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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT

OCTOBER, 1963

NOVEMBER 15, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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HANFORD LABORATORIES
MONTHLY ACTIVITIES REPORT
OCTOBER 1963

DECLASSIFIED

By Authority of CG-PR-2
Robert M. Iten 8/25/92
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PM Eck 10-28-92

Compiled by
Section Managers

November 15, 1963

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRELIMINARY REPORT

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

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Table I - Hanford Laboratories Force Report

Date: October 31, 1963

	At Beginning of Month		At Close of Month		
	Exempt	Salaried	Exempt	Salaried	Total
Chemical Laboratory	144	126	145	131	276
Reactor & Fuels Laboratory	201	187	202	192	394
Physics & Instruments Laboratory	100	73	100	76	176
Biology Laboratory	42	63	43	64	107
Applied Mathematics Operation	18	5	18	5	23
Radiation Protection Operation	44	93	43	94	137
Finance & Admin. Operation	150	117	154	116	270
Programming Operation	16	3	17	2	19
Test Reactor & Auxiliaries Oper.	58	314	62	314	376
General	3	4	4	4	8
TOTAL	776	985	788	998	1786

BUDGET AND COST SUMMARY

October operating costs totaled $2,774,000, an increase of $77,000 over the previous month; fiscal year-to-date costs are $10,689,000 or 33.1% of the $32,269,000 control budget. Hanford Laboratories' research and development costs for October compared with last month and the control budget are shown below:

(Dollars in thousands)	COST Current Month	Previous Month	To Date	Budget	% Spent
HL Programs					
02	$ 79	$ 80	$ 318	$ 1 180	27
03	51	40	158	250	63
04	1 224	1 196	4 638	13 485	34
05	127	127	496	1 456	34
06	269	275	1 067	3 604	29
08	9	9	48	100	48
	1 759	1 727	6 725	20 075	33
Sponsored by					
NRD	173	174	635	1 761	36
IPD	58	68	245	660	37
CPD	137	108	463*	1 668	28
Total	$2 127	$2 077	$8 068	$24 164	33%

* Includes $12 thousand year-to-date adjustment for charges transferred to other accounts

RESEARCH AND DEVELOPMENT

1. Reactor and Fuels

Up to 0.5% of the tritium generated in lithium-aluminum target elements was released from the target alloy during irradiation for 2 months at N-Reactor conditions.

On drop-testing an N-fuel assembly which had been irradiated to 2600 Mwd/ton, the outer tube suffered a brittle transverse fracture, but the inner tube did not appear damaged.

[redacted]

Capsules have been prepared to evaluate the swelling resistance of Zircaloy-clad uranium metal containing a submicron size dispersion of uranium carbide.

Aluminum-clad, simulated target elements containing Al−1 wt% Li cores are being fabricated for cold physics tests in N-Reactor.

Irradiation tests with NaK capsules have provided data on the strain capability of Zircaloy-2 cladding as a function of clad thickness uniformity, temperature, and exposure. Below an average clad temperature of 325-350 C, the clad strain limit is approximately 1.5% before necking occurs; however, this cladding strain limit increases above 350 C. The strain limit is less than 0.5% if the cladding is striated.

Testing was continued in the K-1 Loop to evaluate use of NH_4OH as a substitute for LiOH to adjust pH of N-Reactor primary coolant. The crud levels with NH_4OH are low and no turbidity of the water was noted.

Preliminary tests indicate that peracetic acid can be used without peroxide to dissolve UO_2 and/or U from rupture debris.

The long-term irradiation of N-Reactor graphite continues satisfactorily, with some samples having attained an exposure of 7×10^{21} nvt.

The addition of 1/2% CF_2Cl_2 reduces the rate of air oxidation of CSF and EGCR graphites by factors of 3 and 4, respectively.

Preliminary information was obtained on the effect of heat transfer on corrosion of aluminum in 120 C process water. At a heat flux of about 70,000 Btu/(ft^2)(hr), the corrosion rate was increased by a factor of 3.6.

Examination of PRTR fuel elements showed some gray scuff marks under the wires of the three Al-Pu elements that had been charged into the reactor without the usual preirradiation autoclaving. The spacing wires on these elements are now loose, a feature common to all Al-Pu alloy PRTR elements.

[redacted]

Two UO_2-PuO_2 mixed oxide fuel elements with broken rod wire wraps were repaired in the PRTR basin.

A new Model 1220-B Dynapak machine was successfully used to impact 15 lb cans of UO_2-1 wt% PuO_2 for PRTR fuel elements. More than 300 lb of fuel were impacted on the first day of trial operation.

