusion that yielding in molybdenum is controlled by the rapid multiplication of dislocations. Calculations of the degree of multiplication during yielding have shown that a two-fold increase in the number of mobile dislocations occurs between the upper and lower yield points in unirradiated molybdenum. The extent of multiplication in irradiated molybdenum is much greater, leading in some cases to as high as 20-fold increases in mobile dislocation density.

2. Plutonium Physical Metallurgy

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

As has been reported in the past, the amount of evidence for the existence of definite crystallographic relationships between the alpha and beta phases continues to increase. An effort is currently under way to establish, if possible, the details of this relationship.

The study of the texture established by the transformation from beta to alpha under the influence of a compressive stress offers the most promising approach to this problem. Through the use of the pole figure goniometer it is possible to find regions which behave from a diffraction standpoint essentially as a single crystal.

A technique has been worked out which permits the goniometer mounted sample to be increased sufficiently high in temperature to effect the alpha to beta transformation. With proper attention given to the glovebox atmosphere the surface oxidation can be minimized to an inconsequential level for periods of time sufficient to thoroughly investigate the beta diffraction pattern as well as that of the alpha resulting from a subsequent transformation. These pseudo-single crystal regions can thus be located in the alpha phase and the behavior of a given reflection followed through the transformation and the resulting beta structure related to the orientation of the parent alpha. The effects of interphase cycling on the orientation of both phases will also be given attention.

Samples have been prepared and textured which will be subjected to tensile loading in the hope of producing slip lines in crystal surfaces, the orientation of which is known.

An explanation for the formation of disks of columnar alpha grains has been formulated. The disks appear as bands on polished and etched surfaces and they are particularly evident in plutonium that has been reheated into the beta phase and transformed at high alpha temperatures. They are responsible for the macroscopic veining frequently observed on this surface of plutonium.

The formation of the large disk-shaped volumes with their columnar grains is reminiscent of the internally twinned martensitic plates that form in iron alloys; however, they differ in that they are much larger and that their component alpha grains appear to be columnar crystals and not simply twins.

In order for a disk-shaped transformed region to grow in thickness while maintaining continuity with the parent phase across the approximately planar interphases, the volume change must be accommodated almost entirely by expansion or contraction perpendicular to the plane of the disk. Thus, for a disk, $\Delta L/L_0 = V/V_0$ in the direction perpendicular to the flat interface and $\Delta L/L_0 = 0$ in the plane of the interface. The growth of individual grains in disk shapes is common during transformation. The alpha plutonium disks are unusual because they are polycrystalline.

Experimental evidence indicates that the mode of beta to alpha transformation is martensitic. The first grain or nucleus of alpha to form from the surrounding beta parent phase would undergo an anisotropic shape change. This would result in an anisotropic stress field in the beta matrix near the alpha nucleus. Subsequent nucleation would be favored by the stress in some regions of the stress field and inhibited in others. Also, the orientation of subsequent nuclei would be influenced by the stress field of the first nucleus; those nuclei whose shape change was in the direction to relieve the stress would be favored. The disk shape of the transformed region suggests that the stress field surrounding the first nucleus may have cylindrical symmetries; secondary nucleation is favored in a plane. Thus the edgewise growth of the transformed region would be a continuing process of sympathetic nucleation resulting in a polycrystalline disk of increasing diameter. The growth of the individual alpha grains in the disk would be limited by impingement with their neighbors to growth in the direction perpendicular to the plane of the disk.

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 month, examination and testing included the following:

Photomicrography	- - - - -	59
Photomacrography	- - - - -	62
Photomosaics	- - - - -	9
Macrohardness	- - - - -	93
Microsampling	- - - - -	10
Tensile Testing (Rm Temp)	- - -	17
Tensile TEsting (Hi Temp)	- - -	94
Replication	- - - - -	12
Autoradiography	- - - - -	20
Decladding	- - - - -	41
Dissolution	- - - - -	6
Density	- - - - -	40
X-ray	- - - - -	6
H ₂ Analysis	- - - - -	27
Anneal	- - - - -	2
Fission Gas	- - - - -	4

Replication of Irradiated Components by Molding. Evaluation of candidate molding materials was started to improve the dimensional accuracy of replicas made by remote molding techniques. A new casting resin, "Epoxical" thick-section casting resin No. C-301, was tested. The

dimensional accuracy, reproducibility, and mold release characteristics of this resin were excellent. The resin has a low viscosity (7,000 cps), a working time of 45 to 60 minutes, and low shrinkage.

Stereo Zoom Microscope. The positioning motor for the stereo zoom microscope slide has been modified for either DC stepping or AC operation. This change permits the microscope to be used for remote measuring as well as viewing. As-built drawings of the microscope positioner and control circuits are being prepared.

2. Metallography Laboratories

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

Two samples of montmorillinite have been examined to establish that the electron microscope can be used as a tool for particle size and shape determination of this material. A dilute, aqueous solution of the samples was made and a drop deposited on an SiO covered screen and allowed to dry. The particles are somewhat sensitive to the electron beam but with proper care can be examined and photographed with no difficulty. Many of the clay particles are thin enough to allow transmission microscopy, and electron diffraction patterns can be taken with no damage to the material.

A series of zirconium - 5% beryllium samples with various heat treatments were submitted for metal-quality examination. As a sidelight to the routine examination, the effect on the morphology of this alloy was observed. The morphology, or shape change during solidification, paralleled that of AlSi, a system which has been known for some time. A quick chill from the melt produces a highly acicular pattern of $ZrBe_2$ crystals with zirconium between them. Slow cooling from the melt produces square chunks or cubes of $ZrBe_2$ with large areas of pure zirconium between.

During the month, 726 samples were processed, 261 electron micrographs and a total of 1401 optical micrographs and macrographs taken, 2600 negatives printed, and 11,392 prints processed.

3. High Temperature Lattice Test Reactor (HTLTR)

HTLTR Mockup. Run No. 2 goal core temperature of 700 C was reached and maintained for a 115-hour period. Subsequent to this, the mockup was cooled down for heater element and control and safety rod inspection.

Results of B₄C and graphite samples which were intermittently inserted in the core as follows:

B₄C Samples

Sample No.	1	2	2	2	2	2
Exposure time, hr	91.8	24.2	17.7	2.2	14.7	6
Core Temp, °F	520-1020	1020-1070	1260-1310	1310	1300	1300
Initial Wt, gm	12.1733	11.9005	11.8952	11.8986	11.8977	11.9006
Final wt, gm	12.1687	11.8952	11.8986	11.8977	11.9006	11.9008
Wt Change, gm	-0.0046	-0.0053	+0.0034	-0.0009	+0.0029	+0.0002

Graphite Samples

Sample No.	14	14
Exposure time, hr	24.8	64
Core Temp, °F	1070-1260	1290
Initial Wt, gm	13.6483	13.6438
Final Wt, gm	13.6446	13.6429
Wt Change, gm	-0.0037	-0.0009

The 13 graphite samples which remained in the mockup for the duration of Run No. 2 had weight losses ranging from 0.0010 to 0.0047 gram with an average loss of 0.0030 gram.

The 2-1/2" x 1-1/16" x 3/8" B₄C sample placed between two TD nickel plates exhibited a weight gain of 0.0881 gm. Initial and final weights of the B₄C sample were 28.2720 gm and 28.3601 gm, respectively.

The mockup was inspected after Run No. 2, and no increase of core corrosion over that observed at the end of Run No. 1 was visible. One heater element was removed, and no malfunction or corrosion could be detected.

HTLTR Safety and Control Rods. The following is a resume of the problems encountered in the first hot test of the rods:

Vertical Safety Rod: After significant heatup of the mockup core ($\sim 500^\circ\text{C}$), the rod stuck while pulling from the core. Each time the rod was pulled it stuck at between 70 and 80 inches (measured from its full in-position). Once stuck, the rod would neither scram nor pull any further. If left to cool for a long enough time to stabilize the temperature, the rod could be pulled or scrambled as if it had not been stuck. Sticking, temperature equalization, pulling, and successful scrambling of the rod occurred each rod operating cycle while the mockup was at temperature ($\sim 700^\circ\text{C}$). After cooldown no sticking occurred. Subsequent investigation indicated that the dummy spacer unit, installed outside the mockup shell below the bottom of the rod housing to allow full height drops of the rod, caused the sticking. Modification of the plate liner should correct the problem. The dummy spacer unit does not exist on the reactor so this problem may not be applicable to reactor operation.

Horizontal Control Rod: At temperature, the drive mechanism operated under much higher load during the last 1/10-inch of rod travel (inner end of travel). When under this high load, the scram latch would not release. Backing the rod out about 1/10 inch relieved the high load and the rod would scram satisfactorily. Disassembly of the control rod revealed no reason why the heavy load should occur. It is possible that graphite chips, which were present, could lodge in the interface and cause the sticking. The chips were from the uranium oxide bearing graphite outer cylinders. The problem of the scram latch not releasing under high load had been investigated, and a modification has been added to the latch ring. This modification consists of providing a ramp for each ball which tends to make the ring unlock under increasing load on the latch. This arrangement will be tested during the next operating period.

Critical Mass Physics Support Work. Two concentrator boilup tests, using an $\text{Al}(\text{NO}_3)_3 \cdot 9\text{H}_2\text{O}$ spiked solution as a standin for plutonium nitrate, were completed. The concentrator was filled with 200 g/l $\text{Al}(\text{NO}_3)_3$ solution to within approximately 3 inches of the top. Steam at a pressure of 60 psig was started to the coil and maintained at this pressure until the run was completed. Surging of the condensate was observed while the first four liters were condensed. After this, concentrator operation appeared to be normal.

Each liter of condensate collected was sampled and submitted for analysis. The first 3 liters collected averaged 13 g/l Al. Subsequent liters (4-10) dropped to 0.01 to 0.14 g/l.

Environmental Tests on HTLTR Materials. Two environmental tests were run to determine the effect of moisture in the HTLTR atmosphere on the corrosion of control mechanism material composed of 5% B_4C in a graphite matrix. These tests indicated that with water contents of 5000 ppm and 1800 ppm in the nitrogen atmosphere, corrosion of the

control material will occur at 1100 C. In order to obtain more quantitative data on the reactivity of B_4C with the reactor atmosphere, an exposure test was started involving pure, consolidated B_4C in nitrogen with 300 ppm H_2O at 1100 C. Periodic weight measurements of the specimen will be made.

4. EBWR Fuel Elements

Thermal Cycling of Aluminum Core EBWR Rods. The thermal cycling studies directed at determining possible dimensional changes in three aluminum-core EBWR rods have been completed. The rods were heated in a steam autoclave to 370 C and 40 atm in one hour, held at temperature of 3 hours, and cooled to 90 C and 1 atm over an 8-hour period. Two of the rods exhibited rod lengthening of approximately 39 mils after 6 cycles, but subsequent cycling induced rod shortening. After 48 cycles, one rod had shortened approximately 28 mils and the other approximately 63 mils. Anomalous results were obtained on one rod which exhibited little growth or shrinkage up to 28 cycles, but which suddenly started shrinking, leveling off at 48 cycles with a total decrease in length of 33 mils.

Irradiation Testing of EBWR Prototypic Fuel Rods. Seven capsules containing short lengths of vibrationally compacted, EBWR, UO_2 -1.5 wt% PuO_2 fuel rods are being irradiated in the MTR, and 25 have been discharged. The capsules continue to indicate satisfactory performance and have achieved maximum burnups of 4.7×10^{20} fissions/cm³ (16,700 Mwd/ton of fuel) and 4.8×10^{20} fissions/cm³ (17,000 Mwd/ton of fuel), respectively.

Fuel Element No. 6501, the 21-rod bundle (5x5 array with corner rods missing) with full-size replaceable EBWR fuel rods, is currently under irradiation in the core of the PRTR. Current burnup of the element is 1170 Mwd/ton of fuel or 0.33×10^{20} fissions/cm³. It is tentatively planned to remove and replace one rod when the burnup reaches $(0.4-0.6 \times 10^{20})$ fissions/cm³ (1500-2000 Mwd/ton of fuel).

Continued satisfactory behavior has been noted with the twelve, 3-rod clusters of full-size (148-cm long) EBWR rods under irradiation.

5. Other Customer Work

Pu-240 and Pu-241 Irradiation Samples for Phillips Petroleum.

Twenty-three samples of Pu-240-Al and Pu-241-Al alloys are required for fission product transient measurements in the MTR by Phillips Petroleum Company. These samples are similar to samples fabricated in 1962 and 1964 of Pu-239, U-235, U-233, Au and Li aluminum alloys. The samples are tubular and made by coextrusion cladding. The cladding is a high purity aluminum - 1.15 wt% Si (for hardening) alloy. The Pu-240-Al alloys will be made in five concentrations up to

1.6% Pu and the Pu-241 alloys will be made in five concentrations up to 3.2% Pu.

Because of the relatively short half life of Pu-241, the samples will be fabricated as rapidly as possible. In order to permit more rapid fabrication, a die pressing technique which will reduce the machining and handling time is being developed to pre-shape the plutonium alloys prior to extrusion. The cladding alloys have been cast and extruded. The cladding components, tooling, and molds are being fabricated. The samples are scheduled for fabrication in June.

Al-Pu Discs for the Phoenix Program. Five kilograms of plutonium are being fabricated into discs for the Phoenix program. The discs are similar to those fabricated in 1964 by the "spiral-machining" process. The fabrication process consists of casting aluminum 20 wt% Pu alloy billets, extruding the billets into bars, machining the bars into a spiral turning, cutting discs from the turnings, and covering each disc to final size. Four kilograms of plutonium were alloyed and cast into billets to start the fabrication.

IPD PCTR Al-Pu Alloy Fuels. Sixty fuel rods are to be fabricated for this test. Three groups, 15 rods each, will have plutonium-aluminum alloy cores. The alloys will contain approximately 2%, 6%, and 15% plutonium. The fourth group of fuel will be fabricated from compacted thoria. All the fuel will be clad in aluminum. The elements in each group range in length from approximately 7 inches to 35 inches. It is planned that the 6% alloy fuels will be completed by May 1, 1965, followed by the thoria early in May and the 15% and 2% fuels later in the month.

Pneumatic Impaction of UO₂. Work was completed on three contracts for pneumatic impaction of enriched UO₂ for off-site customers. A total of 27 impaction shots were required, in both 2-1/2-inch diameter and 4-inch diameter sizes. Fifty-seven kilograms of enriched UO₂ was impacted, comprising eight different levels of enrichment ranging in U-235 content from 10% to 50%. This work was performed at the request of the Atomic Energy Commission.

HTLTR Fuel Development. A method was developed to weld TD nickel sheet of 20-mil thickness. This magnetic force welding method provided samples for use in evaluation of weld quality by metallographic examination and tensile testing, and for corrosion resistance testing. Metallographic examination and tensile testing indicate magnetic force welding to be a very positive process for joining TD nickel. The tensile test results indicate that room temperature strength at least 70% of that of the parent material can be readily achieved.

UO₃ Characterization. Characterization of UO₃ from the Chemical Processing Department was begun. The UO₃ which was reduced before ballmilling appears to have better sintering characteristics than the material which was reduced after ballmilling. Pellets which were sintered at 1650 C for an 8-hour period attained approximately 90% theoretical density. The O:U ratio of the presintered material was 2.018.

Isotope Source Fabrication. Preparatory development work for the fabrication of P-32, Pr-143, and Ca-45 isotopic disc sources, in aluminum and lead matrix material, was completed. An aluminum disc and a lead disc (3-inch diameter x 1/4-inch thick) were fabricated to evaluate cold pressing parameters and determine acceptability to the customer prior to dispersion of the isotope in the matrix. Fabrication of six disc sources for Radiological Physics, Physics and Instruments Department is scheduled for completion during May 1965.

Saxton Fuel Element Fabrication. Sixty-eight Zircaloy-4 clad Saxton rods were fabricated by vibrational compaction. Densities varied from 86-88% TD. An additional 20 rods will be fabricated by the end of the month. These will include five Zircaloy-4 clad removable rods for a 9 x 9 cluster and 10 rods clad in 15-mil stainless steel. This will complete the Saxton fabrication. Final shipment will be made on May 5, 1965.

Defect Propagation in Zr-4 Cladding. In order to establish the effect of manufacturing defects on fuel rod performance, a study was initiated to determine the effect of thermal cycling on tubing containing rejectable size internal surface defects. Three fuel rods were fabricated by vibratory compaction using Saxton tubing rejected at Battelle-Northwest because of defects in excess of 0.001 inch. The rods are heated in a steam autoclave to 400 C and 68 atm in one hour, held at temperature for three hours, and cooled to 100 C and 1 atm over an 8-hour period. After a thermal cycle, the rods were ultrasonically inspected in region of the known defects. Two rods remained unchanged, but examination of the other indicated propagation of three defects present in the cladding had occurred. Cycling will be continued until definite trends are established, after which time, metallographic examination will be performed on the defects.

Boiling Burnout Studies for the ATR. Heat transfer experiments were conducted with an electrically heated test section which is a full length simulation of a typical rectangular coolant channel of the Advanced Test Reactor. The test section coolant channel was 2.00" wide by 0.070-inch thick by 49.5 inches long. Both sides of the test section were heated along a 48-inch length by passing direct

direct current through copper-nickel bars which were brazed to the 0.010-inch thick stainless steel forming the flow channel. The Cu-Ni bars were machined in an axial profile along the back side to achieve an axial, chopped-cosine heat input pattern with a peak-to-average value of 1.4:1.

The experiments conducted to date were principally the determination of hydraulic demand curves at constant power input rates with coolant velocities from about 1.5 to 5 and 25 to 50 ft/sec. The lower velocity range represents reactor shutdown or emergency conditions while the higher range is typical of normal reactor operating conditions. These experiments were conducted by establishing coolant inlet temperature, flow rate, and outlet pressure at a given power input rate. The pressure drop along the test section was then recorded as the flow rate was decreased in small steps, by an upstream throttle valve, with other parameters held constant. Flow reductions were continued until the channel pressure drop increased appreciably due to boiling and two-phase flow conditions.

The experimental results show that in general the hydraulic demand curve minima occur at bulk coolant outlet temperatures which are slightly below the saturation temperature corresponding to the outlet plenum pressure. Flow rates, at the demand curve minima, are about 1.05 times the flow rates which would be calculated to achieve bulk saturation conditions. This undoubtedly results from nonuniform temperature distribution across the coolant channel and near the narrow edges of the coolant passage where the heat flux is relatively low. Special thermocouple probes were used to sense water temperatures near the outlet end of the coolant channel. The measurements show that temperature differences of 20 to 25 F between the channel center and the channel corners are common with an average "fuel plate" heat flux of 1.8×10^6 Btu/hr-ft² (which corresponds to an axial peak heat flux of 2.5×10^6 Btu/hr-ft²).

One experiment was conducted using a nearly flat supply curve to simulate the unstable flow conditions which result if the demand curve minimum exceeds the available or supplied pressure drop. For this experiment the power input was held constant at an average heat flux of 1.86×10^6 Btu/hr-ft² (2.6×10^6 Btu/hr-ft² peak) while the supply head was decreased in small steps. A flow path in parallel with the test section was used to simulate other fuel coolant channels and adjust the pressure drop supplied across the test section.

As the supplied pressure drop was reduced to slightly above the demand curve minimum, the flow rate became slightly unsteady and showed random fluctuations of about $\pm 3\%$. Then, as the supply

pressure was reduced further, the flow rate suddenly decreased at least 40%, and the heater bar temperatures rose very rapidly. By sudden interruption of electric power input, and by re-establishing the test section flow rate, damage to the test section was averted even though the heat flux at the middle of the test section was 2.6×10^6 Btu/hr-ft².

Manager
Reactor & Materials Technology

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PHYSICS AND INSTRUMENTS DEPARTMENTMONTHLY REPORTAPRIL 1965FISSIONABLE MATERIALS - O2 PROGRAMREACTORCoproduct and Control Rod Experiments for N-Reactor in the PCTR

Perturbation experiments are planned for the PCTR to compare metal targets as used in NPR and the exponential experiment to ceramic targets used in the PCTR. It is also planned to study control rod enhancement by replacing B_4C by B^{10} , and to compare water to masonite as used in the exponential to simulate H_2O . Preparations for the experiment are completed.

Calculations have been resumed to determine the infinite medium multiplication factor k_{∞} and thermal utilization in the coproduct lattice. The cases being completed are wet, dry, and dry without target. The complete $1/v$ flux traverses in both dry cases have been calculated and verified, and verification of the wet case is in progress.

Spatial Distributions of Reaction Rates in N-Reactor

Prestartup predictions of the spatial distributions of reaction rates in N-Reactor over-estimated the magnitude of the experimentally observed thermal peak just outside the core reflector interface by as much as a factor of two. Since this peak is reflected back into the core, earlier methods of calculation also overpredicted power densities in peripheral fuel tubes. Further, estimates of flux flattening and power density in small peripheral test loads can also be expected to be high. To improve this situation, a detailed physical model of the reactor has been developed in two-dimensional group theory using 9-ANGIE.

To date, the zero power-unexposed spatial distributions of $1/v$ reaction rates along the central axes normal to the process tubes have been studied. These calculations account for streaming in the vented sections of the core and reflector, and are presently underestimating the magnitude of the thermal peak by approximately 14%. Adjustments are being made in the streaming corrections to establish the magnitude of their influence upon the peaking. Upon establishing a satisfactory model that describes the startup results, this work will be extended to studies of past and proposed special fuel loadings of the reactor.

Flooding Effects Upon Migration Area of N-Reactor

Since N-Reactor is undermoderated, the initial effect of flooding increases reactivity, decreases migration area, and simultaneously reduces the strength of the control system. Quantitative estimates of the change in migration area as a function of percent flooding in the gas vents, control channels, and safety channels are being made using Behrens' theory for streaming, and Benoist's theoretical model for the age in heterogeneous systems as extended by P. F. Gast. Gast's extension has been made applicable to a four-region rather than three-region cell to account for the addition of the isolated flooding water.

For flooding up to 30% of the void volume the age has been calculated to decrease 5 cm² per percent flooding. Calculations of the changes in diffusion length are under way. Streaming corrections are also being performed for use in the IBM code FLEX-II as a second approach and partial verification of the Benoist-Gast model.

Development of Calculational Methods for Tube-in-Tube Geometry

Comparisons of four-group diffusion and transport calculations for the NPR standard tube-in-tube fuel geometry have been completed and an informal report has been written. In addition to the "cold" temperature case reported in the last progress report, analysis of the cell under "hot" temperature conditions, i.e., cell components at temperatures typical of NPR Phase III (power only) operation, has been completed. For this case, k_{∞} 's of 1.04186 and 1.03713 were obtained, respectively, from the diffusion and the transport approximations--a difference of only 4.7 mk. The errors introduced in the thermal utilization and conversion ratio by using diffusion theory as compared to transport theory were quite small for both the "hot" and the "cold" case--less than a percent.

Other method comparison studies presently in progress are: (1) Comparing the effect of using the Wilkins heavy moderator approximation rather than using the Nelkin water kernel and the Parks graphite kernel for the purpose of generating thermal group constants, and (2) comparing the use of five thermal groups instead of one average thermal group.

Preparation of PCTR Driver-Target Supercell Experiments

Calculations in support of some planned PCTR driver-target supercell experiments are continuing. The supercells will consist of Pu-Al drivers and ThO₂ targets.

In previously reported calculations a driver/target ratio of unity was assumed and k_{∞} was calculated for a wide range of Pu concentrations in the Pu-Al driver.

In order to obtain greater experimental flexibility, the opportunity of removing a part of the ThO₂ targets may be useful. To estimate the effect of such a removal, calculations were performed for the case of complete ThO₂ removal. The ThO₂ reactivity worth was found to be very large and relatively insensitive to the makeup of the Pu-Al driver.

Survey Calculations for Driver-Target Configurations for Production Reactors

Survey calculations for various Pu-Al driver-ThO₂ target arrangements for the production reactors have been continued. Driver/target ratios of 1/1 and 3/1 for various w/o Pu in Al drivers have been examined and the k_{∞} 's of the configurations have been determined. The purpose of these calculations is to arrive at optimum configurations from the point of view of safety and isotope production capability.

Instrumentation

Required modifications were completed on the second neutron fission counter system at N Reactor and successful operation was achieved. Both subcritical channels are now fully operable. Further refinements will be incorporated as required.

Performance testing was continued on the experimental irradiated fuel cooling age monitor now installed at C Reactor. Operation of the equipment for both enriched and normal fuel elements appears successful. The tests included ratio measurements for the Ba¹⁴⁰-La¹⁴⁰ and Zr⁹⁵-Nb⁹⁵ peaks and the Compton "flats". Both ratio techniques are being evaluated for routine use.

System Studies

The N Reactor conversion simulation was modified to permit study of conversion system transients as they affect primary loop pressurizer injection system. The complete system was set up on an EASE 2133 computer patchboard, except for the reactor and primary system pressurizer, which were programmed for the EASE 1132 computer. Coupling of the two computers permits the effects of primary, secondary, or conversion systems disturbances on all major components in the heat dissipation plant to be studied.

Preliminary test runs indicate that realistic effects of conversion system transients on pressurizer level and pressure are obtained with the new

simulation. Seven square root functions, associated with piping or valve flow pressure relationships, were simulated using Thyrite resistors in suitable feedback amplifier circuits.

SEPARATIONS

Criticality Experiments with PuO₂-Polystyrene Assemblies

Effects of neutron interaction on criticality (such as from pipe intersections) and criticality for complex shapes pose some of the most difficult nuclear safety problems in processing and handling fissile materials. During the month, experiments were performed with PuO₂-polystyrene compacts and the RSTM that provide further data on complex shapes such as a cross formed by the intersection of two finite length columns. Due to the shape of the unreflected arrays, the critical mass was large (up to 62 kg Pu); in order to obtain criticality with the Pu on hand, it was necessary to use a mixture of compacts comprised of 2.2% Pu²⁴⁰ and 8.0% Pu²⁴⁰.

The critical mass for a rectangular parallelopiped (12.2 x 10.6 x 27.9 in.) with an H/Pu atomic ratio of 15 was measured as 62 kg Pu. The effect on criticality of the addition of another array, at right angles to and intersecting the original parallelopiped at its midpoint, was then determined. Criticality measurements were made with the intersecting column having several different cross sectional dimensions and lengths. As a result of the variations in geometry, the critical mass varied between 62 to 40.5 kg Pu. The analyses of the data have not been completed.

Of particular interest in the design of shipping containers, annular type fuel element dissolvers, and annular type vessels for solutions of fissile materials, is the thickness of neutron absorbing materials required to effect neutron isolation. To obtain further data for use in this regard, a series of critical experiments were begun with the RSTM utilizing PuO₂-polystyrene assemblies. The neutron isolation properties of polyethylene, borated polyethylene, polypropylene, and compressed wood will be studied. In some cases, cadmium sheet will be placed between this material and the two portions of the fissile core.

Subcritical Experiments with Enriched N-Fuels for Nuclear Safety Guidance

A series of neutron multiplication experiments were completed with enriched uranium tubes (1.36 in. i.d., 2.35 in.o.d.) of 1.95 wt% U²³⁵ in light water. The multiplication measurements were made on cylindrical arrays of fuel columns of two different heights (27.3 in. and 40.3 in.); three lattice spacings were used. The critical mass was determined from extrapolation of the inverse multiplication curves in each of these cases

with the array containing from 90-95% of the critical number of tubes.

An attempt was also made to determine the effect on criticality of placing target elements within the tubes. However, too few target elements were available and only a crude indication of the effect could be determined from the change in multiplication--that the critical number of target loaded elements would be increased by a factor of two or more.

These experiments provide data needed for nuclear safety guidance in handling enriched N-Reactor fuels. The results are given in the following table. Referring to the table, it is seen that criticality can be achieved with only 26.5 of the 27.3 in. long fuel tubes and 21.4 of the 40.3 in. long fuel tubes.

Critical Mass Laboratory Instrumentation

Work has proceeded on the noise analysis system at the Critical Mass Laboratory. Because of a noticeable voltage drift in the frequency selective amplifiers, a method was needed for quickly calibrating the system before beginning an experiment. It was determined this could best be done with a tape recorder. The center frequencies of the selective amplifiers were found using an oscillator. These frequencies were then individually recorded on tape. When the recorded signal is played back into the noise analyzer, the search for the center frequency is eliminated and an accurate determination of the drift can be made and compensated for.

Consulting Services on Nuclear Safety - Criticality Hazards

1. Nuclear Safety in Pacific Northwest Laboratory

The following nuclear safety specifications were reviewed:

- a. Temporary Specification No. 10 for Metallurgy Development giving limits for casting 4.0 kg of Pu-Zr alloy.
- b. Specification H-1 for Radiometallurgy which revises limits for storing plutonium and uranium in the dry storage cell.
- c. Specification K-1 for PRTR revising limits for handling Mark I Pu-Al and PuO₂-UO₂ fuel elements.
- d. Specification T-2 for Waste Disposal and Decontamination which gives limits for plutonium and uranium disposal in dry burial tubes.

CRITICAL MASSES AND BUCKLINGS FOR 1.95 WT%
ENRICHED URANIUM TUBES DETERMINED FROM
NEUTRON MULTIPLICATION MEASUREMENTS

OFFICIAL USE ONLY

<u>Lattice Spacing (in.)</u>	<u>H₂O/U Volume Ratio</u>	<u>Height of Fuel Column (in.)</u>	<u>Critical No.⁽¹⁾ of Tubes in Cylinder</u>	<u>Effective Radius (cm.)</u>	<u>Estimated⁽²⁾ Extrapolation Length (cm.)</u>	<u>Buckling⁽³⁾ (10⁻⁶ cm⁻²)</u>	<u>Calculated Critical Mass for Spherical Geometry (Kg U)</u>
2.80	1.58	27.3	30.96 ± 0.13	20.78	7.81	8442	533.2
		40.3	28.43 ± 0.16	19.91			
3.10	2.19	27.3	26.48 ± 0.17	21.27	7.62	8309	458.7
		40.3	21.36 ± 0.10	19.11			
3.40	2.87	27.3	27.88 ± 0.10	23.94	7.39	7287	501.3
		40.3	24.44 ± 0.20	22.42			

B-6

- (1) Error estimated from standard deviation on least squares fit to multiplication curve.
- (2) Extrapolation lengths were assumed to have values similar to those for 1.00 wt% and 1.25 wt% U²³⁵ enriched tubes.
- (3) The bucklings were computed from data on the 27.3 in. high cylinders; the cylinders of 40.3 in. height require correction for effect of end caps.

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BNWC-8 4

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

Two Class III shipments were reviewed and approved:

One shipment to Illinois and Pennsylvania consisted of 103 fuel rods of 1.5 wt% PuO₂-UO₂; 17.6 kg plutonium metal; 25 g U²³³; 90 rods of 6.6 wt% PuO₂-UO₂; and 8.5 kg 20 wt% U²³⁵ enriched uranium metal.

The other, to California, contained 200 g (PuNO₃)₄ and 700 g plutonium as metal.

3. Nuclear Safety in CPD

The annual review of Criticality Hazards Controls in the Chemical Processing Department was performed on April 27, 28, and 29, 1965. Members of the review committee were: E. D. Clayton (Secretary), PNL, J. T. Thomas, ORNL, D. R. Smith, LASL, and W. B. Lewis, Idaho Falls. During the review detailed discussions were held with personnel of CPD and various portions of the plant were toured.

4. Nuclear Safety in Eurochemic

An advisory panel was convened by Euratom to perform a safety review of the Eurochemic Plant at Mol, Belgium, to determine the possible effects that the inclusion of high enriched MTR type fuels (up to 93 wt% U²³⁵) might have on safety in the reprocessing program at Eurochemic. By request received from Division of International Affairs, a representative from Pacific Northwest Laboratory was to serve on the Panel. The Panel convened at Euratom Headquarters from March 29 to April 2, with a visit to the Eurochemic site on March 30. E. D. Clayton served as chairman for the review.

Separations Instrumentation and System Studies

Performance testing was continued on the experimental instrument developed to measure Am²⁴¹ distribution in a chemical process resin column at the 234-5 Building. Indications are that no further circuitry or detector refinements will be necessary.

Development work was accelerated on the radiation resistant and high temperature preamplifier welded module (TIMM) circuits. An oven was used to maintain the required 580°C temperature to about ±2°C. The active circuit elements (22917 triode and various other components) were assembled in the oven and voltage-current tests were initiated. The obtained characteristic curves indicate correct performance.

METALLURGY - Nondestructive TestingN Fuels Testing

The surface contamination tester, developed to monitor uranium contamination on finished N fuels, is undergoing production evaluation by N Fuels Quality Control personnel. The end cap test heads performed satisfactorily and the clad test heads are currently being evaluated. An operating and maintenance manual for this system is 80% complete.

The chevron end tester, which is used to test the bond quality of the beveled portion of the chevron end cap, has been delivered to N Fuels for use as a production tester. A production line type of transducer manipulator has been designed, fabricated, and delivered. This relatively complex testing system is successfully calibrated and maintained in the production plant.

Fabrication of the TIG braze closure test, which is to be used to determine whether the braze wire melted during the welding cycle, is proceeding on schedule. Fabrication of the mechanical portion of the testing station has been completed. The electronic portion of the test station is 80% complete. An operational prototype of the system is scheduled for delivery to N Fuels on June 1, 1965.

The circuit design work required to convert the production line bond testers from pulse echo systems to through-transmission systems has been completed. Evaluation of the through-transmission system on the production line will begin immediately.

Fabrication of mechanical portions of the remotely operated irradiated fuel test station (UT-10B) for use in 105-N fuel examination basin is continuing at the 189-F Hot Shop. Certain minor, but time consuming, problems have been encountered in fitting the new components to those removed from the old tester. A major delay was encountered when it was found that the "Buna-N" wheels were not bonded to the drive heads. The necessary modifications were made offsite. The new control circuit was mocked up at 189-F and successfully put into operation. The internal bond test probes were irreparably damaged during the transfer from 105-KE to 105-N. One size of each probe (NOE and NIE) remain available as spares. Control cables between the UT-10 console and the drive motors were lost and replacement cables have been ordered.

Fabrication of a prototype eddy current tester to detect clad thinning striae on irradiated N Reactor fuels is almost complete. Completion is pending receipt of special order items. Most of the special parts have now been ordered for the project phase unit.

The converted penetration tester, used for first beneficial operation, was cleaned up and made ready for insertion in the UT-10 console. A new striae NIE standard has been extruded and will be checked out with the tester as soon as possible. Design and detail work has been completed on the heads and cam lifting mechanism and these parts are being fabricated.

N Reactor Department - Test Engineering Support

Laboratory work has been completed on a study to establish design criteria for a system to nondestructively identify enrichment of fuel elements as they are charged into N Reactor. A report is being issued to document this work. A laboratory test setup was assembled to provide the desired information under simulated charging and loading machine conditions. Laboratory operation was significantly improved by the use of an automatic stabilizer that compensates for photomultiplier tube drift. Geometry problems encountered from simulated field conditions were surmounted by the use of a ratio technique. The activity of 185 keV gamma from U^{235} is proportional to enrichment; however, a ratio of the 185 keV to 250 keV gamma activity was found to be a more effective measurement.

As presently visualized, test heads located at both test points will be connected by long cables to one console in an adjacent room so that only one analyzer installation is required.

AlSi Fuels Testing

Work was accelerated on the development of the dual bond tester which will monitor both the external and the internal bond quality of AlSi fuels using a single external transducer. At the request of the customer, the basic testing technique was changed from pulse echo to through transmission. This change necessitated substantial redesign of previously developed electronics. The complete tester was breadboarded in the laboratory and a subsequent checkout proved satisfactory. Shop fabrication of a working prototype has been initiated.

Efforts were continued to determine the feasibility of detecting the formation of welding defects by spectroscopically monitoring the welding arc. Several pallets (~ 200 pieces) of both AlSi and hot die size fuels were spectrophotometrically monitored while being welded. Although the results were encouraging, in that all types of imperfections (such as low gas pressure, weld overhang, bumps, and dirt) produced some effect on the recorded signal, no pinhole defects occurred in the fuels which were tested. Because there is no way to be sure that pinhole defects can be observed by this test, short of actually observing one, further tests will be necessary. The borrowed equipment which was used in the preliminary tests had to be

returned, and orders are being placed for new equipment to complete the study.

IPD Engineering Support

A study is being conducted to determine the feasibility of applying an ultrasonic technique to determine the percent bonding of ribs welded inside Zircaloy-2 process tubes. High sensitivity, high resolution, 10 Mc PZT-5 transducers were purchased. Using these transducers with a broadband test system, good correlation with the standard was obtained. This system has sufficient resolution to readily detect and separate signals from the unbond and tube wall interfaces which are only .040 inch apart.

Using a prototype manual probe, which rides between the ribs and tests them one at a time, the tube sections provided by IPD were scanned and representative samples have been selected for metallographic examination. A section cut from a 15-ft. long tube was also scanned, but severe misalignment of one of the ribs reduced the reflected signal below a usable level. This condition represents a manufacturing defect and the test system will correctly trip the reject alarm.

Severe case hydriding in two Zircaloy-2 process tubes has created the need for a rapid, accurate, nondestructive method of detecting the presence of hydriding in installed reactor process tubes. During this month, several tests were made in an attempt to detect this condition. One method consisted of measuring the thermocouple voltage produced at the heated junction occurring between a pure zirconium wire and the process tube. This thermocouple, heated by a modified soldering gun, produced about 0.3 millivolt with no hydride and 0.5 to 0.7 millivolt with hydride. Unfortunately, this small output cannot be measured because of background voltages present in the reactor system. Measurements were attempted on one process tube in 105-C and on two tubes in 105-KW and 105-KE. Three readout devices were used with poor success in all cases. A battery-operated, one millivolt meter has been ordered and will be tried with a newly designed probe in the near future.

A preliminary study of the problem indicated that useful eddy current frequencies would be in the megacycle range. A 5 Mc eddy current testing system, built for other programs, was modified to investigate hydride detection on the inside of Zircaloy-2 process tubes. The results indicated that 5 Mc eddy currents can differentiate between hydrided and nonhydride samples if probe spacing is accurately controlled. It appears a positioning reproducibility on the order of ± 1 or 2 mils may be necessary.

An investigation of the feasibility of using ultrasonic techniques to detect case hydriding in small diameter Zircaloy-2 process tubes was also undertaken on an accelerated basis. The wall thickness (nominal .040 inch) prevented use of velocity comparison method being developed for the much thicker NPR process tubes. Brief investigations of surface, critical angle and shear wave tests to detect a thin (.002-.004 in.) hyride case on the Zircaloy-2 wall were conducted without success.

NEUTRON CROSS SECTION PROGRAM - 02/04 PROGRAM

Triple-Axis Spectrometer

Studies have been continued to find an analyzing crystal of good uniformity and reflectivity. These studies have centered on an available copper ingot which is known to have reasonably good reflectivity properties. A method has been developed to cut thin slabs (about 0.060 in.) with minimum distortion to study the gain in reflectivity in stacked crystals.

A study was made in collaboration with Prof. S. W. Peterson, WSU, of the use of the spectrometer for structure analysis. Diffraction patterns obtained with single crystals of NaCl and a copper salt yielded expected diffracted intensities.

Time-of-Flight Spectroscopy for Slow Neutrons

Further measurements were made on the scattering of slow neutrons from zirconium hydride. These included a study of the elastic scattering for an incident energy of 0.1 eV and a measurement of inelastic scattering for an incident energy of 0.38 eV. The widths, energy levels, and angular distributions of inelastic scattering with excitation of each of the first three optical levels have been extracted from the data. Good agreement has been obtained on the results of excitations of the first and second levels with different incident energies. The total scattering associated with the elastic peak and each of the optical levels has been determined to be in good agreement with the cross section values and angular distributions expected from a simple harmonic oscillator. No further measurements are planned on zirconium hydride until better samples of known hydration and crystal structure are obtained.

Attempts to study the interpretation of the inelastic scattering from Zr-H by a multiphonon calculation which rigorously includes the contribution of the acoustical modes has been delayed by deficiencies and errors found in the Harwell computer programs LEAP and ADDELTA. A subroutine in LEAP was found to fail to give convergence for the range of input parameters needed in this calculation. This subroutine has now been modified to converge over a greatly extended range of input parameters. The multi-

phonon scattering formula which is calculated by ADDELT was found to contain an error of a factor of two in one parameter. This rather astounding error is easily correctable, but invalidates all previous attempts at Hanford to interpret the scattering from H₂O using the LEAP-ADDELT calculation.

Fast-Neutron Cross Sections

A series of 3- to 15-MeV total cross section measurements was successfully completed on Li⁶, Li⁷, Ca⁴², Ca⁴⁴, Tc⁹⁹, W¹⁸², and W¹⁸⁶. Prior to the measurements most of the detector electronics were replaced by commercial fast-pulse modules. The system performed admirably during the measurements giving as good a time resolution and better gamma discrimination than had previously been achieved. In the process an instrument malfunction which caused a loss of data in the 8- to 11-MeV region in the past several runs was found to be due to a bad resistor in the chronotron. The ORTEC ion source was used in the Van de Graaff for the first time in these measurements and gave greatly improved performance in beam current, molecular: atomic beam ratio, beam focus, and source lifetime. The loading on the accelerator vacuum column also appears to have been reduced.

Work has also progressed on improvements to the computer programs TAPEWORM, LIL NED, and BIG NED.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Pu-Al Kinetics Experiments with D₂O Moderator

Reactivity measurements of control and safety rods in a D₂O-moderated loading of 19-rod Pu-Al clusters in the Plutonium Recycle Critical Facility (PRCF) have been completed. The measurements were made by singly, and simultaneously inserting three safety and three control rods. The rod worths were obtained by comparing the resulting flux transients to those calculated for stepchanges in reactivity. All control rod measurements were made with a moderator level of 105.25". The safety rods were measured with the moderator level one inch lower. Safety rod #1 which was measured at both levels was worth 4% less at the lower level. In order to keep the (γ,n) neutrons from affecting the measurements it was necessary to insert the rods at different power levels. All control rod (CR) insertions were made after operating at critical at a power level of 2.6 watts. The safety rod (SR) insertions were made at 30 watts for single rods and at 60 watts for all three together. The results obtained with their RMS errors are:

CR #1 - 102.2 ± 1.0 cents
CR #2 = 98.8 ± 0.6 cents
CR #3 = 101.4 ± 1.2 cents

SR #1 = 1786 ± 26 cents
SR #2 = 1882 ± 41 cents
SR #3 = 1970 ± 22 cents

3 CR (together) = 305.1 ± 1.5 cents

3 SR (together) = 6382 ± 32 cents

Three recordings of the reactor noise were made for the Pu-Al fuel loading in the PRCF. Each of the recordings were made with BF_3 ion chambers in a different position with respect to the core. This was done to determine the effects of detector position on the reactor noise spectrum. The first recording was taken with a single BF_3 ion chamber (1" dia. x 14" long) strapped to the central fuel cluster of a seven cluster hexagonal array and 32" above the fuel center. The second noise recording was made from two larger chambers (4" dia. x 22" long) on top of the reactor tank at the vertical axis. The third recording was made from three small BF_3 ion chambers (1-3/8" dia. x 9" long) placed symmetrically near the outer edge of the reactor tank at 27" above the fuel center and at a radius of 33.75". Difficulty was encountered during recordings due to power level drift of the reactor. This generally limited the recording time with no control rod or electronics adjustments to less than ten minutes. However, one 20 minute recording was obtained with no adjustments.

The completion of a 10-channel frequency analyzer to be used in reactor noise experiments is being delayed because of late arrival of parts.

Two measurements were made of the flux transient obtained by shooting a boron arrow into a thimble near the central fuel cluster. The second measurement was made to improve the data for the first two seconds of decay which was recorded by an oscilloscope camera. The flux decay occurring one second after the insertion was recorded at three second intervals by a digital voltmeter. The reactivity worths of the boron arrow obtained from the flux transient data are 45.84 ± 0.45 cents and 46.10 ± 0.30 cents. The worths were also measured by positive periods as 47.301 cents and 47.294 cents, respectively. The analysis of the transient data to obtain kinetic parameters has just begun.