The failure mechanism of a recent vibrationally compacted PRTR mixed oxide fuel element has been identified as gross hydriding which is apparently not associated with gross corrosion (i.e., oxidation), or high fluoride contamination. However, since other PRTR fuel failures may have been related to fluoride contamination, rapid analytical methods have been put in use to scan all materials used in fabrication of PRTR fuel elements for possible sources of halide contamination of the fuel. Several potential sources have been identified and eliminated.

Studies are continuing to determine the optimum conditions of time, temperature, and humidity for purifying PuO_2.

A 4-rod fuel element with two rods containing sintered UO_2 pellets enriched with coaxial, Pu-15 wt% Zr alloy wires, was irradiated to 2×10^{19} fissions/cc, generating a maximum heat flux of 122 w/cm^2.

Ceramographic examination of an irradiated simulated rejuvenated fuel rod showed that the central zirconia tube remained intact and the enriched uranium oxide contained within the tube relocated radially to form a central cavity. A second rejuvenation would be feasible.

Twenty capsules containing hydrogen-sintered ThO_2-PuO_2 pellets in five compositions ranging between 2.23 and 18.48 wt% PuO_2 are being irradiated.

The eleventh PRTR pressure tube discharged had a 20 mil deep flaw. This flaw has been caused to propagate through the tube wall by a pressure of 9200 psig at room temperature. This pressure is more than eight times greater than PRTR operating pressure.

[redacted]

[redacted]

Evidence shows that reactor exposure damage is saturating in the PRTR Zircaloy pressure tubes (i.e., metallurgical properties are now changing more slowly with exposure). Consequently, the rate of removing tubes for destructive examination will be reduced from approximately one every 2 months to one every 6 months (the actual removal rate will be based on Mwd of operation).

Since May 1963, there have been progressively fewer cases of high zirconium concentration in the PRTR primary coolant D_2O, possibly due to the increasing numbers of fuel elements with wide support pads.

Analysis of air flow data taken in the PRTR fuel examination facility indicates that the allowable power level of fuel elements under examination can be increased from the present level of 1 kw to 5 kw.

The volume of microcracks in high quality as-cast plutonium was reduced, and the resulting metal density was increased, by rapidly cooling through the beta-to-alpha transformation during casting.

The deformation of plutonium during alpha-to-beta and beta-to-alpha transformations was determined to increase from zero at 100 psi to 2.5% at 1600 psi at an average transformation rate of 1-2%/hr. The transformation strain was very nearly a linear function of applied stress.

Strain-rate sensitivity experiments with polycrystalline molybdenum have given results which suggest that strain-rate cycling has been conducted between regions of different stress-strain rate dependence and thus can lead to erroneous conclusions. This observation may have far-reaching implications on the validity of experiments in deformation rate theory.

A series of experiments has verified the existence of four distinct regions in the curve of room temperature yield stress vs. log strain rate for polycrystalline molybdenum.

It has generally been accepted that thorium is much more resistant to fission gas induced swelling than is uranium. Recent studies have

[redacted]

revealed that when thorium and uranium are postirradiation annealed at the same homologous temperature (i.e., at the same fraction of the absolute melting temperature), the same volume increase is observed.

A micro ion cross section detector has been found very sensitive to contaminants in helium, suggesting its application to continuous analysis of total impurities in the ATR loop helium coolant.

Forming and heat treating conditions which cause severe grain growth in the nickel base superalloy, Hastelloy X, have been discovered. A duplex heat treatment by which the grain growth can be avoided has been recommended.

Defect testing of Zircaloy-2 clad, Th-2.5 wt% U-1 wt% Zr rod specimens in 300 C water is continuing. Samples from the lead end of the coextrusion show more rapid corrosion attack than did samples from the middle or rear of the extrusion. This behavior is apparently associated with lower bond strengths.

The effect of increased zirconium and uranium additions on the structure, fabrication, and defect corrosion behavior of thorium base alloy fuels is under study. Alloy compositions containing 2.5 and 5.0% uranium and 0-10% zirconium are being double vacuum arc melted to prepare coextruded billets.

Replicas for electron microscopy have been successfully obtained on PuC-Ta cermets.

Two additional samples of plutonium nitride have been prepared for irradiation in the MTR. There are now three samples of PuN, one of PuC, two of PuO_2, and one of beta Pu_2O_3 under irradiation. A UN-20 wt% PuN sample is being prepared for irradiation under power reactor conditions.

Measurements of the thermal conductivity of single crystal UO_2 have produced the first experimental evidence of an increase in conductivity above 1500 C.

A photogrammetric image rectifier has been installed to provide optical correction for the one dimensional foreshortening of reflection electron microscope images.

The induction plasma torch is now capable of continuous operation for periods of several hours, and small single crystals of alumina have been grown.