Spectra and Reactivity Coefficient Studies

Flux traverses in the vertical and radial directions were obtained with a bare and cadmium covered BF_3 chamber in the D_2O -moderated loading of Pu-Al fuel elements. Data of the reactivity of cadmium were also obtained as a function of radius and vertical height.

The counting data from bare and cadmium covered BF_3 chambers from the vertical traverse were each fitted by a cosine function by the method of

least squares. The calibration constant relating the counts from a 1/4" BF_3 chamber which were used to obtain, respectively, bare and cadmium covered flux traverses was measured as 8.02. Using the calibration constant and the parameters which best fit the cosine function, the average cadmium ratio over the fuel was determined to be 46.2 for the BF_3 detectors. The radial traverse obtained with a bare chamber was not symmetric about the geometric center of the loading. It was concluded that the Cd cover was close enough to the bare counter to perturb its count rates. This was verified by a traverse which was obtained with gold foils and was found to be nearly symmetrical. The data of the radial traverse were compared to those calculated using diffusion theory, and the peaking of the flux in the D_2O reflector is not as pronounced as is calculated by diffusion theory.

An experiment to determine the spectral indices T and r in a Pu-Al cluster and in the moderator was completed for the D_2O -moderated Pu-Al loading of the PRCF. The measurements were made using bare and cadmium covered lutetium foils. The activities of the two lutetium isotopes have been counted and analysis of the data have begun using program LULU.

The spectral indices of a neutron environment is calculated with program LULU from activation data. A rough draft of a document describing program LULU has been written.

A rough draft of a description of a program CUTOFF has been written. An effective cutoff energy for transmission of neutrons through a specific filter material covering a given detector material is calculated with program CUTOFF.

The moderator coefficient of reactivity was determined for a moderator level of 89.4" as 1.664 mk/in. This is about 1.7 times what it is for a two-zone D_2O core of Pu-Al and UO_2 elements which was studied previously.

A new input routine for Program SWAP has been written. Program SWAP uses diffusion and perturbation theory to compute reactor parameters and reactivity coefficients. The new routine simplifies the data input for a series of computations which are frequently made to aid in experimental planning. A rough draft of the description of the input routine has been written.

Pu-Al Burnup Experiments

Work has continued on fitting functions by the method of least squares to experimental burnup data. The data have been obtained from samples of

Pu-Al fuel rods in the outer ring of a 19-rod cluster which has been irradiated in the PRTR. Using computer codes GEORGE, MISFIT, and ORTHO, many functions were used in an attempt to analytically describe the plutonium isotopic ratios as a function of Pu-239 atom density. The functions that best fit the data are of the form

$$\sum_{i=0}^n A_i X^{-i}.$$

These functions are readily adaptable to methods of integration used to calculate the ratio of nonfission to fission cross section for Pu-239, α -239, and other cross section ratios.

Experiments with 2 w/o PuO₂-UO₂ Fuel

Preparations for a series of approach-to-critical and exponential experiments which will be conducted in the Critical Approach Facility are complete. Three types of fuel that differ in Pu-240 content will be used in the experiments. The isotopic concentrations of Pu-240 in the three types of fuel are 8, 16, and 27%. The fissionable material in the fuel rods is 2 w/o PuO₂ in natural UO₂, is 1/2" in diameter and 36" in length, and is clad by 0.030" Zircaloy.

The span of lattice spacings required to obtain adequate data range from 0.66 to 1.386". Lattice plates which have been used in other experiments and two new sets which are being fabricated will provide lattice spacings of 0.66, 0.80, 0.93, 1.05, 1.143, 1.32, and 1.386". This entire set of lattice spacings will probably not be used on each fuel type.

Experiments with Control Rod Materials

Rods that contain hafnium and boron are being fabricated. It is expected that they will be available by the end of the month. Plans have been made to measure the worths of these rods (and others that have been fabricated earlier) in the Critical Approach Facility in conjunction with approach-to-critical experiments.

PRCF Experiments with 1.8 w/o Pu-Al

Measurements of the reactivity worth of a copper rod at several radial positions in an H₂O-moderated loading of 1.8 w/o Pu-Al rods in the PRCF

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have been analyzed. If absorption at high energies is neglected and copper is assumed to be an absorber only, then the worth of the copper rod is proportional to the product of the flux and adjoint flux. Thus, under these assumptions, the radial distribution of the adjoint flux can be obtained by dividing the radial worth curve by the measured thermal flux.

PuO₂-UO₂ Graphite Lattice Studies

Measurements in the PCTR on the 6-1/2" lattice, fueled with 19-rod clusters containing 0.9 w/o L_x PuO₂ in depleted UO₂, are complete. The multiplication of the lattice is very close to, but is less than, one. Reactivity coefficients for copper, and the cell were measured. Foils containing Pu, U-235, Cu, and Au were irradiated in the lattice and the thermal column. These data are being analyzed to determine k_∞. The experimental cross section ratios and foil traverse provide a good check for a computer code. Since the multiplication of the 6-1/2" lattice is so close to unity, the measured cross section ratios are close to those for the unpoisoned infinite system. Comparisons are being made between experiment and the values predicted by the computer code THERMOS.

Reactor Physics Code Development

1. HRG

Several modifications have been made to HRG in recent weeks. The only one which influences the calculation of flux and broad group cross sections is a revision of the evaluation of the integral giving the unresolved resonance contribution of a nuclide with resonance parameter table. The integral is now evaluated by a more accurate method than the trapezoidal rule previously used, and the contribution is distributed throughout the fine groups spanning the unresolved region rather than being lumped in particular fine groups as before. Tentative results indicate that total resonance contributions are changed by a few tenths of a percent, but may be of either sign.

Three new options have been added to the code. One permits the punching of either microscopic or macroscopic cross sections for input to HFN, not just macroscopic as before. Another permits both flux and current to be read in, as an alternate to the previous option which sets the current to zero when the flux is read in. This option is useful in those cases where average cross sections are requested for many nuclides appearing in trace amounts, and gives the user the luxury of subsequently adding nuclides to his list without rerunning the entire case. To facilitate the use of this second added option, a third one has been

added which permits the fine-group fluxes and currents to be punched.

The remaining modifications affect the form of the output only. The Sum of Resonance Contributions of a nuclide with resonance parameter tables is now printed under that more accurate caption, rather than the previously used Resonance Integral. The distribution among the fine groups of the components of this sum (each converted to an equivalent fine group cross section by dividing by .25, the lethargy width of a fine group) is also printed. A final modification is the printing of the flux normalization factor to permit a check of neutron conservation to be made.

2. RBU

Writing of the theory and formulation portion of the three volume RBU publication was temporarily discontinued in mid-February in order to complete certain indicated modifications to the system. Of these, magnetic tape "Bad Spot" erasure logic has been programmed for the system, and now is in the debug stage preliminary to being placed in service throughout RBU. This particular logic has long been indicated and should substantially lower the probability of losing a long computer run because of previously fatal magnetic tape failures.

Because of the changes, a limited amount of document rewriting is to be required before the drafts are submitted to Technical Publications.

3. Program UPDATE

A need for updating the Hanford Basic Library with the standard monitor has been met. In the past, the library was updated and then the RBU Library editor was run as a first level error detection system; however, this had inherent problems because updating and editing were carried out by two different programs both of which were written in SOS language. The new UPDATE program was written in FORTRAN IV which has reduced the editor listing from more than one reel of tape to less than one-half reel of tape. UPDATE will not allow the new isotope record to be written on the library tape unless all of the basic error detection schemes approve of its data. If approval is not granted, the old isotope record is copied and updating proceeds. Several of the basic subroutines were obtained from BARNES II and modified to FORTRAN IV for use in UPDATE. UPDATE reduced the computer time requirements from approximately 7-9 minutes per update to approximately one minute, since only the updated record is written as output in the RBU editor format.

4. Program MULE

MULE, a program for multi-level, multi-channel calculations of neutron cross sections with Doppler and resolution broadening, was obtained from KAPL. MULE was originally written for the Philco-2000 computer and required conversion for use at Pacific Northwest Laboratory. MULE uses the Wigner-Eisenbud-Vogt formalism in calculating the neutron cross sections. MULE is debugged and is currently running in FORTRAN IV.

Isotopic Analysis of PRTR Samples

Isotopic analyses were performed on 45 burnup samples of PRTR-irradiated fuel elements. Of these, 2 were uranium samples from $\text{PuO}_2\text{-UO}_2$ fuel element number 5185, 9 were plutonium samples from $\text{PuO}_2\text{-UO}_2$ fuel element number 5187, 19 were uranium samples from UO_2 fuel element number 1009, and 9 uranium and 6 plutonium samples were from UO_2 fuel element number C-1101-1.

Instrumentation and System Studies

General design was completed on the new scanning instrumentation to be incorporated into the PRTR fuel element rupture monitor to effect improved performance and reliability. Fabrication of the scanner will commence as soon as time permits. The new strip-chart recorders were finally obtained for use with the monitor.

Information was obtained for a hazards analysis of the Plutonium Recycle Test Reactor with increased power level. The analog simulation runs were satisfactorily completed.

EBWR PROGRAM

A rough draft of a document describing pre-analysis of planned experiments with the plutonium demonstration experiment in the Experimental Boiling Water Reactor has been written and is being reviewed. This report describes the critical loadings and reactivity changes which are expected during the startup experiments and at various stages of burnup of the core.

PHOENIX FUEL PROGRAM

Pu-Al Polyethylene Experiment in the PCTR

After a reduction in the original boron concentration of about 30%, the Pu-Al lattice with 27% Pu-240 and an H/Pu ratio of 200 was found to be still over-poisoned by about 5%. An accurate determination of the critical

boron concentration has not yet been possible because of poor reproducibility of reactivity measurements.

Tripling the thickness of the fuel layers (maintaining the same fuel-to-moderator ratio) was found to reduce the critical boron concentration by about 10%. It will be necessary to reload the buffer cans for a more accurate determination of this effect.

Phoenix Test Core

The possibility of using the APPR as a Phoenix fuel test bed is being examined. The purpose of the present investigation is to arrive at a rough cost estimate for such an experiment. The core fabrication cost represents a major cost item. Calculations of fuel inventory have started.

The APPR is a 10 Mw thermal, 140 ~~l~~ pressurized water system with $U^{235}O_2$ fuel in a stainless steel matrix and clad with stainless steel. The L/D is 0.989 and the metal-to-water ratio is 0.262. The coolant water temperature is 450°F at a pressure of 1200 psi. The core consists of a 7 x 7 box array with the corners missing. The fuel boxes are 2.8" on a side and contain 18, 20 mil plates with a 5 mil clad. Seven of the boxes are control rods with 16 plate fuel box followers. The initial loading contained 22.5 kg of U-235 and 21 gms of B-10 burnable poison with all followers in.

The first step in determining the amount of fuel needed for a good experiment is selection of a calculational model. At present we are checking the ZODIAC Program against the initial U^{235} -fueled core in the APPR. A homogenized representation of the core gives a k_{eff} of 1.137 compared to an experimental value of 1.072. Inclusion of heterogeneous effects is expected to lower the calculated value.

Pu-Be Studies

Survey calculations for possible PCTR experiments using Pu-Al discs and beryllium metal are about 50% completed. An iterative method for treating the thermal and Pu-240 resonance self-shielding effects is being employed in connection with the ZODIAC code. The proposed poison material to be employed is assumed to be natural boron alloyed with aluminum and in the form of discs.

Calculations of Pu-240 Resonance Integral

An analysis of Nichols' experimental determination of the effective resonance integral for Pu-240 (NS&E, 17, pp. 144-148, 1963) using the resonance integral codes ZUT and TUZ was completed. The objective of this

analysis was to test the application of these codes to the treatment of the broad Pu-240 resonance. The contributions from resonances above 1 eV, resolved and unresolved, can now be calculated for each core. In addition, the cylindrical experimental geometry can be treated explicitly rather than using the rational approximation.

The conclusion of this study is that the calculated values of the resonance integral agree with the observed values to within the experimental error as estimated by Nichols, and that the NR1A approximation and the integral method used in ZUT give essentially identical results for the large 1 eV resonance. The results of the study will be published in an informal report.

FAST FLUX TEST FACILITY PROGRAM

FFTF Nuclear Design

An investigation of various proposed FFTF reflector control schemes has been carried out. Reflector control of the FFTF appears to be marginal, and supplementary in-core control devices will probably be necessary. The results of this study are being issued informally.

To assist in the design of the coolant systems, preliminary calculations were made to determine post-shutdown heating due to both fuel fission and fission product decay. Primary emphasis was given to the analysis of the short term energy release following a reactor scram.

The fuel for the FFTF is presently envisioned to be a PuO₂-SS cermet. A study of the use of alternate fuels and structural materials in the FFTF has been started.

Comparison of Cross Section Compilations

The new Argonne 22-group cross section set has been obtained from ANL. It is now being checked and prepared for use in various multi-group codes.

At present, the 16-group YOM set, the 16-group Hansen-Roach set, and the 26-group Russian set have been checked and are available for use. A comparison of the four sets will be made in the near future.

Fast Reactor Fuel Cycle Code

A zero-dimensional fuel cycle code for fast reactor analysis has been written and debugged. The code can:

- (1) Use up to 26 groups with as many as 40 isotopes (20 burnable).

- (2) March-out at constant power or flux.
- (3) Begin at any initial multiplication constant by iterating on leakage.
- (4) Recompute and print average cross sections at each time step, etc.

A detailed description of the code is being issued as a BNWL document.

Instrumentation and System Studies

Instrumentation studies are under way to provide a conceptual design and cost estimate of the instrument and control system for the reactor and heat dissipation processes, and a detailed estimate of instrument development costs through the completion of the project. Assembly of FFTF prototype instrumentation was begun in the laboratory; work on the display system continues. Contributions to the program during the month include the review of many documents on fast reactor technology, comment on eleven scope drawings, a dissertation on our present control concepts for the benefit of the Nuclear Safeguards and Engineering Section, and the submission of a preliminary system scope document for inclusion in a progress report to the AEC Division of Reactor Development.

Manual hand and machine calculations were conducted to establish approximate dimensions and composition of the planned FFTF shield. Machine calculations to establish the possible need for an optimum location of boron in the side shield were inconclusive because of invalid results at lower energies. The biological shield requirements were estimated on the basis of activation levels of the sodium primary coolant. Refinements will be necessary as parameters become more firmly established. In addition, instrumentation requirement studies will be conducted as required, especially regarding a fuel rupture detection system.

Systems analysis and simulation studies of the FFTF were initiated with familiarization studies of the important design variables. A partial table of nomenclature was made and listed on the IBM card lister. The optimization equations were reviewed and prepared for analog simulation. Design equations were obtained from the Design Analysis Unit and were programmed for MIDAS and for analog computation. The analog program for design calculations requires 38 multipliers and dividers. When the optimization equations are included, the computer facility will lack approximately 12 multipliers.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Mockup Tests

The initial part of the second run of the HTLTR mockup has been completed.

This consisted of a heatup to 700°C, where the temperature was held for 115 hours, followed by a cooldown. The prototype control rod, safety blade, and heater element, when removed for inspection, showed no visible deterioration. The control rod consists of a set of alternate 6" long cylinders of UO₂-graphite matrices mounted inside a set of larger diameter cylinders of graphite and UO₂-graphite. The UO₂-graphite used in the test is not equal in strength and integrity to that called for in the design. It was introduced at this time so that the chemical compatibility of the control rod materials could be observed. Even this weaker material stood up quite well under the mechanical stresses of scrambling at 700°C. A later prolonged test will be carried out with a control rod containing UO₂-graphite matrix materials that meet the design specifications.

During the run a small gas cleanup system was introduced. This reduced the contamination level of water in the nitrogen gas in the mockup to 1200 ppm-- a factor of 3 or 4 lower than the level without the cleanup system. Small graphite samples exposed in the mockup showed weight losses of less than $25 \times 10^{-8} \text{ gm hr}^{-1} \text{ cm}^{-2}$. A B₄C-graphite sample showed a weight gain of $3.3 \times 10^{-5} \text{ gm hr}^{-1} \text{ cm}^{-2}$.

During the second part of the run the temperature will be raised to 1000°C and maintained there for 10 days. Heater elements, the control rod, safety blade, and small samples of materials will again be in the mockup. The weight change data from the small samples will be used to estimate graphite corrosion and boron evolution to be expected in the HTLTR.

Fuel Element Development

The material presently being considered for fuel element cladding is T.D. nickel. The conditions for properly welding this material have been worked out. Metallographic analysis shows that the thoria dispersion does not deteriorate during the welding process. The tensile strength of the weld is about 0.6 of the strength of the parent material at room temperature. Tensile strength measurements will be made at 700°C and 1000°C.

Vibrapacking of UO₂ in a metal tube of annular cross section, approximately of the geometry considered for the HTLTR driver fuel elements, shows that densities of 80% can be consistently achieved to within 1%. A 68% dense packing was achieved in a graphite tube. Graphite is also being considered as a driver fuel cladding material.

Hazards Report

Work on the Final Hazards Analysis document continued and will extend over the next few months. The document will provide the basis for safeguards

reviews leading to startup approvals. A preliminary report, HW-82372, issued in June 1964, is being updated. The new document will reflect:

- (1) A better definition of the nuclear characteristics of the HTLTR system,
- (2) high temperature mockup experience with various components including the control and safety mechanisms, (3) a feasibility determination on the fuel fabrication process, (4) changes in the instrumentation system, and
- (5) miscellaneous other improvements.

FUEL CYCLE ANALYSIS

Comparison of Chemical Analysis of Yankee and Vallecitos Fuel Discharges with MELEAGER Burnup Calculations

A comprehensive chemical analysis of some Yankee and Vallecitos rods, reported in BNWL-45, made available the first experimental data considering such isotopes as Np^{237} , Pu^{238} , Am^{241} , Cm^{242} , and Cm^{244} . These data are compared with MELEAGER results in Tables I and II. The Vallecitos rod data were taken at three different points along a single rod, whereas the Yankee rod data involved separate rods--two of which were in the center of an assembly and two were near a water gap. The exact neutron spectrum was not known for any of these cases, so the procedure chosen was (1) to stop burnup at the U^{235} depletion shown in chemical analysis, and (2) the fuel-to-moderator ratio (water density) was adjusted to produce the proper concentration of plutonium by using the resonance absorption in U^{238} as a spectrum indicator. The fuel exposures agreed within the error limits for the exposures determined by cesium analysis. The plutonium composition agreed with chemical analysis within 1%. Of higher isotopes (Am^{241} , Cm^{242} , and Cm^{244}) Cm^{244} showed the best agreement, while Am^{241} and Cm^{242} --although not agreeing as well--are strongly a function of the power during the burnup history which is not well known. The Np^{237} production is in agreement when the U^{236} concentration is in agreement. However, the U^{236} concentration calculated in this study as well as by Westinghouse for the same samples shows that the U^{236} should be higher for Samples Y-2 and Y-1. For these cases, the calculated Np^{237} concentrations are also too high. The Np^{237} production of the Vallecitos rod was low at high burnups and high at low burnups. However, the concentrations were low, which may lead to large error limits.

MONTE CARLO Calculations

The Monte Carlo computations regarding the 0.6-inch-diameter metal rod in a lattice having a metal-to-water ratio of 1.0, as reported by Kouts and Sher in BNL-486, were extended to two or three hours of MONTE CARLO time. No trends from the one-hour values of thermal utilization, eta, epsilon,

TABLE I

COMPARISON OF CHEMICAL ANALYSIS OF YALKEE RODS TO MELEAGER BURNUP RESULTS

Sample (mg)	Y-2	MELEAGER	Y-2	MELEAGER	Y-3	MELEAGER	Y-1	MELEAGER
Burnup, LWR/K.T.	13,700 ⁽¹⁾	13,200	17,700 ⁽¹⁾	16,000	24,300 ⁽¹⁾	22,400	31,800 ⁽¹⁾	31,000
Concentration, grams/K.T.								
U ²³⁵	21,800	22,003	19,200	19,795	16,100	16,561	11,100	11,175
U ²³⁶	2,530	2,546	2,230	2,678	3,790	3,577	3,690	4,049
Re ²³⁷	242	218	205	233	495	452	475	542
Pu ²³⁸	33	24	32	26	90	81	113	117
Pu ²³⁹	5,740	5,637	5,620	5,517	7,400	7,380	6,350	6,306
Pu ²⁴⁰	927	977	1,240	1,183	1,700	1,840	2,160	2,307
Pu ²⁴¹	468	473	599	556	1,110	1,156	1,300	1,280
Pu ²⁴²	51	53	96	85	200	200	390	334
Am ²⁴¹	69	82	85	93	119	122	139	141
Am ²⁴³	---	5.8	---	8.9	---	39	---	81
Ce ²⁴²	0.03	0.03	0.03	0.04	1.0	1.2	1.6	2.0
Ce ²⁴⁴	0.55	0.55	0.89	0.90	7.9	7.6	16	17

(1) by cesium analysis

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TABLE 11

Concentration of various elements in the soil of the study area

Element	Area	Concentration, mg/kg	Area	Concentration, mg/kg	Area	Concentration, mg/kg	Area	Concentration, mg/kg
	V-5	V-5	V-5	V-5	V-5	V-5	V-5	V-5
	16,010 mg/kg	18,432 mg/kg	25,000.0 mg/kg	24,500.0 mg/kg	28,500.0 mg/kg	28,077.0 mg/kg		
	2,820.0	2,786	1,830.0	1,167.0	1,030.0	1,167.0		
	140.0	123	47.0	63.0	27.0	32.0		
	14.5	17	5.1	4.1	1.1	1.2		
	3,330.0	3,330	2,370.0	2,771.0	1,561.0	2,002.0		
	83.0	830	420.0	420.0	132.0	201.0		
	32.0	277	121.0	55.0	35.0	35.0		
	57.0	209	7.6	8.6	1.2	1.8		
	45.0	43	15	14.2	5.6	4.2		
	0.37	0.29	0.066	0.07	0.18	0.01		
	0.48	0.12	0.72	0.01	0.032	0.00		

mg/kg

TABLE 12

mg/kg

or n-2n effects were noted; although a better understanding of the precision of the calculation was obtained. These factors are given in Table III. The limits given are not error limits, but are indicative of the variation in the computed values after the first hour of the calculation. During the first hour, the variations were more or less inversely proportional to the time spent. One should also note the rather substantial bias of the zone-weighted calculation. This may be due to relative magnitude of the weighting factor (5.0) and the average attenuation in the rod (about 0.8 at high energy and 0.5 at thermal energy). Thus, the probability that the weighted particles would produce added detail was small and the repetitive calculations only served to waste time. The statistical selection of which one of the five particles to follow from the fuel into the moderator seems to have reduced the fast flux interaction between rods even though the selected particle will reappear as five particles if it successfully traverses the moderator.

The resonance escape probability is presented in Figure 1 as a function of computation time. The values plotted represent escape from the total resonance absorption of U^{238} while the value of 0.72 ± 0.027 reported by Kouts and Sher excluded the $1/v$ absorptions. Note that the results seem to be well converged in 40 or 50 minutes, but the subsequent divergence is not explained.

Fuel Utilization Studies

Several revisions have been made to the VESTA code to improve its accuracy and to shorten the running time. An audit of the code has also been completed. Recalculation of the optimum operation of the PWR and the HWR has been completed. This reoptimization includes operation with improved specific powers, lower jacketing and separations costs, together with variations in plutonium value and in the alpha of Pu^{239} . Work is beginning on calculation of the same information for the SGR. As soon as the same data are available for all the 1000 MW fast reactor reference designs and the sodium-graphite, D_2O -organic, and high temperature gas-cooled reactors, all of these reactors will be evaluated for their uranium conservation properties using the VESTA code, which is essentially complete for the handling of uranium-plutonium systems. Some work needs to be done yet on the code to assure that it can properly handle the introduction of breeder reactors when no plutonium availability limitations on initial fast reactor inventories are to be applied.

Linear Programming Studies

The preparation of the linear programming tools which will be needed in our evaluation of the conservation problems of the nuclear power industry

TABLE III

MONTÉ CARLO CALCULATIONS FOR WATER-TO-HEAT RATIO OF 1.0

Run Characteristics	Thermal Utilization	Sta	Fast Effect	n-2n Effect
Experiment (BNL-486)	.946		1.099	(Part of fast effect)
Uniform weights	.9429 $\pm$.0001	1.601 $\pm$.0001	1.0325 $\pm$.001	1.0018 $\pm$.0001
Uniform weights	.9427 $\pm$.0001	1.606 $\pm$.0001	1.031 $\pm$.001	1.0021 $\pm$.0001
Broad group and fast weights	.9427 $\pm$.0002	1.603 $\pm$.0001	1.0784 $\pm$.0005	1.0018 $\pm$.0001
Zone weights	.9420 $\pm$.0003	1.610 $\pm$.0001	1.058 $\pm$.002	1.0011 $\pm$.0001

MONTE CARLO CALCULATION OF RESONANCE ESCAPE PROBABILITY

FIGURE 1

is continuing. A sample problem which consists of one reactor having a commitment for power and plutonium production and a choice of operating methods has been prepared and has been run on the 7090. We have been using it to learn the use of an output analyzer routine which translates the problem solution into a form suitable for report generation. Some of the options available on this output analyzer seem to be working properly. Some difficulty is being experienced in performing certain types of matrix multiplication which this analyzer is supposedly able to do. Another sample problem which came with the analyzer routine has been run; but, to date, not successfully. This difficulty is traceable to the difference between our method of operating the computer and that at the site where the program originated. It should be cleared up within the next few days.

Inclusion of $\sigma(n,2n)$ into MELEAGER

To include the $\sigma(n,2n)$ into MELEAGER, it is necessary to adjust the value of the $\sigma(n,2n)$ so that it becomes the suitable average cross section for the spectrum of current interest. This can be done as follows:

The fission spectrum averaged cross sections are assumed to be proportional to those averaged over some other fast spectrum. One would thus use

$$\sigma_{n,2n}^f = \frac{\sigma_{f.s.}(n,2n)}{\sigma_{f.s.}(f.f.)} \sigma_{f.f.}^f$$

where

$\sigma_{f.s.}(f.f.)$ is the fission spectrum average fast fission cross section

$\sigma_{f.f.}^f$ is the average fast fission cross for the desired spectrum

$\sigma_{n,2n}^f$ is the average $(n,2n)$ cross section for the desired spectrum

This approximation is valid in the perturbation sense; that is, for approximately equal thresholds and moderate spectrum changes.

The fission spectrum averaged fast fission cross section for U^{238} , $\sigma_{f.s.}(f.f.)$, is 307 mb.

Investigation of (n,2n) Cross SectionsElements with $Z > 90$

Known experimental data for n,2n cross sections in this range of elements are limited to the following elements: Th^{232*} and U^{238*} . It is possible that some data exist also for U^{235**} and Pu^{239**} .

For other isotopes, the cross section may be approximately calculated using the method suggested by Pearlstein.* For all fissionable isotopes, this procedure will lead to a double uncertainty; that is, both the uncertainty in the fast fission cross section and in the total reaction cross section must be added for the uncertainty in the (n,2n) cross section. This situation is then compounded by the fact that the fission cross section is usually a large part of the total reactor cross section. Thus, the $\sigma(n,2n)$ is the small difference, and the relative uncertainties are greatly magnified.

Although a number of the desired cross sections have already been calculated by Pearlstein, it was necessary to calculate others. This task was greatly simplified by means of an approximate empirical correlation based upon the available cross sections from Pearlstein's paper.

Inspection of Pearlstein's results indicates very slight cross section shape change in the region of interest. The principal variable quantity which affects the cross section being its plateau or peak value, σ_{ne}' , and the principal factor which affects the spectrum averaging is the threshold. Thus, the ratio of the fission spectrum averaged cross section $\sigma_{f.s.}(n,2n)$ to σ_{ne}' should be approximately only a function of the threshold.

The data for this comparison, taken from both Pearlstein and this study, are given in Table IV and are plotted in Figure 2. Considering the approximate nature of the available cross section information, the roughly 5% scattered about the curve is easily tolerated. By means of the threshold and σ_{ne}' , the fission spectrum average cross section is obtained for other elements in this range of Z from Figure 2. These are summarized in Table V along with the long term (n,2n) reaction products.

* S. Pearlstein, Analysis of (n,2n) Cross Sections for Nuclei of Mass $A > 30$, BNL-897, December 1964.

** J. J. Schmidt, Neutron Cross Sections for Fast Reactor Materials, p. III graphs, p. II tables, KFK-120.

TABLE IV
AVERAGE (n,2n) CROSS SECTION CORRELATION
(from Pearlstein)

Isotope	Threshold σ_{nc}		$\sigma_{f.s.} (n,2n)$ mb	$\sigma_{f.s.} (n,2n) / \sigma_{nc}$ $\times 10^3$
	MeV	mb		
Ra ²⁶⁶	6.36	2,780	18.0	6.47
Th ²³²	6.34	2,320	16.0	6.90
U ²³³	5.90	330	0.33*	1.00
U ²³⁴	6.77	1,430	7.2	5.03
U ²³⁵	5.24	840	15.0	17.86
U ²³⁶	6.40	940	6.5	6.92
U ²³⁸	6.04	1,650	15.0	9.09
Np ²³⁷	6.76	250	1.3	5.20
Pu ²³⁹	5.62	160	0.2*	1.25
<u>Additional calculations and corrections</u>				
Pu ²³⁹	5.62	370	5.0**	13.51
Pu ²⁴⁰	6.21	880	7.4	8.41
Am ²⁴¹	7.19	132	0.5	3.7

* $\sigma_{f.s.}$ from Corvini
** Corrected data

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CORVINO-6 4

CORRELATION OF FISSION SPECTRUM AVERAGED CROSS SECTIONS
WITH THRESHOLD AND σ_{ne}

FIGURE 2

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mp	mp	mp
Product	Product	Product
Pu-231	13.0	Pu-231
U-232	21.5	U-232
U-233	3.3	U-233
U-234	7.2	U-234
U-235	15.0	U-235
U-236	6.5	U-236
U-237	15.0	U-237
U-238	1.3	U-238
U-239	2.6	U-239
U-240	5.0	U-240
U-241	3.5	U-241
U-242	7.9	U-242
U-243	7.4	U-243
U-244	18.5	U-244
U-245	18.5	U-245

TABLE A

TABLE A

Bismuth

The experimental (n,2n) cross section at 14 MeV is available in the second edition of BNL-325.

$$\sigma(n,2n)_{14 \text{ MeV}} = 2400 \pm 150 \text{ mb}$$

The calculated value of Pearlstein at 14 MeV is 2490 mb which agrees well with this result. Pearlstein's fission spectrum averaged value is 6.2 mb.

Beryllium

Considerable experimental data are available for the beryllium (n,2n) cross section in the second edition of BNL-325. Averaging these data over a fission spectrum yields a value of 113 mb.

The range of applicability of the fission spectrum averaged beryllium cross section is more limited than that of the heavy element fission spectrum averaged cross sections since beryllium is a moderator. This cross section should be used only for dilute intimate mixtures of beryllium and the fuel.

CODE DEVELOPMENTJASON Code

Changes were made to the JASON code to enable it to calculate more accurately input parameters for MELEAGER when targets and products are being considered. Since the flux in MELEAGER is based on the unit fuel volume, the target material must be included with the fuel when the value of SNF (macroscopic nonfuel cross section) and the volume weighted isotopic fuel densities are calculated. By including the target with the fuel during the SNF calculation, the value of SNF does not reflect the target absorption. Thus, parasitic absorption of neutrons is maintained at the proper level. These revisions have been made and the code is now in use.

To enable the density optimization to be more automatic and more accurate, JASON has also been revised to read input parameters out of common after they have been installed there by the OPTIMIZER code. This change results in a more accurate calculation of the fast effect, and it will also produce fitted values of the flux depression factors, FCONST and FEXP.

Part of these changes have been made and debugging of the code is in progress.

ALTHAEA Code Development

Preliminary layout of a four-group diffusion code version of ALTHAEA for one-dimensional burnup of fast or thermal reactors has been completed. The code will handle up to 24 regions with up to 20 fuel- 15 fission product- and 10 economic pseudo-isotopes which may have different concentrations in each region. The user may specify a fourth order prescription for absorption cross section variation in the thermal and epithermal groups if used as a thermal reactor code or microscopic transfer and transport cross sections if used as a fast reactor code. The same basic cross section constants must apply in all regions.

SEPO Code

Changes to the separations cost code, SEPO, have been made to simulate the Nuclear Fuel Services plant schedule. This model will be utilized in the fuel density study which is currently in the beginning stages.

A new Fuel Cycle Analysis Chain Tape is being prepared which will be used for shipment to people requesting our tape. This master tape will include the entire fuel cycle analysis chain, with the exception of CURVES and HESTIA. The current version will include JASON CASE GENERATOR, JASON SUPERCASE GENERATOR, MELEAGER CASE GENERATOR, MELEAGER, PROTETUS, QUICK, OPTIMIZER, MULTIREGION EDITOR, PREP, and PLOTTER. Previous fuel cycle analysis tapes that have been used to supply requests have contained only a limited number of the available codes.

NEUTRON FLUX MONITOR PROGRAM

The two regenerating ion chambers, which had initial coatings of approximately 90% U^{234} and 10% U^{235} , remain in operation in the KW Reactor test facility. Due to progressive degeneration of the chamber's insulation, voltage is being applied only when data are being obtained on an intermittent basis. The total exposure received to date is about 2×10^{21} n cm⁻² or about 40% of the anticipated "useful lifetime." Analysis of data is extremely difficult because of insulation deterioration; however, it has been firmly established that regenerative action is being successfully achieved since the measured reduction in sensitivity is minor compared to that expected (for the same exposure) for the regular, fully-enriched U^{235} chambers. The buildup of fission products in the chambers has not become a problem, since the dynamic range has remained essentially unchanged. Two more chambers will be fabricated for further testing.

An improved design of a B¹¹ beta-current detector assembly was conceived to improve the efficiency over the present, parallel-plate concept and to obviate the need for precision assembly of specific components to cancel background currents. The new design incorporates an electrical guard and a cylindrical geometry to improve beta collection, a thick attenuator to reduce background effects, and a false (dummy) element to provide background cancellation. Twin-wire, magnesium-oxide cables will be employed with the detector. Fabrication effort is being accelerated.

In addition, a further analysis was performed on the data from the latest cable irradiations at the KW Reactor facility. It appears that manganese becomes the primary source of transient signals once aluminum is removed from the test assembly. The use of magnesium oxide insulation demonstrated a marked reduction of background noise over aluminum oxide insulation; however, the reasons are not understood.

For planned microwave cross-coupling (attenuation) tests at KW Reactor, a gas tube was fabricated and attempts were made to determine the effects on cross-coupling between two waveguides. The achieved plasma densities were such that complete cutoff of energy coupling occurred, indicating the plasma density was too great for the size of gas tube employed. Additional gas tubes are being fabricated for further tests.

In preparation for planned high temperature work, quantities of niobium and tantalum (refractory metals) are being obtained. Experimental cavities and other components will be fabricated from the materials as soon as possible.

DETECTION OF COOLANT IMPURITIES AND HIGH TEMPERATURE MEASUREMENT PROGRAM

Laboratory experiments were conducted to determine the feasibility of measuring minute amounts of water vapor in normal air. Air of varying humidity was passed through a microwave resonant cavity and the degree of detuning was observed. Temperature changes of the air, prior to cavity injection, were troublesome; however, it was possible to measure the relative humidity within a few percent. Work is being accelerated to determine the limits imposed and the specific frequency changes which occur. Two gas-sampling chambers were designed and fabricated to permit experiments regarding the microwave attenuation and phase shift introduced by air of different relative humidity levels.

Specifications were prepared and a requisition submitted for acquiring an infrared analyzer for use in program investigations. In addition, components have been ordered for assembly of a laboratory test loop.

NONDESTRUCTIVE TESTING RESEARCH PROGRAMElectromagnetic Testing

Vector voltage locus plots of the multiparameter eddy current test signals were recorded from ten drilled holes of various depths and diameters in a stainless steel tube. These tests show that the signal amplitude was a nonlinear function of the hole dimension for hole diameter in excess of 0.060 in. and the hole depth which exceeds 10 percent of the wall thickness. A reduction in the ratio of the two test frequencies, application of non-linear techniques, and changes in the test coil assembly are being considered for corrective measures.

Improvements are being incorporated in a modified Model 1004 eddy current tubing tester to be designated Model 1004A. The major changes are the use of a synchro-resolver to provide 360° phase rotation in place of the previous 90° circuit and the use of a new balanced detector circuit which eliminates the need for an opposing battery in the dc zero adjust circuit. Other improvements are being made in the parts layout and convenience of operation.

Fundamental Ultrasonic Studies

Studies are being conducted to determine if pulse phase velocity changes due to attenuation can be predicted with the previously developed wave model. It is important that this phase shift be known when measuring the velocity of wave propagation in all materials since attenuation causes apparent delay in the pulse transmit time. This results in

velocity measurements which are in error unless the amount of delay is known and corrections are made. Current attempts to evaluate this delay have been hampered by the mathematical complexity of the problem.

Pulse shape recordings were taken for a variety of experimental conditions not previously considered, and these data were compared to pulse shapes predicted by the theoretical attenuation model. Particular care was taken to assess and correct for gain variations during the experiments. The calculated pulse amplitudes did not agree too favorably with the measured results, but pulse shapes appeared to be in better agreement than in previous experiments.

Heat Transfer Testing

A non-contacting thermal wave transducer has been successfully used to differentiate samples of different materials. The transducer used an electrically heated stainless steel foil heat source coupled to the samples through a 0.010 in. thick air gap. Foil temperature and power input to the transducer were plotted as the ordinate and abscissa of Lissajou figures during tests. Large volume samples of air, teflon, stainless steel, aluminum, and lead were tested. No two different materials or combinations of different materials gave the same temperature phase and amplitude at the transducer.

The thermal expansion problem previously encountered with the stainless steel foil was reduced by using Nilvar foil which has negligible thermal expansion over a 100°C range. The Nilvar foil is capable of handling peak power inputs of at least 28 watts/in² without buckling or sagging during extended tests. Wave shapes were found to be more nearly sinusoidal, and signal to noise ratios are better because of the larger amplitudes attainable.

Thermal testing techniques developed on this program were used to test several Zircaloy-2 clad UO₂ fuel rods. The rods were scanned with a low power heat source and an infrared radiometer. Internal air gaps between the ends of the fuel and the end caps of the cladding were detected as temperature increases on the outer surface. In addition, a slight increase in surface temperature was detected approximately two feet from one end of the fuel rods and is believed to indicate a decrease in fuel density at that point.

Quantum Physics Device Research

A research program to investigate the possibility of obtaining high energy ultrasonic pulses by the interaction of pulsed laser light energy

with a material medium is being conducted. It is anticipated that such an ultrasonic pulse will have application in solving a variety of future nondestructive testing problems.

Present investigations have been concerned with developing a broadband detection method for analyzing these ultrasonic pulses. Optical methods, using both Schlieren and interferometric techniques, are being tried. In the experiments to date, a conventional ultrasonic transducer has been used as a source of pulsed ultrasonic energy. However, the two optical methods currently being used are interrelated and the physical phenomena being measured are present in both. These interactions are complex and further investigation will be necessary to determine the properties of each that will give an accurate picture of the ultrasonic wave.

Results of this study to date have shown that a 10 Mc ultrasonic pulse in water may be detected by this method. In this case, the detection of the beat frequency between two or more of the Doppler shifted diffraction orders was used to obtain the pulse information. The continuous wave laser was used as the light source for these experiments, and a photomultiplier was used to convert the light intensity to a recordable electrical signal.

Experiments on methods of utilizing radiation damage in piezoelectric semiconductors to produce a hypersonic transducer were continued. Single crystals of cadmium sulfide, approximately 5 x 5 x 1 mm, were irradiated with 2 MeV electrons and 1.8 MeV protons in efforts to damage the lattice and thus produce an n or p-type semiconductor from this normally insulating material. Integrated doses of 10^{18} electrons/cm² and 10^{15} protons/cm² were used.

Electrostatic Transducer Development

In the continuing development and evaluation of the electrostatic transducer, its response to narrowband frequency signals was evaluated. The ability of this transducer to generate and detect narrowband ultrasonic pulses over the frequency range from one to twenty-five megacycles was shown.

For these tests, electrostatic transducers were placed on each end of a 3/3 in. diameter by 1-1/2 in. long aluminum rod. One transducer was used to both generate and detect ultrasound by pulse-echo operation. The other transducer was used only for detection of the transmitted sound.

Narrowband transmit pulses from an Arenberg pulse oscillator were used to drive the transmit transducer. The center frequencies of the narrowband transmit pulses were varied from one to twenty-five megacycles while the voltage changes induced in the detection transducer were amplified and displayed on an oscilloscope simultaneously with the transmit pulse. The envelope shape of the detected pulse was then compared with the envelope shape of the transmit pulse.

A prototype of the ultrasonic test station for nondestructive inspection of Isotope Heat Sources was fabricated and preliminary evaluation tests conducted. This equipment includes a mechanical system capable of remote operation, specially designed ultrasonic transducers, and electronic circuitry capable of accurate tests with high resolution.

The prototype equipment is currently being used to further evaluate the proposed ultrasonic tests. Information on the shear and longitudinal wave velocities in the heat source material and the transmission characteristics of kilocycle frequency signals through the core material is needed. This information is needed to accurately interpret test signals for a variety of heat source conditions.

Discs of lead zinc titanate were obtained for evaluation and use in the nondestructive measurement of plutonium properties. Tests showed that much of this piezoelectric material gave poor beam symmetry and large beam amplitude variations. Results indicate that pre-characterization of beam symmetry, amplitude, and beam ringing must be carried out to ensure transducers which will provide reproducible test results.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

Additional measurements of the turbulent fluctuations in wind velocity in the lower atmosphere were obtained near the Meteorology Tower, utilizing the Portable Mast and the Wind Component Meter. Data were collected at heights of 3, 6 and 12 meters during neutral atmospheric density conditions and at 12 meters height during unstable conditions, providing a significant addition to the stable condition data collected earlier. Analog computer analysis for filtered power spectral density estimates and cospectral estimates was completed for the stable atmospheric density condition data in the frequency range 0.01 to 4 cycles per second. Investigation of the filter response characteristics in the spectral program provided correction factors for each filter to account for deviations from the design response. The analog program for obtaining the spectral density tensor components through the Fourier Transform was

completed and appears to be providing good results.

A draft of the section on calculation of precipitation scavenging of pollutants in the atmosphere for inclusion in the second edition of Meteorology and Atomic Energy was completed and forwarded to reviewers for comment. The contribution is intended to provide an up-to-date review of information pertinent to quantitative prediction of washout parameters. Collection efficiencies, washout coefficients for particles and gases, rain spectra and washout by snow are discussed and values given, some of which have not previously been published. Extensive "area within a washout isopleth" charts are shown for three atmospheric stabilities and several washout coefficients. A paper titled "Snow and Rain Washout Coefficients for Inorganic Iodine Vapor," by R. J. Engelmann and R. W. Perkins, was submitted to Science for publication.

Radiological Physics

A calibration of the plutonium counter for lung deposited plutonium in humans is now being made. The plutonium is mixed with a lung equivalent plastic and inserted into the lung cavities of an Alderson phantom. The calibration was completed for dogs. Different isotopic compositions and different distributions in a dog phantom were studied. These calibrations agreed with those obtained from counting dogs which were later sacrificed and analyzed. Several dogs containing plutonium were counted for the inhalation toxicology laboratory.

In order to accurately determine body burdens of plutonium by X-ray counting, the number of X-rays per disintegration in the isotopic mixture of plutonium must be determined. An X-ray alpha coincidence counter was constructed and initial tests indicate that this method will be useful for determining the X-ray per alpha particle ratio.