Irradiation of previously reported W-50 wt% UO_2 and Mo-50 wt% UO_2 cermet rods continued.

Tungsten-cladding tubing can be made by spark discharge machining with a structure relatively unchanged from that of the original tungsten rod.

A molybdenum-clad UO_2 fuel rod was successfully irradiated to generate 2.0×10^6 Btu/(hr)(ft^2) surface heat flux.

A 1/8 in. hexagonal honeycomb grid of 80 vol% $W-UO_2$ cermet was fabricated directly by pneumatic impaction.

Thorium oxide material was pneumatically impacted, crushed, sized, and vibrationally compacted into aluminum cans for reactor testing purposes.

The flash method for measuring thermal diffusivity has been satisfactorily extended down to a temperature of -195 C, and measurements have been made on TSGBF and TSX graphites down to this temperature.

At the request of AEC-DRD, a one-month study is being made of the potential for reducing the size of the MCR, a 15 Mwt fast compact reactor, by using plutonium fuel. Using a similar pin-type core, it appears that core size could be reduced from 15 in. to about 9 inches. Total weight reduction would be about 8900 lb, or 21% of the reactor and shield packages. Use of alternate plutonium fuel designs might reduce the core dimensions to between 7 and 8 inches. However, requirements for fuel burnup, heat transfer, hydraulics, and reactor control increase in severity as the core size decreases.

2. Physics and Instruments

Absolute k_∞ values have been determined with PCTR experiments for KVNS fuel elements in Zircaloy-2 process tubing in the K-lattice for both wet and dry cases. Results will be reported in HW-79325.

Preparation is proceeding on schedule for measurement of lattice physics parameters of N-Reactor during the forthcoming startup tests.

N-Reactor startup test instrumentation also is progressing on schedule. Water-cooled thimbles for cooling in-core detectors are under fabrication. HTR experiments have been made to substantiate estimated neutron flux levels which will emanate through perforations in the N-Reactor biological shield. KE-Reactor tests of one fuel rupture monitor gamma spectrometer showed definite increase of N^{16} activity with changes to water sampling system; however, no photopeaks were observed.

N-Reactor primary coolant simulations have established approximate controller settings for actual system startup. These studies showed that the best overall system performance is not obtained with each controller optimized with respect to its individual loop. Coolant flow disturbances and temperature fluctuations accompanying loss of primary pumps have been determined.

Good progress was made on instrumentation for the fuels testing loop at PRTR. Flow control equipment is installed, the delayed neutron and gamma spectrometer systems are on site, special preamplifiers have been fabricated and tested, and installation of electrical power services and instrumentation racks is nearing completion.

Continuing critical mass experiments with PuO_2-polystyrene compacts mounted in seven separate configurations on the Split Table Machine have provided good data on the reflector savings of Lucite, and have elucidated the effect of core shape on criticality.

[redacted]

First field use of the pulsed neutron method of measuring k_{eff} of subcritical systems was achieved with measurements at an underground waste crib and a storage igloo near 234-5 Building. Data processed to date indicate successful results.

Monte Carlo calculations have given the interesting result that an unmoderated Pu^{240} system has a larger k_∞ than a similar U^{235} system, thus inferring that a Pu^{240} critical sphere would have less mass than one of U^{235}. However, these results are very sensitive to the cross sections used and must be considered tentative.

Studies of graphite-moderated uranium and plutonium systems with the Horowitz modified gas model have been compared with studies of such systems with the more exact neutron balance equation. The modified gas equation produces reasonable results over large ranges of uranium concentrations and moderate ranges of plutonium concentrations. In other neutron thermalization studies, a report was written describing the improvement in scattering law calculations by treating exactly two resonances of the spectral function. Work continued on the RBU thermalization model.

Work on Phoenix fuel concept continued with initiation of studies on Be-Pu systems, and indications are that a harder spectrum than that in the present design is needed for Phoenix action to be discernible in the MTR. Other reactor studies included analysis of the SNAP-50 fueled with plutonium, evaluation of various proposals for a special purpose power source, and N-Reactor utilization studies.

The gamma scan at the Radiometallurgy Laboratory is being used to obtain fine detail on plutonium burnup within a Pu-Al rod. These data will be used with other data for studying the buildup of the plutonium isotopes as a function of position in the rod. Two-dimensional analysis of the ΔK measurements in the PRCF on the irradiated Pu-Al PRTR elements is being carried out.

[redacted]

An extensive study is being performed on the effect of variation of physics parameters on the accuracy of fuel cycle analysis. Some preliminary results indicate that η and α of the fissile species are particularly important parameters in determining the initial reactivity and lifetime of a reactor core.