Another trip to Alaska for whole body counting of natives was completed. All the Eskimos at Anaktuvuk Pass were counted. The Cs-137 burdens were about the same as they were in January, which is about one-half what they were last summer. The final calibration of the simplified whole body counting method was made at Anaktuvuk and this method was used at two other villages, Arctic Village and Ambler. The Cs-137 body burdens at Arctic Village at the present time are higher than at Anaktuvuk, but they have their maximum body burdens now and are starting to decrease whereas the Anaktuvuk people will reach their maximum this summer and their maximum will be higher than the maximum at Arctic Village. At Ambler the man who had the highest Cs-137 content we have ever measured (3000 nCi) was measured again and he had dropped to 1736 nCi. However,

he still had the highest body burden in the village. With this simplified method we were able to go into these two villages, count about 20 people, and return within a few hours for each village. By using our past methods, it would have taken about a week at each village plus much higher transportation costs to obtain the same information.

Some initial studies on the background reduction for whole body counting by use of anticoincidence shields and multidimensional analysis have been made. These studies indicate that significant increases in sensitivity can be obtained for certain isotopes in the body.

The positive ion accelerator performed satisfactorily all month with no unscheduled down time. Prof. L. W. Seagondollar, University of Kansas and consultant to Radiological Physics, spent a week working with us in further attempts to separate helium ions for use in the Van de Graaff accelerator. A combination of an Einzel lens and a gap lens was found to provide a better focus of the beam. This reduced the amount of molecular hydrogen produced, which has the same $\frac{e}{m}$ as the doubly charged

helium ions, and allows better separation of the helium ions from the molecular hydrogen.

The optical discrimination between neutron and gamma ray induced pulses in a stilbene crystal was tested by using protons from the Van de Graaff accelerator. This method of discrimination uses the difference in path length of the recoil protons produced by neutrons and the recoil electrons produced by gamma rays. Because of the shorter path length of the protons, the light formed will undergo more self-absorption in the crystal. The experiment with the accelerator showed the expected absorption to occur, but proton energies below 1.1 MeV could not be resolved. Better light coupling, light pipes, and low noise phototubes and preamplifiers should allow lower energy protons to be resolved.

Initial efforts to observe exoelectron emission from alpha particle bombardment of CaSO_4 have not been successful. The emission is masked by increased background in the counter as the temperature of the sample is increased.

Instrumentation and System Studies

Each of the four bench-model channels (blood pressure, respiration rate, temperature, and thyroid radionuclide uptake) of the experimental animal physiological function and radionuclide uptake telemetry system is now performing correctly when operated separately. The next step is to secure simultaneous operation of all four segments without cross-channel

interference. For the receiver portion of the system, a different demodulator circuit, based on ratio detection methods, was developed and has demonstrated improved performance. This will be incorporated in the information-recovery circuitry. The thyroid and temperature telemetry channels are now being fabricated in miniature form.

Circuit refinements, using insulated-gate field effect transistors, were incorporated in the solid state electrometer portion of the experimental raindrop charge and velocity measuring instrument. The charge measuring circuit is now essentially complete. Discussions are being held with Atmospheric Physics personnel to establish both the velocity measuring requirements and the general configuration of the detector assembly.

Experimental investigations were started with both germanium and silicon solid state detectors of the lithium-drifted type. Liquid nitrogen is being employed for cooling the detectors to improve the noise characteristics. Difficulties are being encountered in obtaining a satisfactory seal on the vacuum chamber and new components are being ordered. Gamma spectral data were obtained for Co^{60} and Cs^{137} sources using the silicon detector and a multichannel analyzer; however, the vacuum and cooling difficulties encountered prevented valid results. The Compton edges could be observed, but the photoelectric peak information was not satisfactory. The improved vacuum system and better cooling should materially enhance the performance.

The solid state circuitry for the experimental mixed neutron-gamma dose rate measuring instrument was partially completed for the neutron channel, and the polyethylene moderator for the Li^6 solid state neutron detector was received. An air-equivalent wall ionization chamber and all solid state circuitry will be used for the gamma dose rate measuring channel.

Fabrication of a dual-cylinder ionization chamber, which will be employed in a rodent whole-body counting system, was nearly completed. Circuitry development work was initiated.

Investigations were continued into methods for measuring the activity of small, caged rodents, and several different sensors (infrared and light-activated diodes) were ordered for experimental work. It is certainly apparent that a relatively complex system will be required to effect a solution to the problem.

All of the necessary circuits for the experimental algae radionuclide uptake measuring instrument have now been designed in final form,

following successful bench-model tests. Layout drawings are being made to permit fabrication of the instrument.

Design and fabrication were completed on a respiration register instrument to be used at the Biology Inhalation Toxicology facility during inhalation experiments. The instrumentation, which uses a thermistor detector inserted in the inhalation tube of the canine, provides a one-digit advance on the register for each inspiration.

Engineering assistance was rendered to Atmospheric Physics regarding procurement of a four-channel real-time airborne particulate measurement system. The requirements were established for the complete system and further assistance is planned during preparation of detailed specifications.

Comparisons of the light-coupling properties of Dow-Corning "Sylgard 184" (a clear epoxy resin) with regular optical coupling grease demonstrated the superiority of the resin. A 10% to 15% improvement in signal amplitude output from a multiplier phototube, used with an NaI(Tl) crystal, was noted during the tests. The resin appears to have considerable application in scintillation detector work.

Special, low-noise, junction field effect transistors (just obtained) have demonstrated a factor of 2 to 4 improvement in signal-to-noise ratio over regular junction and insulated-gate field effect devices. Thus, it appears the new units, although quite expensive, can provide improved performance for low-noise, charge-sensitive amplifiers.

The power spectrum and cross spectrum analyses for the Meteorology wind turbulence program were started. Approximately three hours of new wind data were recorded this month.

A study of P-32 effects was made using the biological system radio-isotope model for standard man. Preparation of a MIDAS program for this model was started.

WASHINGTON DESIGNATED PROGRAM

Isotopic analyses were provided on only a minimal number of program samples due to the continued need to study the ion-optical properties of the new ion source to insure valid and reliable operation in the mass spectrometer.

A paper describing the new vacuum-lock sample changer was prepared for presentation at the ASTM Committee E-14 Conference on Mass Spectrometry in May 1965.

The design of a vacuum lock for the new mass spectrometer was completed and progress was made of procurement of components for this instrument.

EXPERIMENTAL REACTOR PHYSICS FACILITIESPRCF

Experiments were completed on the Pu-Al D₂O core.

Reactor noise and step reactivity change (boron arrow) measurements were conducted at three different chamber arrangements: Center of core, top template, and around the fuel perimeter. Rod drop measurements were made for individual control and safety rods and for control and safety rod groups. Two irradiations were performed using cadmium covered Lu foils and Au foils. Reactivity measurements were made in an aluminum thimble which was voided, filled with 99.51% D₂O, 71.95% D₂O, and demineralized H₂O. An additional moderator level sensitivity point was obtained at 89.3 inches about 7.6 inches below the top of the fuel meat. The core was discharged and UO₂ elements were placed in storage in preparation for the following two-zone loading.

The reactor was deactivated for 8 days to permit ventilation work in connection with project CAH-119. Outage efforts were directed at maintenance, safety rod inspections, housekeeping, and experimental equipment development.

PCTR Operation

Operation of the PCTR continued routinely during the month. There was a one-day maintenance outage to correct a slipping clutch in the low speed face drive.

The 19-element cluster, PuO₂-UO₂ fuel elements in a 6-1/2 inch pitch graphite lattice experiment was completed.

The Phoenix fuel experiment was resumed after the number of borated polyethylene disks used in the Pu-Al fuel assemblies was changed.

TTR Operation

The TTR was operated once during the month to irradiate a hafnium rod to determine the hafnium density along the rod.

Critical Approach Facility

Three lattice (2-3/4 inch, 3-1/8 inch, and 3-3/8 inch pitch) using 1.96% enriched NPR fuel elements with and without targets were measured for the critical loading determination.

The three lattices were modified to take two control rods and four safety rods before the measurements were started.

Development Items

The design of the 10-1/2 inch graphite lattice was modified to permit use of more flux modifying columns in order to gain a good match of the neutron spectrum.

A transistorized logarithmic count rate meter with period and high and low level trip circuits has been designed and built and has been under operational test for the last month. Blueprints are being prepared by Facilities Engineering Drafting Unit. After completion of the prints, the design will be reviewed by others and considered for use in the PCTR control circuit.

A double dump valve assembly for the CAF has been designed by Maintenance and Equipment Engineering, TRES, and the Design Change has been approved. This will provide a backup system with resulting increased reliability.

Studies have been requested and initiated by MEES on a system to allow partial draining of the CAF tank and on a method to allow faster opening of the PCTR movable face. Partial draining and return of the temperature equilibrium water will provide remote reactivity control for those experiments in which the control and safety rods must be removed. Quicker opening of the PCTR movable face will save ten minutes per loading cycle; during an approach to critical procedure, this can amount to one "free" cycle per day.

COMPUTER FACILITIES

Purchase of the currently leased G. E. 412 computer upon availability of funding is now officially approved.

Dataphone transmitter and receiver stations are now on order with Western Electric. These units will be used to relay nondestructive test signals to the 412 computer for analysis, storage, and readout.

A new program was written for pre-editing PRTR data tapes. Badly punched data may now be deleted or corrected at the computer console. The last six weeks' tapes are now being run.

The new drum memory to increase the 412 capacity from 16,000 to 32,000 words was placed on order with G.E. The additional memory capacity is needed to accommodate the large number of utility and process control programs being developed.

Analog computer utilization was as follows:

<u>EASE 1132</u>	<u>EASE 2133</u>	
352	336	Hours Up Time
0	16	Hours Scheduled Down Time
<u>0</u>	<u>0</u>	Hours Unscheduled Down Time
352	352	Hours Total

Both computers were utilized on day shift and swing shift.

Problems considered during the month:

1. Containment Systems Experiment.
2. FFTF Core Design.
3. N Reactor Conversion Study.
4. Meteorology Study.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Meteorological services, viz., weather forecasts and observations, and climatological services were provided to plant operations and management on a routine basis. Consultation was provided to 1) NRD, IPD and AEC on reactor confinement concepts and the utility of tall stacks, including attendance at a meeting with the Division of Production in Washington, D. C., on April 14 and 15, 1965, 2) HUP0 on 300 Area stack release problems, and 3) HUP0 on weather conditions occurring at the time of a fatal accident.

The Mid-April forecast for the Columbia River was prepared and distributed. The forecast is for a lower crest this year than last and for river temperatures during the period July-September to average below normal, but still slightly higher than last year.

Weather Summary

<u>Type of Forecast</u>	<u>Number Made</u>	<u>% Reliability</u>
8-Hour Production	90	86.1
24-Hour General	60	85.3
Special	115	90.4

April was warmer and drier than usual. The earthquake of the 29th was felt by most personnel in the building at the time.

Mass Spectrometry

Isotopic analyses were provided on 6 uranium samples in support of Purex U²³³-production studies.

Instrumentation and Systems Studies

Recommendations were made to Field Design and Chemical-Metallurgy regarding instrumentation for the gas loop of the Advanced Test Reactor. A four channel contamination monitoring system was proposed.

Further detector probe refinements are being incorporated in the assault mask beta-gamma contamination monitor, which was developed for use at the Hanford Laundry facility. The modified probe configuration should enhance the sensitivity uniformity over the outer surfaces of the masks.

An improved scintillation well-logging probe for underground gamma emitters and a new core-sample radionuclide counting head assembly were designed and fabricated for use by Geophysical and Geochemical Research of Chemical Effluents Technology. Both instruments performed satisfactorily in laboratory tests and the instruments were delivered for general field testing.

The physiological function telemetry work was extended to include development of a specific system for human measurements for Industrial Hygiene. The system being designed includes both temperature and EKG signal channels, and it will be used with humans working in specific high temperature locations at Hanford. At present, the system, which includes some commercial portions, provides only non-simultaneous measurement for the two channels; however, the circuits are being modified to permit constant surveillance of both channels and to achieve less drift in the temperature channel.

Detailed programming for the HTLTR instrument system is in progress.

Continued evaluation of several types of multiplexer relay for the Containment Systems Experiment has shown a single dry reed unit to offer the best performance. The use of this relay together with solid state switches in a two-stage multiplexer configuration will provide a signal scanning rate in excess of two thousand points per second. Electronic components for assembly of the multiplexer were ordered along with interface units for connecting multiplexer, high-speed punch, and multi-channel analyzers to the computer.

The planned mass spectrometer instrumentation was changed to include a magnetic tape memory to permit high-speed program loading and data storage.

The control track amplifiers in the 6144-channel time-of-flight analyzer were found to be working marginally. Circuit changes and the addition of proper terminating loads corrected the trouble.

The containment system analog simulation was prepared for the Chemical Laboratory. A workable MIDAS program was furnished from which an analog simulation was prepared and debugged. Further effort on the program is awaiting additional data from the customer.

Development work continued on a method of detecting crack propagation in a containment vessel. The general method for the detection of operational crack growth in metals utilizes the ultrasonic energy released. Limited testing and evaluation was completed on various transducers and a tensile test was made using boiler plate steel from the outer containment vessel in the 231-T Building. The test assisted in selecting instrumentation and transducers for use in the containment vessel proof test. Additional tensile tests are required for determining transducer characteristics.

A sonic test was made on the large containment vessel in the 231-T Building, 200-W Area. The vessel was gradually pressurized to 1-1/4 times the design pressure over a period of six hours and ten minutes. During the pressurization and at times of constant pressures of 75 psi and 99.75 psi, data were recorded on magnetic tape for later analysis. Approximately 50 minutes of data were recorded, mostly at pressures above 75 psi.

The instrumentation system consisted of three high sensitivity charge accelerometers, mounted on the outer wall of the containment vessel in a triangular configuration. Signals from the accelerometers were

recorded on magnetic tape running at 60 inches per second.

Fabrication and installation of the data processor for use with the micropositioner readout portion of the creep data logger in use at KW facility was completed.

Fabrication of a wattmeter transducer assembly for use with the creep instrumentation system at the KW facility was completed. The transducers were designed to measure the power dissipated in each of the capsule heaters. Installation of the transducers was delayed pending possible instrumentation relocation in the vicinity in which the transducers will be mounted.

Test Engineering Support

Processes are being developed to solidify separations plant's liquid wastes and store the solid material in steel pots. A pot inspection facility, utilizing nondestructive testing techniques, is being developed to provide periodic measurements of pot wall thickness so as to detect the presence of corrosion. Of secondary interest is an inspection of pot contents for large discontinuities such as cracks and voids.

Experiments were performed, using various types of available piezoelectric transducers, to optimize bandwidth and sensitivity. Of the three types which were evaluated, lithium sulphate had the best resolution and PZT5A the best sensitivity.

A PZT5A crystal with epoxy resin facing and backing was irradiated in a gamma pit at the rate of 1.2×10^7 R/hr. Small changes, which would slightly impair resolving ability, were noted after about 2-1/2 hours. During the first night--between 4-1/2 hours and 21-1/2 hours of exposure--the crystal failed. The exposure rate during the actual test is expected to be no more than 10^6 R/hr. Consequently, a minimum transducer life of about 54 hours is expected. This is equivalent to 54 inspection sequences, which is considered adequate.

A microwave, liquid level probe, used previously in a waste storage pot test, was salvaged for use in another storage pot. The test was run in the second pot with limited success. The used probe had apparently been damaged during the first usage and as a result the oscilloscope readout was somewhat distorted. Reasonably good over-all results, however, have prompted the installation of a new probe in the latest test pot.

Long term oscilloscope drift continues to be a problem. A 12-in., all platinum, probe is being fabricated for glass melter tests which are scheduled to start next month.

A study has been completed on the applicability of nondestructive testing techniques for measurement of the thickness of plutonium deposited on a metallic substrate. Data were taken on the stability of prototype eddy current equipment from which it was estimated that accuracies in the 3-5% range could be obtained if the probe to specimen spacing can be kept below 5 mils. The results of this study have been compiled along with the results of ultrasonic tests into a rough draft report.

In addition to the eddy current techniques, ultrasonic techniques were also evaluated. A new sample was received--15 mils of lead-tin solder brazed onto a half-inch thick copper plate. Attempts to locate an amplitude variation of a short time duration packet of oscillations sent through the sample while frequency was varied, were only partially successful. It was concluded that the pulsed oscillations resonance approach would be unreliable for this application.

The second approach would measure the time duration between the surface echo and interface echo. Although resolvable echoes were received on a sample of .020 inch of lead brazed on 0.213 inch of solder, the echoes from the 15 mil sample were too scrambled to interpret.

Optics

The design of a nondarkening relay lens for periscopes has been completed. The spot size or resolving power of this lens is in the 10^{-5} in. range. The design of the objective lenses for the PRTR and 100-N periscopes is nearing completion. At the present time the spot size for both of these periscopes is in the 10^{-5} in. range.

A set of interference filters has been fabricated with transmittance of 1 percent to 2.2 percent in steps of 0.2 percent transmittances. The high vacuum deposition equipment was used in the fabrication of these filters.

Some changes in the glow discharge component of the NRC vacuum system were made with no appreciable gain in the cleaning ability of this equipment. The glow discharge power output was measured to be approximately 50 watts, which is about a factor of 10 to 20 lower than the desired level.

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

was performed:

1. Four ultrasonic test specimens were fabricated.
2. Three shutters were repaired.
3. One Redox head was repaired.
4. Three borescopes were repaired.
5. One periscope eyepiece was fabricated.
6. One pyrex seal was fabricated.
7. One set of glass bearings was fabricated.
8. Several lenses were reconditioned.
9. Several front surface mirrors were fabricated.
10. Three cadmium sulfide crystals were coated with indium.
11. Two quartz tubes were coated inside and outside with iron.
12. Several bell jars were inspected.

Physical Testing

Demand for leak testing has increased considerably. An emergency call to locate a propane leak in the 329 Building was found, allowing the contractors to continue their work. Five Freon-12 leaks were located in the 702 Building cooling system. The N Area fire protection sprinkler system supply line was located and identified as the temporary line from the 100-200 export line; a leak was located in the Liquid Sodium Loop for Equipment Development Unit; a check was made for leaks in the 300 Area contaminated waste network; and lines were staked out in the old Hanford Townsite for the AEC.

Other work included etching and autoclaving eight K type, Zircaloy-2 process tubes; eddy current and ultrasonic tests on 109-N steam generator tubes; ultrasonic testing of fuel elements for PRTR; and the primary water system pump motors in the 100 Areas. A total of 5663 end caps for fuel elements were inspected by fluorescent penetrant as were the 109-N turbine diaphragms and blades. Density variation in fuels, vibration analysis, and accelerometer studies were completed on PRTR fuel rods.

Radiography on construction projects for the month includes the TC-3 dissolver; the 1706-KER corrosion loop; the D-14 concentrator, TA-3 and TB-3 dissolvers. HUPO Electrical Utilities ordered radiography of their fiberglass boom and ladder for safety protection. PRTR requested examination of eye bolts used in reactor operation. Defects were found in both instances. New work consists of radiography of 1200 Flow Splitter fuel elements and the processing of 50 autoradiographs per day for N Fuels.

An ultrasonic crack test was applied to monitor the rotor bars in a 4500 hp motor. The motors are used to drive pumps supplying the reactor primary water. The test found extensive cracking in one of four motors tested. Metallographic examination of the suspect rods confirmed the ultrasonic results. A program to monitor additional motors was outlined. If cracks are found, a system to monitor the rate of crack propagation will be required.

INSTRUMENT EVALUATION

Modification and evaluating testing work was continued on the eight alpha-only and alpha-beta-gamma hand and shoe contamination monitors. Component layout and wiring problems remain to be resolved before general evaluation testing can be initiated.

Evaluation testing was continued on a commercial gross, high-level, alpha air monitor. The instrument, if acceptable, will be used for air monitoring at the Biology facilities.

Manager
PHYSICS AND INSTRUMENTS DEPARTMENT

RS Paul:mcs

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CHEMISTRY DEPARTMENT
RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - O2 PROGRAM

IRRADIATION PROCESSES

Overheated N-Reactor Fuel Study

Experiment design and facility preparation were resumed after the several months devoted to thermal stressing experiments with special target elements.

Reactor Effluent Disposal

A reactor effluent disposal test in which effluent was diverted into a trench at 100-F Area was terminated because of seepage through the trench embankment. The final steady state infiltration rate was not determined because of flow interruptions during the test; however, no decrease was detected during a ten-day period in which 2×10^8 gallons of effluent were disposed. Radionuclide analytical data indicated that water moves at a rate of about eight feet per hour through the soil in this area and that decontamination factors for Cr-51 and Zn-65 were 2.6 and 118, respectively, for travel through about 700 feet of soil. The total beta decontamination factor was 21.

SEPARATIONS PROCESSES

Disposal to Ground

Large volume ground-water samples, obtained from three wells located from two to seven miles southeast of 200 East Area, were given special analytical processing to define the isotopes contributing to routine gross beta activity determinations. Results confirmed that essentially all of the activity was attributable to Ru-106 - Rh-106 and tritium. Technetium-99 was detected in all original and duplicate samples; the maximum concentration noted was 3×10^{-7} $\mu\text{c/cc}$ (168 hr. MPC_w is 3×10^{-3} $\mu\text{c/cc}$). The closest well to 200-East Area, 699-34-39A, showed total strontium concentration 1.5×10^{-8} $\mu\text{c/cc}$; this well was resampled for confirmation.

Two, 25-foot, core-barrel holes were put down immediately adjacent to two 16-foot deep buried trenches in the 200-W Dry Burial Garden #2. Samples were checked for indications of radionuclide (specifically

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plutonium) migration into subsoils beneath the trenches, which were filled in 1953. No contamination was detected in any of the samples.

Plutonium Aerosol Studies

The fraction of plutonium airborne and transported from small metal specimens oxidizing at 400 C was found to be very small, of the order of 10^{-5} to 10^{-6} percent. Air flow over the specimens was 3 cm/sec in one case and 13.5 cm/sec in another. Particles larger than 15 μ were air-washed in an elutriator from the total oxide to determine the fraction which might become airborne. Only a very small fraction could be removed by air-washing at the terminal velocity of a 15 μ PuO_2 particle. Some 3 to 4 μg of a total initial weight of about 12 grams was collected. One rather surprising result was that of this amount from 80 to 90 percent by mass consisted of particles of smaller than about 1.5 microns diameter.

Preparation was started for examination of airborne releases from "overheated" pure plutonium compounds and from burning combustibles containing plutonium in various forms.

Spray Calcination of Thorium Nitrate

A series of spray calcination runs, aimed at determining the effect of calcination conditions on thoria product quality, was completed and evaluation of the products is continuing. In addition to CP thorium nitrate, thorium nitrate solution from the recent "6-ton" plant campaign was also calcined. The thoria from the latter work, which will be used for dose-rate and shielding measurements, is much more radioactive than that from the laboratory material. Gamma spectrometry showed a high fission product content (primarily Zr-Nb and Ru) as well as Pa-233. Further purification will be required for meaningful measurement of dose-rate associated with thorium decay products.

Thoria Dissolution Studies

Aluminum jacketed target pieces, containing 300 pounds of thoria, were charged to the pilot plant model of the Purex annular dissolver. The aluminum jackets were removed by a caustic decladding operation and the thoria formed a mud-like layer approximately 13 inches deep in the annulus, corresponding to a charge of 4,800 pounds of ThO_2 in the plant dissolver. The air lift circulators operated quite satisfactorily under these conditions. The thoria was dissolved in two approximately equal cuts using 12.3M HNO_3 , 0.04M NH_4F , and 0.10M ANN. In the first cut, the dissolvent achieved a 1M Th concentration in 10 hours; in the second cut (wherein the circulators were operated to promote upward

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flow of liquid through the bed of solids and the depth of solids was less) a $1M$ concentration was reached in approximately 6 hours.

Investigation of the excessive abrasion of the dissolver floor which can be caused by thorium particles suspended in a liquid was completed. The long-term runs at relatively low velocities were finished; the results confirmed the fact that the velocity of the impinging stream controls the abrasion. In the velocity range of interest, the rate of penetration into the metal is approximately a seventh power function of the jet velocity. Restricting the jet velocity from the air lift circulators in the Purex annular dissolvers to less than 50 ft/sec will minimize this effect.

Thorium Processing

In support of a co-dissolution (aluminum cladding and thorium) procedure for processing irradiated thorium in the Purex plant, the solubility of thorium nitrate in nitric acid-aluminum nitrate solutions was determined as a function of nitric acid and aluminum nitrate concentrations. As expected, increasing aluminum nitrate concentration and nitric acid concentration depressed the solubility of thorium nitrate. For example, at $0.3M$ aluminum nitrate, thorium solubility decreased from $1.94M$ at $2.75M$ HNO_3 to $0.93M$ at $9.8M$ HNO_3 ; at $0.9M$ $Al(NO_3)_3$, thorium solubility decreased from $1.5M$ at $2.75M$ HNO_3 to $0.67M$ at $5.7M$ HNO_3 .

Corrosion by Thorium Dissolvents

Long-term tests indicate a significant increase with time in the rate at which 304-L stainless steel corrodes in thorium dissolvents. For example, in boiling $12.5M$ HNO_3 - $0.01M$ HF - $0.1M$ $Al(NO_3)_3$ the rate increased from about one mil/mo during the first 48-hour period to about seven mils/mo during the sixth 48-hour period. Similar increases are obtained with other proposed dissolvents. These tests are continuing; similar tests on HAP0-20 alloy are underway.

Polonium Process Chemistry

Experiments have shown that mechanical attrition of bismuth oxide containing polonium does not affect the rate at which polonium is evolved upon heating. The fraction of polonium removed as a function of time from samples which were periodically subjected to mechanical attrition was almost identical to controls which received no mechanical treatment.

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Measurement of the vapor pressure of Bi_2O_3 has proved to be difficult, mainly because of the uncertainty in determining sub-microgram quantities of Bi in the presence of similar amounts of the more volatile Po; using Bi_2O_3 traced with Bi-210, an upper limit of 2×10^{-6} atm at 1000K was determined. A probably more accurate polarographic technique places the vapor pressure at ca 6×10^{-8} atm at that temperature. The latter value indicates a separation factor for polonium of about 3×10^3 which is larger than previously indicated.

Solvent Extraction Recovery of Po from Irradiated Bi

Miniature mixer-settler and batch contact studies were undertaken to test a dibutyl-carbitol solvent extraction separation of polonium from bismuth. Feed solution was prepared by dissolving irradiated bismuth (aged more than two years) in nitric acid to produce a 2M HNO_3 -2M $\text{Bi}(\text{NO}_3)_3$ solution containing about 0.02 curie of Po-210 per liter. The mixer-settler run tested extraction-scrub and strip column conditions recently proposed. Approximately 20% of the Po was lost to the extraction column raffinate and about 8% was not stripped in the strip column. Batch contact studies indicated extraction and stripping both to be relatively slow. Unfavorable kinetics and an inadequate number of stages (5 extraction, 7 scrub, 6 strip) may have been responsible for the poor recovery; further tests at higher Po level are planned.

Bismuth Oxide Reduction

Reduction experiments with carbon, hydrogen, and carbon monoxide showed the most effective reductant for Bi_2O_3 to be carbon monoxide; complete reduction of a 3 g -150 mesh sample was obtained in a flowing stream of carbon monoxide in 1.5 hours at 800 C. Carbon in an argon atmosphere gave complete reduction of 25 g of -150 mesh bismuth oxide in a static atmosphere in two hours at 800 C; stoichiometric carbon to form Co was necessary, however, because excess carbon forms a porous composite with the bismuth and this in air at 700-800 C gives oxidation of the bismuth and not the carbon. Exposure of a 3 g -150 mesh sample of Bi_2O_3 in a flowing stream of 92% Ar-8% H_2 mixture gave only an 80% reduction after three hours at 800 C.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Zeolite Storage of Sr

The Sr-90 ion exchange loading operation was continued to an instantaneous breakthrough of 30% and an integrated loss of 3.3%. The loaded column contained about 2.5 meq of strontium per g of zeolite

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and had about 52 watts thermal output. The column was water washed, dried at 400 C, and transferred to Radiometallurgy for helium purging, closure, and leak testing. It will then be maintained at 500 C and observed for pressure buildup.

In other Sr storage studies, the canister which was loaded about a year ago was eluted with 30 column volumes of 1M HNO_3 . A high activity peak after the first few column volumes was followed by a decrease to very low levels; more definitive analyses were underway.

Semiworks Support

A storage tank of strontium-rare earth crude concentrate was recently flooded in the plant with process waste solution. The material was subsequently concentrated and plans made to recover both the strontium and the rare earth (promethium). A flowsheet to accomplish this, utilizing installed B-Plant facilities, was developed. Analysis indicated the presence of a high concentration of sodium and lead and lesser concentrations of aluminum and fluoride. Because of solubility problems with the aluminum and fluoride, an acid-side flowsheet was developed and tested in the Analytical Hot Cells. This employed (1) precipitation of strontium and rare earths (as sulfates) by addition of NaHSO_4 , (2) a caustic-carbonate metathesis followed by nitric acid dissolution (to remove lead), and (3) a conventional oxalate separation of rare earths from strontium. Losses of both strontium and rare earths were less than 5% and the excellent character of the precipitates rendered the operations amenable to being performed in the plant tanks, rather than in centrifuges.

Promethium Solvent Extraction

Knowledge of the equilibria and kinetics of extraction of Pm and Ce into D2EHPA solvent systems is being extended and refined by systematic measurement of the extraction coefficients as a function of time, temperature, added aqueous-phase complexant, pH, etc. Preliminary data indicate that Ce(III) and Pm(III) may differ sufficiently in kinetic behavior to allow a separation without use of oxidants.

Fission Product Separation and Packaging

The adsorption and elution behavior of Rb^+ and Na^+ with Linde AW-500 zeolite was studied to evaluate possible effects on Cs^+ loading and the purity of the Cs^+ eluted from a storage canister or from a process purification column of AW-500 zeolite.

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Results indicated that about 60% of the Rb^+ in simulated Purex alkaline supernatant waste was adsorbed on the zeolite from 60 column volumes. About 10% of the adsorbed Rb^+ was removed in the scrub cycle, most of it during the latter part. An increased volume of scrub solution should increase significantly the amount of Rb^+ removed.

A simulated cesium packaging solution was made up with the assumption that half of the Rb^+ was carried over in the cesium separation step. The Cs:Na and Cs:Rb ratios in the solution were 1:8 and 10:1, respectively. This solution was pumped through an AW-500 column until 25% Cs^+ breakthrough was obtained (16.5 column volumes). Eight column volumes of 0.1M $(\text{NH}_4)_2\text{CO}_3$ scrub was pumped through the column to remove Na^+ and Rb^+ . Analytical data indicated that most of the Na^+ , about half of the Rb^+ , and 16% of the Cs^+ were removed. The final Cs:Na and Cs:Rb ratios in the eluate were 14:1 and 18:1, respectively.

Shipment of Fission Product Cesium on Zeolite

The particle size distribution of a sample of 14 x 30 mesh Linde AW-500 zeolite was determined in order to evaluate the effects of vibration, freezing and thawing during rail transit. The results were compared with those of a sample of the same zeolite taken prior to shipment. The amount of material in the 30 x 100 mesh range increased from 1 to 6% and the amount passing a 100 mesh sieve from 2.1 to 2.5%. No data are available for Decalso for comparison but the physical stability of the AW-500 appears to be satisfactory for use in the shielded transfer casks.

Recovery of Np, Pu and Am from Purex Stored Waste Sludge

Further tests to determine the path of Np, Pu and Am during water sluicing of Purex sludge and nitric acid leaching of residual solids were completed. About 25, 25, 75 and 5 percent of the Np, Pu, U and Am, respectively, present in samples of Tank 104A sludge was found in water leaches (based on flow volumes suggested in HW-79235) of the sludge. Corresponding numbers for Tank 101A sludge were 20, 25, 30 and <1 percent. Treatment of the residue from water-leached 104A sludge with nitric acid at 80 C solubilized residual neptunium and plutonium. Either a single large-volume of HNO_3 or successive small volumes of HNO_3 gave satisfactory removal of both. Similar nitric acid treatment of water-leached 101A sludge effectively removed neptunium, but less than 50% of the plutonium.

EQUIPMENT AND MATERIALS

Corrosion of Boron-Containing Stainless Steel

Corrosion tests made on 304-L stainless containing 1% natural boron

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supplemented data obtained previously for 304-L containing 0.2 and 2% boron. Corrosion rates for the 1% alloy in Zirflex dissolvent at temperatures from 65 C to boiling were 20 to 30 mils/mo. The rates were essentially the same as for as-received, sensitized and welded samples. As-received and annealed samples corroded at rates of 2 to 4 mils/mo in boiling 40-60 w/o HNO_3 ; sensitization increased the rate in 60 w/o HNO_3 to 130 mils/mo. Photomicrographic inspection of corroded samples indicated no preferential leaching of boron.

Titanium Corrosion

A series of exploratory tests was made in an unsuccessful attempt to determine cause for the failure of two Ti tube bundles in the Purex waste concentrator. Exposure of stressed titanium samples alternately to HNO_3 -HF- $\text{Al}(\text{NO}_3)_3$ and to simulated Purex 1WW solutions has not produced stress cracking. Salt cakes were deposited on Ti samples by evaporation of simulated Purex 1WW. The caked samples were then exposed to HNO_3 -HF- $\text{Al}(\text{NO}_3)_3$ and alkaline tartrate solutions and these dried at about 150 C. No unusual attack on the Ti occurred. No galvanic effect was found where Ti and 304-L samples were coupled to carbon and exposed to boiling Purex 1WW solution containing 0.1M HF. A Ti sample exposed to Purex 1WW during and after sugar treatment to destroy nitrate showed no unusual attack. Titanium samples were exposed to simulated Purex H-4 bottoms containing fluoride concentrations ranging from 0.001 to 0.025M. Significant corrosion occurred only at 0.025M fluoride; liquid and vapor phase rates were 0.2 and 0.7 mils/mo, respectively.

Slag and Crucible Dissolver Development

A glass mock-up of the redesigned S&C dissolver was completed and placed in operation. Circulation tests using sand as a stand-in for fluoride-insoluble residues demonstrated that circulation can be maintained with gas sparging, steam sparging, or by thermal circulation; the combination of all three is sufficient to restore circulation even when the equivalent, settled residue from 250 S&C cans is present. During dissolution runs, pressure surges were experienced due to the reaction of fines for which mock-up was not designed; this phase of the program was curtailed. Slight modifications are underway to obtain more precise geometric similarity to the plant dissolver.

REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMSalt Cycle Process Development

Studies of the behavior of Np and Am in the salt cycle process were started. Neptunium codeposited with UO_2 in a partition-type UO_2 electrodeposition with a 50% HCl -50% Cl_2 sparge. About 73-80% of the Np codeposited with 13% of the U. On the other hand, Am tends to remain in the melt under either UO_2 electrodeposition or UO_2 - PuO_2 electrocodeposition conditions; Am DF's of ca 4000 and ca 100 (U and Pu bases, respectively) were measured for a UO_2 partition deposition while DF's of ca 7 and ca 100 (U and Pu bases, respectively) were measured for a UO_2 - PuO_2 codeposition.

Development of the process continued in the High Level Radiochemistry Facility. Analytical results showed that decontamination factors of 20 for fission product Pd from electrodeposited UO_2 will be achieved under both Pu partition and Pu codeposition conditions.

In one run with equimolar LiCl - KCl , HCl sparge, and a graphite liner, about 10% of the U was electrodeposited from the salt bath containing 19 w/o U. The Pu enrichment factor was 0.045. The Pd decontamination factor was 28 on a uranium basis. In the following run 75% of the U was removed from the bath by electrolysis with a 90% O_2 -10% Cl_2 sparge; the Pu enrichment factor was 0.84. The Pd decontamination factor was 19 on a U basis and 16 on a Pu basis.

The results indicate that the noble metal fission product Pd, which contributes about 6% of the fission product in a 30,000 MWD/t thermal plutonium recycle system, will not codeposit with the recycled UO_2 - PuO_2 but will remain behind in the molten salt.

Salt Cycle Process Control

Work continued on the study of "back EMF" measurement as an indicator of the progress of PuO_2 - UO_2 codeposition. The "back EMF" now appears to depend on several variables in addition to the PuO_2/UO_2 ratio in the cathode deposit so that the technique may not provide the desired accurate control for the codeposition process. The measurement does give a good indication of the approach to codeposition of UO_2 and PuO_2 and a warning of the onset of PuO_2 precipitation during codeposition.

Long life of a GE No. 1962 quartz light bulb at high temperatures and with thermal cycling between 25 and 600 C has indicated that a successful photometer, designed for use in a salt melt, can be designed around the use of a submerged lamp.

High Exposure Plutonium Handling

It has been postulated that appreciable concentrations of Pu-236 and Pu-238 in high exposure plutonium might require remote plutonium fuel fabrication methods and thus markedly affect fuel fabrication costs. Pu-236 decays to U-232 with a 2.85 year half-life and could potentially cause the same problem with plutonium that U-232 causes with U-233 from thorium irradiation, i.e., the hard γ associated with the Bi-212 and Tl-208 daughters.

There is not complete agreement on the threshold value at which the U-232 concentration in U-233 presents a serious radiation problem. Apparently, 1 ppm of U-232 in U-233 could be processed without undue difficulty because of radiation hazard. A concentration of approximately 5 to 6 ppm U-232 could be processed with special precautions, rapid processing, special cleanup procedures and added costs. On this basis, and factoring into the situation the relative growth of daughter activity from Pu-236 as compared to U-232, it appears that between 10 and 100 ppm Pu-236 in plutonium might be tolerated with no more than special attention to cleanup precautions.

An isotopic analysis of a Pu sample from a Yankee fuel element irradiated in two steps to 24,600 MWD/t shows 0.21 ppm Pu-236 in the plutonium. This indicates that Pu-236 in high exposure plutonium is not likely to be a problem.

The Pu-238 content of high exposure plutonium from power reactor fuels will be considerably higher than the Pu-238 content of high exposure plutonium processed to date in the Plutonium Fuels Laboratory (1 to 1.5% compared to 0.04%). The Pu-238 will be more of a concern than Pu-236 but whether or not this alone will force remote processing is not yet clear.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENTGlass Studies

Recent results of the systematic glass studies, including the results of leaching tests, are summarized in the January-March, 1965, Quarterly Progress Report on the Fixation of Radioactive Residues.

The leaching studies have been in progress for a total of 143 days. Three types of storage were studied: in dry air, moist air, and water. Leaching rates of glass samples from the three types of storage differed significantly: water storage increased the leaching rate about two-fold and moist air storage about ten-fold. Leaching rates in Hanford ground water (pH 7.8) were found to be twice as great as in laboratory distilled water.

Laboratory-scale equipment simulating the pot process was operated to produce 250 ml of melt in a small 304-L stainless steel pot. The feed used had the same composition as that used in earlier ORNL run in which severe corrosion was encountered. Three liters of feed were calcined and melted during a 48-hour run. Pot temperature was 850 C for the first eight hours and 900 C for the remainder. The solid product at room temperature was a mixture of black and light-green particles. There was a layer of the light-green material completely surrounding the solid mass; some void spaces were present. No observable corrosion of the pot occurred; it had been held at 900 C for some time prior to use and the heavy oxide coating appeared to be intact after the completion of the run.

Calcined waste simulating Nuclear Fuels Service first cycle waste was combined with phosphoric acid to give M^+/P ratios of 5 and 6. Complete fusion was not obtained with either of these mixtures at 1040 C. A series of eight melts was prepared from calcined simulated NFS combined first and second cycle wastes with added phosphoric acid. The range of M^+/P was from 1.7 to 3.1. Melting points for all of these mixtures were within the range 850-880 C. The melts were very fluid; solids from the melts were generally rock-like and not uniform in color.

Materials for Melt Pots

The alloy 50 Ni-50 Cr was catastrophically attacked by 1WW-phosphate and NSF-phosphate melts during 24-hour exposures at 1100 C. Similar attack occurred at 1000 C in a 1WW-phosphate melt at an M^+/P of 1.8 when potassium dihydrogen phosphate was used as the source of phosphate. The corrosion rate was about 14 mils/mo at 1000 C in a 1WW-phosphate melt with an M^+/P of 1.8 and phosphate added as phosphoric acid.

Spray Calciner-Melter Development

Four spray calciner-melter runs were made to determine operability of simulated NFS first and second cycle wastes with added phosphoric acid. Feed rates were set at five gallons per hour and the calciner furnace was operated at 700 C. Melt was produced at a rate of 0.6 ft³/day.

High sodium NFS first and second cycle waste with added phosphate ($M^+/P = 2.7$) produced partially melted deposits on the calciner walls and plugged the filters during a seven-hour run. Deposits were formed by the presence of a high concentration of sodium nitrate which melts at 308 C.

Sodium concentration of the feed was then reduced to one-seventh of the previous value and five-hour runs were made with M^+/P of 3.0, 2.7 and 2.4. Feed concentration was 22⁴ gal/ton U. A ring of powder formed in the conical bottom of the calciner while operating with feed having $M^+/P = 3.0$. Operation with M^+/P 2.7 was most successful; calciner wall deposition of only 1/16-inch resulted. Feed with $M^+/P = 2.4$ produced powder deposits which plugged the calciner outlet. Filter performance for the three runs was satisfactory. Maximum pressure drop across the filters was 19 inches of water.

The platinum melter was operated at 1100-1150 C during these runs. The melt was discharged intermittently through the overflow weir by applying vacuum to the melt collection pot. This batch discharge technique generally prevented stalagmite formation in the collection pot. The melts were relatively non-viscous fluids at temperatures of 1000-1100 C and the solidified melt was a brown, homogeneous crystalline material.

Design Verification Tests (WSEP)

Pot calciner run CPC-8 was completed using a draft tube to promote internal circulation and to delay scaling within the pot. A shortened draft tube enabled operation at a liquid level lower than that of similar previous runs so the upper section of the pot could be maintained at a temperature suitable for use of a thermocouple liquid level probe. This provided a good indication of the operating level and allowed a high average feed rate of 20 liters/hour to be maintained. The feed, modified (lower Na^+) NFS first and second cycle waste with Na^+ added, was totally calcined in 50 hours and produced 89 kg of calcine, the pot being filled to capacity.

By controlling individual zone temperatures of the six-zone induction heated furnace it was possible to decrease variations in the temperatures so that all calcine temperatures went from 600 to 800 C as a group. This procedure shortens the time the calcine is at high temperatures and should help to prevent volatility of sodium, cesium, ruthenium and sulfate compounds.

WSEP Instrumentation

A study was concluded comparing the relative merits of the 600 point data logger and the GE-412 computer in data accumulation and data processing for Waste Solidification Engineering Prototype. The GE-412 was recommended for on-line data processing of WSEP experiments in FRPP.

Two cold runs were monitored with the 412 computer. Relocation of the logging typewriter to the control room caused some programming

problems during the initial run but the second proved the most successful to date from the standpoint of both hardware and software performance.

Additional instrumentation work included revision of the evaporator viscosimeter to facilitate cell installation and to narrow the viscosity range for improved accuracy. An alternate viscosity measuring device, based on response of a vibrating reed, was received and installed in a test stand for calibration and performance testing. Work was underway on improved readout devices for use with waste pot pressure transducers. Also under test was an inductive proximity gage for measuring deflection of a stainless steel diaphragm; poor linearity was observed above 10 psig but hysteresis and repeatability were well within the 5% required by the application.

CONTAINMENT SYSTEMS EXPERIMENT

Major Facilities and Equipment

The construction of the three-part containment systems vessel was completed by the end of March and the pressure tests were performed successfully in early April. The soap bubble tests on the containment vessel (75 psig) and dry well (150 psig) revealed no leaks at the design pressure. However, some leaks were discovered in the bulkhead seams of the wet wells. The defective welds were removed and the seams rewelded. A subsequent pressure test revealed no leaks.

Design of the containment systems appurtenances was completed and construction was started.

Plans for installing and using the SM-2 vessel for instrument development studies were revised in an effort to lower both the installation cost and the future operating costs. The vessel will now be located on a concrete pad at 271-CR Building in 200-E Area instead of at the 321 Building location in 300 Area.

Mathematical Models

The blowdown portion of the computer program was extended to include the effect of vessel sensible heat and to indicate temperature and pressure gradients in the vessel wall. The effects of the pressure-temperature state at saturation and the instantaneous temperature difference between the wall surface and the liquid were evaluated using the theoretical equations of Zuber and others.