Code development continued on a number of different codes including HFN, HRG, BARNS, RBU, chaining of some codes, and modification of others. The HRG can now be used for calculating effective resonance integrals of heterogeneous, plutonium-fueled configurations.

Approach-to-critical measurements were completed on 2.0 wt% Pu-Al (16% Pu^{240}) fuel rods and the results will be reported in HW-79054.

PRCF measurements on three irradiated fuel elements show their reactivities to be 63, 31, and 15% of that of an unirradiated element for the 30, 56, and 90 Mwd exposure elements, respectively.

PRTR automatic controller tests in which the reactor was simulated by analog computer have shown the system to be stable and well regulated. Moderator levels correctly and rapidly changed in response to transients purposely fed into the system at simulated reactor power levels ranging from 10 to 70 Mw.

Experimental studies are under way to evaluate materials compatibility problems at the elevated operating temperatures planned for the HTLTR. Feasibility calculations also were made regarding methods for reducing the temperature coefficient of reactivity of the reactor.

Chemical separation of fission products produced by the irradiation of experimental regenerating neutron flux detectors is proceeding. These data will help to establish the validity of key analytical procedures used to establish optimum detector designs. A second prototype B^{11} neutron detector is being fabricated for continuing reactor experiments.

The possible applications of ultrasonic boundary waves to practical inspection problems are becoming more promising in light of recent laboratory investigations. The existence of a lateral shift in an ultrasonic beam reflecting from a solid surface at large incident angles has been positively established. Normal zirconium gives rise to a measured 2.45 mm shift relative to aluminum; Zircaloy hydrided to 2000 ppm increases this shift by 0.30 mm. These shifts are subject to sensitive measurement by interferometric techniques. Signal amplitudes which change by a factor of 3 are observed on samples with 600 ppm hydrogen as compared to normal metal.

Studies of the cooking of reindeer meat and of the uptake of Cs^{137} from reindeer meat by people were made to see if they suggested that the Cs^{137} burdens of Eskimos counted in 1962 were at a stable level or not; apparently they were not.

Backgrounds were determined for the plutonium scintillation counter with people in it; the backgrounds were variable but may be predictable from other data taken during the count.

New calorimetric measurements of Pm^{147} are being made to measure its half-life.

The field model of the zinc sulfide real time sampler, a sensor which measures the concentration rather than exposure values, was successfully operated during two diffusion experiments in October. These tests show that the sampler can detect concentrations as low as 4×10^{-7} g/m^3, an order of magnitude lower than that detected by the prototype tested earlier this year. Data that will be obtained with this equipment will permit studies related to acute toxicity problems.

An instrument which counts insects as they crawl through a tiny glass tube has been developed and placed in operation at the Biology Laboratory.

[redacted]

Preparations were completed for a complete eddy current testing of the installed N-Reactor process tubes to provide preirradiation reference data to which later process tube integrity test data can be compared.

The ultrasonic translator leak detector is being successfully used to detect gas leaks in lead covered telephone cables. Promising results also were obtained with this device to detect malfunctioning components in heavy rotating machinery by the abnormal sounds they generate during operation.

3. <u>Chemistry</u>

In reactor tests, data were obtained which show that P^{32} in the effluent water is principally formed by the (n, p) reaction on sulfate-bearing water treatment chemicals.

Substitution of the present sulfate water treatment system by a nitrate system appears to be a practical process for reducing the P^{32} content of reactor effluent water by a factor of 2.

The reaction, $Th^{232}(\gamma, n)\ Th^{231}$, was found to contribute to the formation of U^{232} during thorium irradiation as does the $Th^{232}\ (n, 2n)\ Th^{231}$ and the $Th^{230}(n, \gamma)\ Th^{231}$ reactions.

Application of multidimensional gamma spectrometry was applied to the analysis of U^{232} in the presence of U^{233} and fission products; 1 ppm U^{232} can be readily measured in 100 μg of U^{233}.

The buildup of potentially troublesome isotopes in recycled thorium was investigated; it was determined that several thorium isotopes, from Th^{227} to Th^{234}, with half-lives ranging from 18 days to 7300 years will be present in irradiated natural thorium. It was also calculated that the neutron yield (α, n reaction) from U^{233} containing 1 ppm of U^{232} is less than that from Pu^{239}.

Experimental studies on the stability of uranium(IV) in Purex solvent indicated that 40 to 60% of the uranium(IV) is oxidized to uranium(VI) in 5 hr. Hence, a long standing-time between extraction of uranium(IV) into the solvent and introducing the solvent into the 1BX column is undesirable.

Measurements of the extraction of BAMBP from solvent solutions indicate that either sodium hydroxide or sodium carbonate can be used as a wash for the CSREX solvent without serious loss of BAMBP.

Based on calculations and experimental heat transfer studies, a finned surface has been incorporated in the design of the annular circulator to be used for in-tank solidification of intermediate level plant waste.

The radiation stability of Duolite C-3 resin loaded with about 1 curie Cs^{137}/g of resin under flowing conditions was demonstrated to be good. Following an exposure equivalent to 1-2 years of plant usage with Redox SX supernatant waste, the resin retained 85% of its original capacity.

The pilot plant for processing radioactive condensate waste (Purex Tank Farm condensate) went "hot" this month. An unidentified solid previously not present in this waste solution forced early termination of the first run.

Soil column tests indicate that neutralization of Purex process condensate with sodium hydroxide to pH 9 markedly improves soil removal of Sr^{90} even after a significant amount of acidic waste has passed through the soil column.

The last link needed for careful evaluation of possible numerical errors in the permeability calculation method for the electrical analog of ground water flow was obtained.

Reduction of heat transfer rates in titanium tube bundles may be due to deposition of silica; current laboratory tests indicate no buildup of silica in simulated evaporator bottoms containing 8$\underline{M}$ nitric acid, in contrast to substantial deposition under similar conditions, but with 10$\underline{M}$ acid.

[redacted]

Effects of gamma radiation on the neutron counting ability of BF_3 ionization chambers were measured; results of the measurements provide an explanation for previous failure of a neutron monitor installed in Purex. A modified installation in a lower gamma field is expected to solve the problem.

Alpha counting yields and beta-gamma discrimination of scintillating glass alpha detectors have been measured to evaluate their usefulness for in-line plutonium monitoring.

Cold testing of the salt cycle process equipment in C-Cell of the High Level Radiochemistry Facility was successfully completed. Nine, half-length, unirradiated PRTR fuel elements were declad, the UO_2 was oxidized to U_3O_8 and dissolved in molten LiCl-KCl, and a 24 lb batch of UO_2 was electrolytically deposited and recovered in a form suitable for refabrication by vibratory compaction.

In a single batch contact of 0.3M trilaurylamine-Soltrol with an equal volume of a solution simulating dissolved PRTR spike fuel elements, 95-99% of the plutonium was extracted.

Experimental studies were conducted with the "cold" 18 in. diam spray calciner to determine internal recirculation rates and heat duties as functions of the draft tube geometry, throughput rate, and wall temperature.

The continuous powder melter was operated with calcined synthetic Purex waste containing added phosphate, lithium and calcium. No foaming or pressurization of the vessel was observed, although some problems were encountered with the automatic powder feed system and with plugging of the outlet valve.

Pilot plant studies of the cesium purification process proposed for the Hanford Isotopes Plant were successfully completed. In the final series of runs, using BAMBP solvent extractions, cesium losses were consistently less than 0.6% in the extraction column, but were considerably higher in

[redacted]

the stripping column. Overall sodium and potassium decontamination factors as high as 13,800 and 1000, respectively, were obtained.

4. Biology

The concentration in the Columbia River of the reactor tube cleaning material, Vertan 690 after N-Reactor Department people dumped in some 120,000 gal of the mix on October 2, was higher than that expected. Concentrations lethal to fish appeared as far downstream as 150 yd from the outfall.

It appears that the resistance of fish to the columnaris disease increases with age of the fish, possibly due to an antigen-antibody response.

Although Zn^{65} concentrates in the gill filaments and the GI tract of trout following short-term exposures, 100 days after Zn^{65} administration, the gut was still the highest in Zn^{65} concentration, but the gills were quite low.

A test which may turn out to be a sensitive indicator of radiation damage due to bone seekers is being developed in our tests of the chronic toxicity of Sr^{90} in pigs. This involves the response of an animal's neutrophils to the injection of an endotoxin.

Our laboratory studies on I^{131} transfer using dairy cattle were concluded this month. Until suitable facilities (now in the planning stage) become available only limited field and short-term studies can be accomplished with dairy cattle. The need for additional information, however, remains acute.

Although it was shown that DTPA can effectively remove cerium oxide from the lungs of dogs and rats, the material is effective only when the oxide is prepared at room temperature. When it is prepared by calcination at about 400 C, DTPA is ineffective as it is in removing plutonium dioxide similarly prepared.

[redacted]

Program commitments require our maintaining a colony of about 150 dogs, one-half of which are on long-term experiments. Crowded conditions in the colony are responsible for increasing numbers of dog fights which can seriously affect our studies. This is becoming of increasing concern since the 60 new dog runs approved for construction is now scheduled for completion in September 1964.

Male rats which survived 150-275 rads of neutron radiation produced no offspring when mated with control females. However, when similarly irradiated female rats were mated with control males, half the irradiated females produced offspring.