A more refined model of the blowdown event was programmed which allows for superheating in much of the liquid during blowdown with vaporization occurring from only a limited region near the top of the vessel.

The effects of inertia and pressure drop through the core on local stagnation pressure near the break are approximated in the model. These refinements are believed to represent more nearly the actual situation and to result in a closer match to the limited amount of existing experimental data.

Fission Product Simulation

The suitability of the cooled quartz tube method for melting and retaining clad UO_2 specimens for long periods was proven by a two hour run. During this period the aerosol generation rate was between 130 and 170 mg of metal oxides/min. The higher generation rate was achieved in the second hour when both the temperature and air flow were increased. This experiment provided increased assurance that aerosols could be generated for injection into the containment systems over prolonged periods.

Another test demonstrated the capability of melting UO_2 at distances up to 20 feet from the high frequency generator. A 20-foot coaxial lead was used at 3.2 mc to melt stainless steel tubing. The capability of doing this was deemed to be sufficient proof of its capability to melt UO_2 .

CSE Computer Programming

Plans and schedules were developed for programming the PDP-7 computer recently ordered for CSE use as a multi-channel analyzer in tracer tests. In the later blowdown test phase, the computer will serve as a rapid (2000 points per second) data logger, recording voltage signals and representing transient system variables on magnetic tape. Reduction of the data to engineering units and subsequent analysis will be accomplished using Fortran programming, either on the PDP-7 or on the IBM-7090. The fast transient data logging program is essentially independent of the types and locations of sensors, needing only voltage inputs; it can thus proceed independently of the sensor evaluation and selection work.

OFF-GAS DISPOSAL TO GROUND

Although an optimum depth of gas injection does exist (see BNWC-8 3 C), the depth is too great to use for the practical injection of gases. It is of value, however, as a scaling factor to reduce design curves to manageable form. The travel times and flow rates may then each be related to the scaled depth by single curves.

Travel times and flow rates computed for an incompressible flow system cannot be related directly to the corresponding quantities in a

compressible system. If the dimensionless potential from an incompressible system is a sufficiently good approximation to the dimensionless potential of the compressible system, then the potentials can be equated. However, the analogy should stop there and all further calculations should be performed in the compressible system.

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BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The dynamic equations describing multiphase flow in heterogeneous porous media were shown to satisfy the special conditions of classical complex lamellar velocity fields. Although such fields are not exhaustively treated, there are some useful theorems available. Specific areas of possible application are the provision of better fitting functional forms for potential data and the provision of energy integrals which may prove the minimal characteristics of travel time predictions, based upon barotropic and reduced forms of flow in homogeneous media.

Several significant changes were made to increase the utility of an existing hydrology computer program, GENORO. Work was started to define the permeability boundary values by a function (through GENORO fitting) rather than using them as tabular data.

Correlation of basalt cuttings from the recently completed artesian irrigation well in upper Cold Creek Valley to the logs of old artesian wells farther east confirms the general off-lap relationships of geologic units upward in the geologic section. All units there dip eastward; hence, each successively lower unit crops out at a higher altitude and more distant from the Hanford Project. Each successively lower aquifer can be expected then to have a higher piezometric head owing to its higher site of recharge; limited data confirm that. Well 699-49-48 evidently reflects the head of the lower Beverly Member aquifer of 432.7 feet where the unconfined ground water and the head of the upper Beverly Member aquifer are at 405 feet. At the west edge of the Hanford Project the head in the Sand Hollow basalt flow (the artesian aquifer tapped by the irrigation wells) is 500 feet higher than the unconfined ground waters there. In general, the movement of ground waters in the confined aquifers will be upward, where pathways exist, into aquifers of lower head, thus minimizing the movement of contaminants into the basalt series. If great enough heads are created in higher aquifers by buildup of mounds, as at 200-East Area, flow reversal can occur.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Cosmic-Ray Produced Radionuclides

Thin sections of an iron and a stony meteorite were counted on the new multidimensional gamma ray spectrometer. The Al-26 which is produced

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by cosmic ray bombardment of the material in space was measured for both samples. Although the Al-26 content of stony meteorites has been previously measured by direct coincidence counting, this is the first direct measurement in an iron meteorite. The measurement was made possible because of the extremely low background of this analyzer system. Determination of the absolute amount of Al-26 in the meteorites awaits receipt of a standard for that radionuclide.

Multidimensional Gamma Ray Spectrometry of Fission Product Mixtures

The measurement of Ru-103, Ru-106, Ce-141 and Ce-144 in five-month old fission product mixtures generally requires chemical separation since the gamma ray spectrum of the mixture cannot be resolved; however, these radionuclides can be measured readily in such a mixture using the multidimensional gamma ray spectrum. The Ru-106 and Ce-144 can be measured from their coincidence gamma ray peaks in the energy-energy plane with very little interference so that a precise correction for their contribution to the photopeaks from Ru-103 and Ce-141 can be made. Ba-140 is also easily measured from its coincidence gamma rays with no interference from other fission products.

Fallout Studies

Studies of the trace radionuclide behavior in Alaskan food chains show higher concentrations of Na-22, Cs-134, Cs-137, Co-60 and Zn-65 in wolf than in caribou flesh. The main food of the wolf is caribou as it is for the Eskimo. The liver and kidneys of wolves show concentrations of several of the trace elements much higher than in muscle and also show Ag-110m and Mn-54 which have not been observed in muscle.

Livers of local beef animals have measurable amounts of Na-22, Mn-54, Co-60, Zn-65, Ag-110m, Cs-134, Cs-137 and Th-232. These are the same radionuclides found in Alaskan caribou livers but are at concentrations lower by factors of 10 to 1000. Ra-226 can also be measured in these samples.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Aerosol Studies - Deposition Velocity Measurements

Monodisperse aerosols prepared with the spinning disc generator were employed for measuring the deposition velocity, K. Use of monodisperse particles and improved techniques of measuring wall deposition are permitting better definition of the effect of air flow, particle size, and tube diameter on the transport of particles. In experiments using 0.5-inch O.D. tubes and 1.9 μ diameter particles, K/V, the mass transfer

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coefficient, showed a minimum with decreasing s^+ (reduced stopping distance) at a Reynolds number of about 5,000. Earlier work using polydisperse zinc sulfide had not shown this minimum, although in some measurements a less rapid decrease in K/V , with decreasing s^+ , had been noted than for the bulk data, suggesting such a phenomenon.

Work with 1-inch O.D. aluminum tubes confirmed the general shape of the transfer coefficient, K/V , vs. ϕ_{s^+} curves. The tube diameter term exponent appears to be nearer to zero than the 0.84 previously determined to achieve correlation. Reinforcement of this conclusion will require further experimental determinations.

Deposition of 5.6 μ particles in 3-inch O.D. tubes was measured, over a flow ranging from a Reynolds number of about 120 to nearly 32,000. For Reynolds numbers from 120 to around 4,000, the mass transfer coefficient decreased. At still higher Reynolds numbers deposition increased until at about 32,000, K/V was again equal to that at about 120.

The use of monodisperse particles has greatly facilitated this study, but the better-resolved deposition data will require development of models which account for the relatively higher deposition at low Reynolds numbers.

RADIOLOGICAL AND HEALTH CHEMISTRY

Iron-55 in Humans

Determination of the Fe-55 concentrations in blood and in many food samples has provided an understanding of the distribution and routes of this fallout radionuclide. The main sources of Fe-55 in humans are meat and grains. Estimates of the total body burdens of Fe-55 are: Anaktuvuk Eskimos, 61 nCi; Richland residents, 8 nCi; Alaskan caribou, 728 nCi; and Washington cattle, 440 nCi.

Distribution and Excretion of Technetium in Humans

Tracer technetium administered orally to humans as the pertechnetate was measured by whole body gamma scanning and excreta analysis to determine the metabolism of Tc so administered. Although the data are still being evaluated, the findings differ in several respects from those in the ICRP Committee II Report. The values of effective half-life and of the fractional absorption from ingestion found in this study were about two days and over 0.95 retention; the Committee lists one day and 0.5 retention. These data suggest that a change in the MPC may be indicated.

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ISOTOPE DEVELOPMENT - 08 PROGRAM

ISOTOPIC FUELS DEVELOPMENT

Prototype Heat Source Development Equipment

Installation of the Pm purification equipment was started in the hot cell. The G.M. manipulator will be installed through one of the other cells after repair. It was predicted that at least two months will be required for construction of the "Fab" boxes, one month for room modification and installation, and one month for equipment installation and shakedown for a minimum of four months from start to hot operation.

Microspheres

Use of Pu-238 as insoluble dioxide microspheres was proposed by Mound as a contribution to the solution of the aerospace re-entry problem. Scouting studies were initiated here to examine this approach for Pm also. Using Sm as a stand-in, microspheres of excellent appearance were formed with either 50 or 100 μ mean diameter by dropping suitably sieved, crushed or ground oxide material through a plasma torch. The fused particles formed were spherical, transparent, hard and glossy, and had bulk density ca 60% theoretical. They appear to be insoluble in water. Quantitative measurements of solubility, both in fresh and sea water, were planned for a later batch traced with Pm-147.

Thermal Conductivity of SmF_3

The thermal conductivity of SmF_3 was determined, from 20 to 400 C, by use of a heat pulse method on a silvered wafer of vacuum-melted material. The results are in close agreement with those for SrF_2 and support the thesis that the burnup characteristics of PmF_3 and SrF_2 should be very similar.

Compatibility Testing

Compatibility tests of Sm_2O_3 and Nd_2O_3 as stand-ins for Pm_2O_3 continued at both low and high temperatures with a variety of metals. The short-term low temperature tests at 1100 C showed no attack, but it is premature to draw definite conclusions.

Promethium Shielding Measurements

A method of high sensitivity was developed for experimentally measuring the dose rate from small sources as a function of shielding. An advantage of the method is that quite thick absorbers can be used.

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The transmitted radiation is counted with a gamma scintillation spectrometer and the results are then coded and fed into an IBM 7090 computer code which reduces the spectral data to a dose rate. Excellent agreement was obtained between the spectrometer measurements and a dose rate measuring instrument for samples to which both were applicable. Reasonable agreement was obtained with calculated dose rates but the measured dose rates through thick shields were less than the calculated. The work will be extended to other shields and other isotopes as time permits.

ISOTOPIC FUELS ENGINEERING STUDIES

Encapsulation Techniques - Pneumatic Impaction

The evaluation of pneumatic impaction as a means of isotope encapsulation continued. The 2-1/2 in. diameter punch and die system was installed and 27 simulated cylindrical heat sources using SrTiO_3 as the fuel form were compacted from initial bulk density as low as 2.5 g/cc to final density as high as 4.8 g/cc without failure of any parts. Punch pressures of about 300,000 and 400,000 psi were achieved in tool steel and tungsten carbide punches, respectively. Changing the punch and die lubricant from a high temperature copper lubricant to a water base colloidal graphite (Aquadag), and changing the punch protector from mild steel to brass provided the following benefits: (1) an improved capsule surface, (2) an easier ejection of the capsule from the die, and (3) a reduction in die cavity wear.

To demonstrate the ability of the impaction machine to fabricate complex geometries, a 2-1/2 in. diameter by 2.8 in. long capsule containing five 5/16 in. I.D. tubes was developed. The tubes run the axial length of the capsule and could be used to heat a working fluid.

A study of possible limitations in the handling of radioisotope powders due to self-heating effects was started. Thermal conductivity measurements were made on SrTiO_3 powders of several particle size ranges at 250 C.

Dynapak Instrumentation

Instrumentation to measure and record strain at several locations on the Dynapak, including the punch, was installed and successfully operated. Several types of strain gage cement were tested to find the one that produces the best bonding to the tungsten carbide punch; several protective coating compounds for strain gage use were under evaluation. Manipulation of the strain gage signals was studied to gain better insight into the deformation process. Signals can be electronically differentiated or integrated to provide strain rate or energy input.

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ANALYTICAL DEVELOPMENT

Determination of Excess Oxygen in UO_2 -Tungsten Cermets

Work continued on the development of a method for the determination of tungsten-bound oxygen in UO_2 -tungsten cermets. The procedure depends on the use of a hydrogen gas sweep at 700-1000 C for removal of oxygen from the tungsten and measurement of the water produced. In measurements on a series of two-gram aliquots of the same sample, the method has a precision of $\pm 4\%$ or ± 10 ppm, whichever is larger. Little oxygen pick-up occurs while a sample is being powdered, but the finely divided material picks up oxygen slowly.

Determination of Total Uranium in UO_2 -Tungsten Cermets

A simple procedure was developed for the determination of total uranium in UO_2 -tungsten cermets. It includes the following steps: (1) a basic carbonate fusion of the cermet sample, (2) dissolution in bicarbonate solution, and (3) determination of uranium concentration in the solution using a thorium internal standard.

M. T. Walling, Jr.

Manager
Chemistry Department

MT Walling:DWP:cf

BIOLOGY DEPARTMENT

A. ORGANIZATION AND PERSONNEL

L. C. Cronk, a former General Electric Employee, was an addition to Biological Analyses Section on April 4.

A. M. Harty, Biological Technologist in Biological Analyses Section, resigned on April 6.

L. R. Richardson, Biological Technologist in the Metabolism Section transferred to Inhalation Toxicology Section April 5.

E. M. Uyeki, Senior Research Scientist in the Metabolism Section, resigned to accept a teaching position with the University of Kansas on April 1.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Columnaris

No columnaris organisms were isolated from over 200 fish sampled during April from four sites along the Columbia River. The decreasing trend in titre values and percent of fish with titres which was reported last month is continuing. The percent of fish with titres is relatively greater in the Hanford area than in any of the three others sampled.

No columnaris organisms were obtained from fingerling salmon currently in the McNary spawning channel. These fingerlings are scheduled for release at the end of April and in some years heavy infection with columnaris has been identified in this spawning channel.

Reactor Effluent Monitoring

Monitoring of salmon in 0, 2, and 4% reactor effluent at three different temperatures continues at 100-KE. All lots are now feeding. The mortality after hatching shows a trend of increased mortality at both the 2 and 4% reactor effluent heat increments, but as yet, no definite trend is related to effluent concentrations.

BIOLOGY AND MEDICINE - 06 PROGRAM

Phosphorus

An experiment designed to study the metabolism of P^{32} in fish with respect to the variance between individuals, and the effect of the feeding regimen was initiated.

Eighty whitefish and 80 rainbow trout were divided equally into two treatment groups of starved and fed fish. The fed group's diet is Clark's commercial fish food at the rate of 2.5% of body weight daily. An oral dose of 0.05 μCi P^{32}/g fish is being fed daily for six days. Subsequently, tissue samples from each group will be counted for P^{32} weekly for five weeks post administration.

Strontium

Two additional animals (over 3 years old) in the 125 μCi $\text{Sr}^{90}/\text{day}$ group developed clinical cases of leukemia (for a total of 10 suspected and/or confirmed in this group out of 26 over 2 years old) as diagnosed from the peripheral blood. Both of these swine received Sr^{90} in utero and by ingestion after birth. One neoplasm appears to be a lymphocytic leukemia, the other a myelogenous leukemia.

Cesium

Six ewes in varying stages of pregnancy (avg. 123 days) have received Cs^{137} daily throughout the gestation period. Cesium-137 pellets are administered orally according to the individual feed intake. Body burdens of these animals reached eight times the average daily cesium intake after the first 30 days of feeding and are now 10 times the average daily cesium intake. One ewe was sacrificed 113 days after breeding to determine the Cs^{137} concentration of the fetus. This animal maintained a body burden of 12 times the average daily intake of Cs^{137} . The body burden of the single female fetus weighing 2 kg represented 0.86% of the total burden of the dam (or $\sim$ one-tenth of the average daily Cs^{137} intake of the dam). Three male sheep were incorporated into this study to compare the sexual differences, if any, of Cs^{137} retention. These animals are exposed to the identical environmental factors.

Ten thick-walled LiF dosimeters were surgically implanted in each of five yearling rams. Daily feeding of 50 μCi Cs^{137} began when the animals had recovered following surgery. Two of these sheep are confined in small pens, the other three are allowed to move about in a larger pen. The body burdens measured by the whole-body counter are $\sim$ 220 μCi after 7 days of restricted movement and $\sim$ 150 μCi for those which exercise.

Six other male sheep were fed 100 μCi Cs^{137} per day for 8 days, after which 50 μCi per day was fed. After 15 days the body burdens were about 450 μCi . When approximate equilibrium is reached, LiF dosimeters will be implanted. At this level, the dosimeters will give reliable readings after one week of exposure.

Inhalation Studies

Rats treated with potassium cyanide (0.75 mg/ml drinking water) for 30 days showed slightly lower rates of clearance of inhaled iron

particles than control animals. However, the inhibitory effect was less than that observed for rats treated with cyanide for three months.

Dogs exposed to aerosols of $0.15 \mu \text{Fe}^{59}_2\text{O}_3$ particles deposited 68-85% of the total inhaled. The same dogs deposited 31-83% of total $0.3 \mu \text{Cr}^{51}_2\text{O}_3$ particles inhaled.

Femurs of rats sacrificed four months after exposure to uranium ore dust (pitchblende) contained five to 10 times, and livers 25 to 75 times more Th^{230} than U^{238} . Lungs contained about nine times more Th^{230} than U^{238} . All of the data from the study indicate that uranium contained in the inhaled ore particles was selectively translocated from the lung and excreted, whereas thorium was selectively retained.

Therapeutic Removal of Radionuclides

Combined treatments of DTPA and DFA failed to effect any greater removal of Ce^{144} from rat tissues than did DTPA alone.

Gastrointestinal Radiation Injury

Eight rats given 1.0 mm "insoluble" particles each containing 0.87 mCi of Sr^{89} by gastric intubation showed no gross pathologic signs of radiation injury. The particles were excreted on days 1, 2 and 5 and showed a 50% decrease in surface radioactivity after passage through the GI tract.

Subsequent administration of eight particles to a single rat caused diarrhea and apparent gross intestinal injury. The particles were excreted from day 1 through 6. Regions of retention were not determined; however, hemorrhages were noted in the glandular stomach and blind pouch of the cecum.

SNPO

The gastric mucosa and skin of three Hanford miniature swine were irradiated with particles of simulated reactor debris having surface dose rates of approximately 10,000 rads per hour. For irradiation of the gastric mucosa, particles were attached to a stomach balloon which was then swallowed and inflated. An exposure duration of one hour did not produce observable gastric mucosal lesions after 2, 3 or 15 days. Since the exact sites on the gastric mucosa irradiated could not be determined, marginal damage might have been overlooked. One hour exposure to the particles did produce a slight, very localized skin erythema between 10 and 15 days. Skin irradiation of a fourth pig using a 4-hour exposure caused initial flushing with definite erythema by the 7th day. Therefore, the gastric mucosa of other pigs will be irradiated using this longer exposure duration.

Calibration of LiF Thermoluminescence Dosimetry System

The study of the effect of temperature of irradiation is continuing. The immediate goal is proper evaluation of dosimeters exposed in vivo at body temperature. (Our equipment has been augmented by the addition of borrowed equipment to record glow curves for proper evaluation of thermoluminescence and to determine what type of glow curve recording apparatus to purchase. We have received two solid crystals from Harshaw Chemical Company at no cost. Preliminary results using these LiF crystals are good.)

Effects of Water Temperature on Fish

Several histochemical techniques for mammalian tissue have been modified and adapted for fish tissue. Unexercised and exercised male and female rainbow trout acclimated to 5° C were sampled for liver, red and white muscle to determine the effects of temperature acclimation and exercise on their morphological and biochemical characteristics. Histochemical techniques were used to determine total lipids, glycogen, and cytochrome oxidase. Hematoxylin and eosin are used for gross morphology of the tissues. The white muscle is composed of large fibers which are low in total lipids and cytochrome oxidase. Red muscle is high in lipids and cytochrome oxidase. More sensitive biochemical methods are required to quantitatively differentiate between the nearly equal muscle glycogen contents of the two tissues. These results correlate well with our electronmicroscopy observations of the fine structure of the red and white muscle of fish.

Microbiology

Pretreatment of germinating conidia of N. cross with L-tryptophan enhanced the uptake of labeled L-tryptophan. Similar pretreatment of the conidia with indole also enhanced tryptophan uptake. Pretreatment with D-tryptophan did not affect the uptake of the L-labeled form. These results indicate an inducible mechanism for the uptake of this amino acid.

Terrestrial Ecology

Annual goose-nesting surveys were begun. Initial results indicate that a lower population of nesting pairs and considerable predation on Locke Island, site of about 35% of the nesting in past years, will result in an all-time low production of young.

Alaskan Studies

Anaktuvuk Pass residents were counted during early April and Cs¹³⁷ body burdens were nearly the same as during January. Average values in 25 control adults were 650 $\pm$ 160 nCi during April and 640 $\pm$ 190 nCi during January.

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The small whole-body counter was calibrated with the shadow shield counter and was used to measure Cs¹³⁷ body burdens at Arctic Village and at Ambler. Average values in adult men were 50-70% greater than values obtained last summer and reflect the greater utilization of caribou flesh. Attempts to go to Nunivak Island were unsuccessful due to weather conditions.

W. J. Bair
BIOLOGY DEPARTMENT

WJB:sw

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APPLIED MATHEMATICS DEPARTMENT
MONTHLY REPORT - APRIL 1965

ORGANIZATION AND PERSONNEL

Mr Eugene F. Witeck joined the Statistics Section on April 15 as a Senior Scientist.

ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

A lecture on applied statistical techniques was delivered before engineers from the Fuels Section.

Irradiation Processing Department

An extensive set of single-stage sampling plans was developed for use in procurement of reactor hardware.

Run-to-rupture experiments are to be conducted to obtain comparative data on the HDS and AlSi fuel elements. A simplified decision rule was proposed for selecting the preferred fuel type.

The necessary number of experimental units was determined for an experiment designed to assess the relationship between irradiation time and warp of certain fuel elements.

An analysis was completed on experimental data concerning the effects of certain bonding procedures on irradiation-induced distortion of fuel elements.

Investigations continued on mathematical models for stratified fluid flow from large reservoirs

Chemical Processing Department

Debugging of the EDPM program for the mathematical model of the proposed chemical counter current crystallization process is well underway. Numerical problems have forced the use of some double precision routines.