Successful treatment of radiation damage by injecting bone marrow into the irradiated individual has never been proven. However, this work has stimulated research into the mechanisms by which an animal will respond to a graft from another animal. Work in this laboratory is contributing to understanding the complex theories of tissue interactions that occur when "chimeras" are produced.

5. Programming

Analysis of fuel exchanges among large reactor combines have been initiated. Studies are being made of the French and British efforts, as their relative large planned economies provide a framework for analyses. It is one of the objectives of the Hanford work to show that a system operating on minimum costs is superior to the highly regimented system, or at least, not catastrophic as regards conservation of uranium.

Plutonium-seed U^{238} blanket studies are continued. The comparison of results of our rather simple computer model with far more sophisticated codes indicates that our model will be quite satisfactory for the initial phases of our evaluation.

If Cm^{244} and Pu^{238} have significant values (\$100 to \$1000/g) their precursors will have enhanced values. Calculations were completed showing that the significant Cm^{244} prices will increase the values of the precursors as far back as Pu^{239} and, in the case of Pu^{238}, as far back as Np^{237}, with U^{236} values only increased nominally.

Initial studies were completed of U^{233}-Th breeders fueled with solid thorium. It is shown that the breeding ratio is, at best, nominal (in view of processing losses) and passes through a maximum as a function of exposure. At too low an exposure, the Pa^{233} has not decayed enough to U^{233} to contribute positively to the neutron economy. At high exposure, neutron absorption in fission products overrides the absorption in fertile fuel. The apparent breeding ratio decreases as the specific power of the reactor increases; however, the doubling time goes through a minimum as specific power is increased because the fuel inventory is reduced with specific power.

Plutonium enriched thorium (nonbreeder applications) was studied with emphasis on the reduction of the thorium concentration in the fuel. It appears that, with plutonium high in Pu^{240}, thorium fuel attains best performance with reduced effective thorium concentrations, as does U^{238}.

AEC approval was granted for increasing PRCF power level from 100 w to 10 kw and for operation of the PRTR Fuel Element Rupture Testing Facility with a standard PRTR pressure tube.

TECHNICAL AND OTHER SERVICES

A thyroid burden of 70 pc of I^{131} was measured October 19, 1963, in a $3\frac{1}{2}$ yr old boy living at the West Richland farm where the maximum I^{131} concentrations were measured in milk samples collected after the recent Purex I^{131} release. The thyroid burden of this boy's 8 yr old sister was <30 pc I^{131}. The dose for the boy (~4 g thyroid) resulting from this I^{131} emission is calculated to have been 0.03 rem (about 2% of the applicable annual guide). It is probable that this represents the maximum offsite

[redacted]

thyroid dose attributable to the unplanned release of 65 curies of I^{131} during the week of September 2, 1963.

An IPD operator received an unplanned whole body exposure as measured by the film badge dosimeter of 0.5 rems gamma and 1.2 rems beta during spline removal operations at the 105-B Reactor on October 27, 1963. The operator used a long pole in an attempt to dislodge a spline can containing an activated spline from the spline coiler apparatus. When this was ineffective, the operator opened the door to the spline coiler apparatus. He discharged both the spline can and the broken piece to the tank by striking them with a wrench. The dose to the employee's hand was estimated to be approximately 15 rems. This calculation was based on data from the employee's film badge dosimeter and facts brought out at the IPD investigation.

One new plutonium deposition case was confirmed by routine bioassay analysis during the month, the body burden for this employee, and operator in Fission Products Processing Operation, was estimated to be 1% of the maximum permissible body burden. The MPBB for plutonium (bone as reference) is 0.04 μc. The total number of individuals who have internal plutonium deposition at Hanford is 325 of which 235 are currently employed.

Work has begun on designing an information system for maintenance activities connected with B- and C-Reactors. This system will encompass recording, transmission, and filing of maintenance data as it is done, and will yield maintenance schedules and reports.

Formulas were developed which allocate total reactor projection to various position zones of N-Reactor.

A statistical analysis was made of the preirradiation dimensional and weight measurements of fuel elements canned under the AlSi and hot-die-sizing processes. The results consisted of the averages, the standard deviations, 95% confidence limits on the mean, 50% and 95% tolerance statements for 95% of the population as well as comparisons of the process variances. The characteristics used were the weight and length, the maximum and minimum outside diameters measured at three points, the warp, and the inside diameters measured at three points.

A statistical analysis was completed on experiments to test whether such variables as the air pressure in bellows, the probolog speed, and the water flow in the tube affect the probolog readings. Eight experiments were run for different wall thickness and annulus values. The results in general indicated that the probolog values depend on some of these variables.