ACTIVITIES WITHIN BATTELLE-NORTHWEST2000 Program

The EDPM program TREND being written for the process control computer application to a C-column is now approximately two-thirds complete. TREND is that portion of the control program which contains the statistical analysis of the sampled data.

A study was initiated to develop an appropriate multi-stage experimental design for investigating the contours of the response surface associated with certain waste solidification variables.

Investigation was begun to obtain a statistical decision rule for the monitoring of the continuous discharge of waste products by the reactors.

An analysis was carried out on experimental data obtained in accordance with a design (1/9 replicate of a 3^6 factorial in blocks of nine) previously recommended. This experimental effort has resulted in the identification of a desired optimal dissolution solvent.

An appropriate experimental design (1/3 replicate of a 3^4 factorial in blocks of nine) has been formulated for the initial stage of a response surface approach to an optimal solvent in a certain dissolution problem. The design was chosen to ensure the property of rotatability in the subsequent quadratic model analysis.

An analysis was completed relative to the sulfate concentration in the Columbia River at Hanford and at Vernita.

4000 Program

Data were analyzed in accordance with a non-linear model relative to the relationship between the extent of cracking and the force applied to certain reactor metals.

Collaboration with Physical Measurements on authoring an expository paper on stress wave transmission and attenuation in visco-elastic media is now complete.

An analysis was completed on experimental data relative to the reliability of a fuel-enrichment tester in discriminating between fuels of different enrichments.

An investigation was continued in estimating parameters in certain non-linear models for chemical data relative to studies of graphite.

An analysis was continued on particle size distribution data.

The analysis was continued on experimental data relative to the Nupac and Vipac subsystems. The second-stage experiment is currently being designed.

5000 Program

The computer program for approximating (via iterative search) least squares estimates of model parameters in the actinide research program was expanded to increase the search capability in new applications.

A modified version of GEM was completed using weights which reflect instrument instability as well as counting statistic variation. Some tests have been made to determine the magnitude of the instability factor as a function of instrument used to count the sample. A preliminary evaluation indicates that the modification will extend the range of the GEM program to give valid standard deviation estimates of nuclide decay rates in the greater than 10,000 count range.

Work continued on the power function calculations for the uniformly most powerful similar test to detect radioactive decay in the presence of random background. The Fortran language program recently completed and now being used to perform the power function calculations is hampered by scaling problems. Modifications are being performed in the Chi-square cumulative distribution function subroutine to eliminate these difficulties. Computation of the critical points for the uniformly most powerful unbiased test of the null hypothesis that the variance of a normal distribution lies in a specified interval is nearing completion. All calculation has been done for three levels of significance and degrees of freedom up to 40. Underflow problems in the Fortran program are currently being investigated so that the calculation can be extended beyond 40 degrees of freedom.

6000 Program

An analysis was continued on survey data concerning the levels of Cs¹³⁷ in Alaskan caribou.

Work was continued toward obtaining an operational computer program to convert counting data from gastrointestinal studies into corrected counts-per-minute for samples with more than one radionuclide present.

An analysis was completed on experimental data relative to the protection afforded by cystamine during irradiation.

An analysis was initiated on inhalation toxicology experimental data relative to the effectiveness of certain agents in removing Cr⁵¹ or Fe⁵⁹ from exposed dogs and rats.

After consultation with Atmospheric Physics, a test study of the EDPM fallout and deposition program is being made. A 30 to 60 day halogen release under specified atmospheric conditions is being simulated.

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An EDPM program to calculate both stream lines and equipotential lines in a two-dimensional ground water flow system has been written, debugged and placed in service.

Other

The analysis was completed relative to GE-Cincinnati data concerning the effect of neutron radiation on a certain hardened alloy.

A sampling procedure was suggested for estimating the percentage of IBM mechanical drawing cards in the AEC Records office which are currently "defective." The desired estimate and a confidence interval were subsequently calculated from the sample data obtained.

C. A. Bennett, Manager
APPLIED MATHEMATICS DEPARTMENT

CAB:lg

RADIATION PROTECTION DEPARTMENT
REPORT FOR APRIL 1965

Organization

Jean K. Lucas accepted a position with Nuclear Safeguards & Engineering effective April 19.

Laboratory Experience

There were two incidents involving two employees which required special bioassay sampling for plutonium analysis. In addition to the incidents involving plutonium there were two incidents involving three persons that required special evaluation for possible intake of other radioisotopes.

A Physics & Instruments Department employee received an unplanned exposure resulting in an estimated 3.5 rem to the right hand during the removal of a graphite tube containing two radium beryllium sources, each of which contained one-half gram of radium.

Environmental Evaluation

The average concentration of P-32 in the flesh of whitefish caught between Ringold and Richland during the month of March was about 60 pCi/g. This was substantially less than the 440 pCi/g measured last year and the range of 100 to 500 pCi/g experienced during the past several years.

The shoreline of the Columbia River was surveyed for radioactivity between Hanford and the Richland Water Plant. The results were generally lower than those measured in March 1962, possibly because the river flow rate was higher and less of the river bed was exposed. One identifiable radioactive particle was found but not recovered. Generally the dose rates were between 50 and 100 μ R/hr. The higher dose rates were found on rocky beaches where the rocks were covered with algae.

Concentrations of fallout materials in the air in the vicinity of the Hanford project averaged 0.4 pCi β /m³ during the four-week period ending 4-16-65. This average reflects the continuation of the annual spring influx of older fallout materials which began in March. The average for the four-week period ending 3-19-65 had also been 0.4 pCi β /m³.

Nuclear SafeguardsPlutonium Recycle Test Reactor

A study was made of the allowable rate of moderator level rise for the short core. The former limiting rate of reactivity addition of 5¢/sec will result in over-long times to raise the moderator. Even at 10¢/sec the time to take the reactor critical will be much increased.

Work is continuing on the PRTR high power density core hazards investigation. Accident analysis associated with the proposed PRTR boron shim pilot tests was presented to the Pacific Northwest Laboratories Safeguards Council in a document coauthored by R. I. Smith, L. J. Nitteberg, and T. F. Demmitt. The document describes the boron shim test system, physics parameters, and possible hazards to reactor operation during the tests at < 20 ppm boron. The proposed test and report were reviewed by the Safeguards Council.

Critical Facilities

A complete file of documents and letters of authorization governing the nuclear safety aspects of operating the Physics Constants Testing Reactor, the Thermal Test Reactor, the Critical Approach Facility, the Plutonium Recycle Critical Facility, and the Critical Mass Laboratory was assembled and forwarded to AEC-RLOO at their request.

BNWM-28, Physical Constants Testing Reactor, Operating Specifications, was revised to include comments of the several reviewers and was routed for final approval.

BNWM-29, Thermal Test Reactor, Operating Specifications, was routed for comment, revised, and then routed for final approval.

The Final Safeguards Analysis of the Critical Approach Facility was reviewed by the Pacific Northwest Laboratory Safeguards Council. The document will be revised in accordance with the Safeguards Council's recommendations.

Fast Flux Test Facility

Assistance to the conceptual design effort for the Fast Flux Test Facility was continued. A tabular comparison of the design nuclear safety provisions of various American fast reactors was prepared. Studies of the hazards of sodium-air reactions were initiated.

Production Reactors - GE

Another study was begun on the consequences of loss of coolant accidents

of the production reactors to determine the degree of confinement required in order that Hanford production reactors meet AEC siting criteria established for power reactors. The general approach to this problem is to estimate the consequences as a function of building leakage rates. In this way, a leak rate can be determined which results in meeting siting criteria at the present project boundaries. This information will then be used to determine building modifications required. Computer runs for nine leak rate conditions are now being analyzed.

Dosimetry Studies

A review of employee exposure records to determine the availability of data for an AEC-sponsored epidemiological study was completed. A summary of the work to be performed and a cost study were prepared for the program director, Dr. Mancuso.

A decision was made between Battelle-Northwest, U. S. Testing Company, and the AEC to transfer the responsibility for make-up, delivery, and accountability of film dosimeters to U. S. Testing Company. A detailed set of procedures and accompanying forms were prepared for review by the AEC. In connection with the transfer of responsibility, assistance was provided to the U. S. Testing Company in the preparation of Beta-Gamma Film Badge Dosimeters for the April monthly badge exchange.

The double moderator used with the BFQ neutron dose rate monitoring instruments was studied with an outer moderator of polyethylene containing 5 weight per cent of boron. An outer moderator of this composition should provide improved direct reading neutron dose rates for this instrument at low neutron energies. With the currently used polyethylene moderators, an increased neutron sensitivity is observed for neutrons below 100 KeV. In the use of the current double moderator dose rate monitoring instruments, a correction factor for this increased response can be determined and the actual dose rate calculated. Calibration charts to provide these correction factors are provided on all neutron dose rate monitoring instruments at this time.

Environmental Studies

Forty-one fishermen were examined in the Mobile Whole Body Laboratory as part of a special off-site study. The fishermen were examined during three days of counting at locations adjacent to Casey's Pond and in Columbia Park. Only one fisherman had results appreciably different than the Hanford employee average.

The effluent from the 100-B reactor was marked with dye on two occasions in order to measure the time required for the effluent to reach downstream points under low river flow conditions. A few more runs of this type (three from the

old reactors and four from N-Reactor) will be carried out after the freshet season in order to complete a series that provides the correlation between travel time and river flow.

Radiological Development

Thermoluminescent dosimeters (TLD's) and silicon diode neutron dosimeters were exposed to reactor bursts of about 7×10^{16} fissions. The irradiations were made utilizing the Oak Ridge Health Physics Research Reactor located at the Oak Ridge National Laboratory. Gamma and neutron dose measurements were made in air and on a burro. The TLD and silicon diode results for the neutron dose measurements were in good agreement.

An experimental, portable, battery-powered film densitometer circuit was developed. The circuit features a meter readout for 3% accuracy and a null and balance network for a 0.1% readout accuracy.

Dose rates through manipulator openings were calculated for the waste calcination cell at the Fuels Recycle Pilot Plant. Source geometry has changed quite drastically since the original design. The radiation source can now be as high as 10 feet above the cell floor instead of 3 feet as originally specified for shielding calculation. This results in a radiation beam coming through the manipulator through-tube. These through-tubes have approximately 8 inches of lead in them for shielding and require about 17 inches to meet the design criteria. The manipulator manufacturer was contacted and additional shielding will be provided. The problem is the moving tapes for manipulator control. It appears that this may be an area where castable shield material could be used.

A study was completed on the exposure rates to be expected in the PRTR fan pit during fuel handling in the new storage basin addition. It was concluded that additional shielding may be required in the fan pit to reduce the radiation to the gaseous exhaust monitors from fuel elements handled above the -12 foot level of the storage basin addition.

Outside Consultations

A total of 288 pieces of personnel dosimeter film was exposed to a variety of x-ray energies and to the PuF_4 and PuBe neutron radiation sources for the Nevada Test Site Radiological Sciences Department. The films were exposed in a new personnel dosimeter under development by the Nevada Test Site group. Complete copies of all working records and doses delivered were forwarded for their information. The dosimeter system utilizes filters slightly different than those provided in the Hanford film badge dosimeter but includes a high dose or criticality evaluation section identical to that developed and in use at Hanford.

Mr. V. O. Milligan and Dr. Best, Reynold's Electrical and Engineering Company, Mercury, Nevada, consulted with H. V. Larson on plutonium wound counting and bioassay evaluation techniques.

P. C. Jerman and W. N. Koop, IPD-GE, discussed an evaluation of the impact of fuel element failures on Columbia River contamination with R. B. Hall. They were provided with data collected by the environmental surveillance program showing the concentration and rate of transport of I-131 and Sr-90 during 1964 and 1965.

A proposed autoclave rupture debris disposal system for the 306 Building was discussed with S. C. Linn of Equipment Development and Testing, IPD-GE. It was concluded that the uranium and thorium concentration in the rupture debris would be low enough for direct disposal to the process ponds. Gaseous effluents from the proposed stub stack are thought to represent no serious problem; nevertheless, an air sampling program will be initiated.

Consulting was provided IPD-GE during a meeting with AEC Division of Production in Washington, D. C., providing information on the consequences of postulated reactor accidents.

Russell Stanford, Edison Institute, New York, N. Y., consulted with C. M. Unruh and L. G. Faust on April 21 on radiological problems associated with a fuel reprocessing facility designed specifically for plutonium processing.

A. R. Keene served as a consultant to the International Atomic Energy Agency, Vienna, Austria, April 12-14, on the final editing of an IAEA Manual, "Environmental Monitoring in Emergency Situations." A brief visit to the Battelle Laboratory at Frankfurt, Germany was made en route to Vienna.

Staff Activities

J. P. Corley and R. F. Foster attended a Public Health Service sponsored symposium on April 14 at Corvallis, Oregon. The subject of the symposium was "Comprehensive Planning for Water Pollution Control." Informal discussions were held with a number of Public Health Service personnel on the status of general programs of mutual interest.

J. P. Corley participated in a planning session of the Industrial Wastes Committee of the Pacific Northwest Pollution Control Association at Portland, Oregon on April 21. A discussion of analytical methods and the possibility of running comparison samples was held with Allen Moore, Public Health Service Water Laboratory in Portland.

W. V. Baumgartner and L. G. Faust visited the Lawrence Radiation Laboratory, Berkeley, California on April 1 through April 5 to perform experiments on

the 88-inch cyclotron.

J. F. Honstead, R. L. Junkins and C. M. Unruh visited the Naval Radiological Defense Laboratory, Hunters Point, California on April 7 to discuss radiological matters of mutual interest. A trip report detailing the discussions was prepared.

W. V. Baumgartner visited the Boeing Company at Seattle, Washington on April 12 to consult on the use of the Dynamitron for special high energy gamma calibration studies. Contract negotiations to arrange Battelle-Northwest use of this equipment are in progress.

B. V. Andersen visited the Washington State University, Pullman, Washington on April 15 to present a seminar on "Air Sampling" to graduate students and faculty of the Sanitary Engineering Department.

G. W. R. Endres visited the University of Washington on April 22 to present a seminar on "Neutron Detection and Dosimetry."

Manager

RADIATION PROTECTION DEPARTMENT

AR Keene:np

REACTOR DEVELOPMENT - O4 PROGRAMPLUTONIUM RECYCLE PROGRAMPlutonium Recycle Test ReactorOperation

Reactor output for April was 1225 MWD for both an experimental time efficiency and a critical efficiency of 66%. There was one operating period during the month which was terminated manually to begin a scheduled outage. This operating period, which started at 0719 on March 23, 1965, and ended at 0045 on April 18, 1965, for a total run time of 617 hours and 26 minutes, established a new longest continuous run record for the PRTR. A summary of the fuel irradiation program as of April 30, 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			3	1218.5	76	17666.5			79	18885.0
Maximum				431.0		479.8				
Average				406.1		232.4				
In Basin	7	572.5	30	3806.1	64	10201.2			101	14579.9
Buried							1	7.3	1	7.3
Chem. Process.	68	5465.3	35	1965.3					103	7431.6
Program Totals	75	6038.3	68	6990.5	140	27867.7	1	7.3	284	40903.8

(Note: MWD/Element x 20 $\approx$ MWD/TU for UO₂ and UO₂-PuO₂.)

Heavy water loss and indicated helium loss for the month were 1162 pounds and 132,082 scf., respectively.

Equipment Experience

A total of 2086 craft manhours were utilized during the April reactor outage as follows:

Repair	247 manhours
Modification and Improvement	562 "
Operations Assistance	314 "
Preventive Maintenance	363 "

Total preventive maintenance effort for the month was 452 manhours, or 9.8% of available assigned manhours.

Improvement Work Status (Significant Items)Work Completed

Access Manholes for Containment Vessel - "A" Cell
Air Lock Door Operators
PRTR Steam Utilization
Creep Test Facility
Relocate Rupture Loop to Center of Reactor
Instrumented Fuel Element Thermocouple Installation

Work Partially Completed

Corrosion Loop Installation
Modification to PRTR Warehouse 3718-C
Additional Fuel Storage and Examination Facility
Flux Wire Scanning System
D₂O Cleanup Facility
Pneumatic Irradiation Facility
Boric Acid Prototype Facility

Design Work Completed

Sump Filter High Pressure Helium Compressor No. 2
"B" Cell Instrument Tubing Penetration
H-80 Control Modification

Design Work Partially Completed

Instrument Power Supply
PRTR Experimental and Building Facility Addition
PRTR Increased Power Level
PRTR Vacuum Deaerator

Process Engineering and Reactor Physics

Eight of the 58-inch UO₂ - 2 w/o PuO₂ elements are under irradiation, including #6500 (Salt Cycle) and #6502 (Thermocoupled element). The maximum exposure of an HPD element as of April 30, 1965, was 113.1 MWD (3412 MWD/TU). The exposure of element 6500 (Salt Cycle) was 21.4 MWD (664 MWD/TU). The power generation on the standard HPD elements continued in the 1000 - 1350 KW range.

PRTR Test Number 79-2, Reactivity Worth of Voiding the PRTR FERTF Channel, was performed during April. Loss of H₂O coolant from the FERTF, unfueled, with the channel in location 1550, resulted in a reactivity increase of ~3.4 mk. Loss of H₂O from the FERTF, fueled with an HPD

element, resulted in a reactivity increase of ~ 0.5 mk. In both cases, loss of coolant produced a reactivity increase. Because of the size of the increase for the unfueled case, it is not planned to operate the reactor at power with the FERTF unfueled in its present location.

Reactivity measurements made as a part of PRTR Test 99, In-Reactor Creep Measurements, established that the reactivity worth of the creep capsule and thimble, dry, was about 1.34 mk and the reactivity worth of the coolant in the assembly was about 0.08 mk. Loss of coolant from the creep capsule, therefore, will not create a reactivity hazard.

BNCC-47, the Use of Soluble Poison for Reactivity Compensation in PRTR, was issued for review by the BNW Safeguards Council. Permission was given to proceed with the Phase I work, concurrent with a continuing study of the corrosion problems (if any).

PRTR Test Number 113 (LWI Capacity Test) was completed. The diesel well pump and the supplemental emergency pump provide a flow of 835 gpm at a pressure of 145 psig at Valve E-7. Tube power limits are being recalculated on the basis of this flow and pressure.

Performance of PRTR Test Number 117 (Linear Flux High Level Trip Settings) was begun. One of the three linear channels is being operated with the trip set 10 percent above the indicated flux rather than the previous 30 percent. Little difficulties have been experienced to date.

PRTR Test Number 119 (Increased Power Level) was prepared and approved. Reactor power level was increased to 74 MW before temperature and equipment limits were reached.

Experimental Reactor Services

The status of the various test elements at the end of April 1965, is shown below. Those elements discharged prior to April 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		296.6
14	Basin	5099	Vipac-0.48%	5/8/62	4/18/65	306.0
40	1156	5150	0.48% ($\frac{1}{2}$ " x $\frac{1}{2}$ " pads)	8/1/62		305.2
54	Basin	5116	0.48% (clip-on pads)	5/8/62	4/18/65	314.6
54	1554	5118	0.48% (clip-on pads)	5/8/62		479.8
61	1653	5136	0.48%-Physics	5/28/63		317.2
61	1651	5192	0.48%-Physics	6/13/63		325.7
67	1047	5117	0.48% (repaired wire)	10/20/63		277.2
72	1253	5253	1% (Zr coupons)	9/1/64		189.0
85	1855	5230	Vipac - 1%	1/30/64		248.8
108	1954	6501	EBWR Cluster	1/30/65		27.0
118	1946	6500	Salt Cycle, 2%	3/16/65		21.4

Seven fuel elements, including four HPDC fuel elements, were visually examined in the storage basin. No significant changes were observed. Fuel Element 5186 had a broken rod wrap repaired and is now in the PRTR.

Fuel Element Rupture Testing Facility

FERTF Test #6, Test Irradiation of a Salt Cycle Fuel Element, was completed during April. Examination of the element showed areas of white corrosion product at the ends of the fuel regions and a brown deposit on the lower bracket surfaces and on the first inch of the cladding surfaces. No damage was apparent and the element was recharged under PRTR Test #118 to continue the irradiation in the main coolant loop.

FERTF Test #13, Irradiation of a 1600 KW Termocoupled HPD Fuel Element, was begun at month's end, with the FERTF channel in location 1550.

Efforts to improve the flow and temperature instrumentation on the FERTF have met with some success. A venturi on the inlet of the in-core section now provides flow data as precise as the normal PRTR equipment and supplies the data directly to the data handling equipment. A calibrated matched pair of strap-on RTD's, carefully installed to minimize environmental effects, now measure the inlet and outlet coolant temperature to obtain a more precise determination of the temperature increase across the test section and supply the data direct to the data handling system.

Document BNWL-40, PRTR Fuel Rupture Test Facility Capabilities and Project History, by P. C. Walkup, was issued in April.

Chemical Processing of Spent PRTR Fuels

The chemical processing campaign again has been cancelled.

TECHNICAL SHOPS SECTION

A total of 22,745 hours of shop work was produced with a backlog at the end of the period totaling \$206,225 or 22,914 hours. To meet emergency requests for service, it was necessary to work 1,579 overtime hours or 5.9% of the total hours produced in Technical Shops.

LABORATORY MAINTENANCE SECTION

Total productive time was 23,400 hours of 24,900 potentially available. Craft overtime worked during April was 2.8% of the total available hours. Manpower utilization (in hours) was as follows:

A. Shop Work	1 700
B. Maintenance	7 800
1. Preventive Maintenance	2 100
2. Emergency or Unscheduled Maintenance	1 200
3. Normal Scheduled Maintenance	4 500
C. R&D Assistance	13 900

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>
D. F. Carroll and J. B. Burnham	Fabrication of Nuclear Cermet Fuels
L. A. Bray	A Process to Remove Zirconium from Purified Strontium-90
W. J. Van Slyke	Chemical Processing Equipment: Specifically, an Improved Semi- Continuous Ion Exchange Contactor
W. W. Schulz and S. J. Beard (GE)	A Wash Treatment to Restore the Quality of Degraded D2EHPA-TBP Fission Product Extractants.
L. T. Pedersen	Glue Carrier

DATE

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10/24/94

END