An analysis of R data from nine tubes has been completed. The results, which indicate the nature of the temperature imbalance in a tube as the tubes become corroded with time are to be used as aids in designing tubes with maximum life expectancies. Regression relationships were given for each tube.

Closed form solutions were obtained to a partial differential equation model for ground water flow in a medium of variable diffusivity. These solutions now provide the first opportunity of checking the accuracy and reliability of an EDPM numerical approximation method which has been proposed to handle more difficult problems.

The final draft of an HW formal report, "Fixed Time Estimation of Counting Rates with Background Corrections" is being printed for distribution.

Work progressed on the Monte Carlo and triangular diffusion studies. Several modified versions of the EDPM programs for each have been successfully debugged, the data machine plotted, and the results are now being studied and compared.

The statistical analysis of data from a study to investigate the discrimination against Sr^{90} relative to Ca^{45} as related to the age of miniature swine at the time of radionuclide administration was completed.

A computer program was written to record data and calculate certain factors from data of radioactive particle inhalation studies. The program also fits the percent body burden as a power function of time by method of least squares and calculates 95% confidence limits.

SUPPORTING FUNCTIONS

PRTR output for October was 1248 Mwd, for an experimental time efficiency of 78% and a plant efficiency of 57.5%. There were four operating periods during the month, one of which was terminated for scheduled refueling and planned maintenance, one was terminated manually due to ventilation unbalance, one was terminated by a spurious log N period trip, and one operating period continued through month-end. A summary of the fuel irradiation program as of October 31, 1963, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd	No.	Mwd
In-Core	18	1636.7	0		67	7063.6			85	8700.3
Maximum		100.9				171.8				
Average		90.9				105.4				
In Basin	25	1895.0	32	3815.0	13	382.0	1	7.3	71	6099.3
Chemical Processing	32	2309.3	35	1965.8	—	—	—	—	67	4275.1
Program Totals	75	5841.0	67	5780.8	80	7445.6	1	7.3	223	19074.7

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

A total of 71 reactor outage hours were charged to repair work. Main items were:

D_2O and helium leaks	16 hr
Flash tank repairs	11 hr
Ion exchanger replacement	11 hr
Valves	8 hr
Instrumentation	8 hr

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The October heavy water inventory indicates a loss of 859 lb for the month.

Operation of the PRCF was routine for the month, involving both irradiated and unirradiated PRTR fuel elements with a D_2O moderator. The facility was shut down at month-end for scheduled conversion to H_2O moderator.

Three operating runs in the Fuel Element Rupture Test Facility were completed. The final run went 8 days without interruption and survived several externally caused disturbances.

Total productive time in the Technical Shops Operation for the period was 23,275 hr. Distribution of time was as follows:

	Man-Hours	% of Total
N-Reactor Department	2 876	12.36
Irradiation Processing Department	4 280	18.39
Chemical Processing Department	480	2.06
Hanford Laboratories	15 639	67.19
Hanford Utilities and Purchasing Department	0	-

Total productive time for Laboratory Maintenance Operation was 18,500 hr of 20 000 hr potentially available. Of the total productive time, 91% was expended in support of Hanford Laboratories components, with the remaining 9% directed toward providing service for other HAPO organizations. Manpower utilization for October was as follows:

A. Shop Work		2200 hr
B. Maintenance		7000 hr
1. Preventive Maintenance	2200 hr	
2. Unscheduled or Emergency Maintenance	1400 hr	
3. Normal Scheduled Maintenance	3400 hr	
4. Overtime (Included in above figures)	760 hr	
C. R&D Assistance		9300 hr

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At the request of RLOO-AEC, estimates of $420,000 for construction and $80,000 for research and development were developed for FY 1964 for initiation of the Containment Test Program. Total construction costs of the facility to be located in the T Plant are estimated at $870,000.

The N-Reactor model was installed in the Visitors Center; approval to operate the Visitors Center has been extended without a terminal date.

The following data show the magnitude of Hanford visitations:

	Number of Visitors	
	In October	Since June 13, 1962
Visitors Center	1488	58,113
Plant Tours	197	n. a.

HAPO professional recruiting activity this month is summarized below:

	Visits	Extended	Acceptances Received	Rejections Received	Open Offers at Month End
Ph.D.	3	1	0	4	1
BS/MS (Direct Placement)	0	2	1	0	2
BS/MS (Program)		4	0	0	4

One technical graduate was placed on permanent assignment. Two graduates joined the program, bringing the current strength to 76.

Authorized funds for seven active projects total $6,424,500. The total estimated cost of these projects is $10,269,000. Expenditures through September were $1,261,000.

R S Paul

for Manager, Hanford Laboratories

HM Parker:JKG:dph

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REACTOR AND FUELS LABORATORY MONTHLY REPORT

OCTOBER 1963

TECHNICAL ACTIVITIES

A. FISSIONABLE MATERIALS - 02 PROGRAM

1. Metallic Fuel Development

High Temperature Irradiation of Tubular Fuel Elements. Post-irradiation examination of a KSE-5 element irradiated in KER Loop 2 to 1350 MWD/T at a volume mean fuel temperature of 535 C has been completed. The fuel experienced 2.28% increase in volume during irradiation, and most of this increase appears attributable to fission gas porosity. There is microscopic evidence, however, that a small part of the volume increase may be attributed to grain-boundary tearing. In the lower temperature zones of the fuel near the bonded end caps and near the cladding, small irregular holes have formed in grain boundaries. A small part of the volume increase may also be attributed to intergranular cracks observed at the center of the fuel section. These cracks are associated with the thermal contraction of the fuel which occurs at reactor shutdown.

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

The eight target elements which were discharged from Loop 1 after 60 days at full reactor power are being evaluated in Radiometallurgy. No over-all dimensional changes in the target elements were detected with the exception of a 0.1% increase in the length of three of the elements. The measured bulk densities of the lithium-aluminum core, the OD of the core and the volume of the central hole of the core all agreed with pre-test values and show that no measurable swelling had occurred.

Gas samples taken from the central holes of four of the target elements showed that up to 0.5% of the total tritium was released from the target alloy. However, gas samples taken from the annulus between the aluminum can and the Zircaloy-4 can of each element contained no measurable tritium. Various samples of the Zircaloy-4 and aluminum cans have been taken and will be analyzed for tritium. Samples of the Li-Al cores have been taken and will be analyzed

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for tritium, helium, and Li^6. These data will yield total gas production and evidence of mass transport.

Drop Tested NAE. Visual examination of the outer and inner component of an "N" fuel assembly that was drop-tested in the KE viewing facility has been made at Radiometallurgy. The outer tube broke transversely near its center in what appears to be a brittle manner. There is no evidence of bond failure or fuel fragmentation associated with the fracture. The outer component tested had attained 2600 MWD/T exposure and experienced 1.4 vol% increase during irradiation. The inner tube of the assembly did not appear damaged as a result of drop testing.

Fuel Fouling Detector. Eight thermocoupled fuel rods have been fabricated for use as prototype fuel fouling detectors for N-Reactor. These elements are thorium - 1.5 wt% enriched uranium - 1 wt% zirconium alloy clad with Zircaloy-2. Four were prepared using double-sheathed Zircaloy-2 clad thermocouples brazed into the fuel rods with zirconium - 5 wt% beryllium alloy, and four are assembled with a stainless steel sheathed thermocouple sealed into each rod using a compression ring. One element having a Zircaloy-2 thermocouple was sent to Idaho Falls for short term irradiation in the ETR P-7 Loop, to evaluate its performance in high temperature pressurized water.

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

To investigate the ability of a submicron dispersion of uranium carbide to reduce uranium swelling, three NaK capsules are being prepared for irradiation in the MTR or ETR. Two of these capsules will compare the swelling performance of two Zr-2 clad uranium fuel rods identical in uranium composition, but with one having a uranium carbide size of 2-5 microns and the other a carbide size of less than 0.5 micron. The fine carbide uranium samples were produced from chill cast uranium shot which was coextruded with Zircaloy-2 to give a clad rod of near theoretical density. These two capsules will be irradiated to approximately 0.3 and 0.6 at% burnup. The third capsule will contain two fuel rods of the fine carbide and will be irradiated to approximately 1.0 at% burnup.

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Fabrication of the fuel samples and capsule parts is completed and assembly of the capsule started. Irradiation of these capsules is scheduled to begin in December.

Target Element Development. Component fabrication for a physics test and for a poison column loading is under way. Massive aluminum castings were made and extruded on the 333 press into billet tubes. Extrusions were made cold, using a combination of the new "Cindol" aluminum lubricant and Fisks "604." The surface quality of the extrusions was poor. Extrusion constants, lubricants, die core angle, extrusion speed, die temperature and billet temperature will be determined on the 306 extrusion press using pilot billets. For this purpose ten composite billets have been made together with two die sizes. Billet size was adjusted to give the same extrusions. Final composite billet assembly is almost complete. Billet shells have been machined from the billet tubes, and castings have been made of the aluminum - 1% lithium core alloy. A major problem was encountered in melting the alloys. The lithium attacked the graphite crucible at temperatures over 500 C through intragranular corrosion, resulting in complete failure within minutes. This was overcome by washing the crucible with a heavy layer of ZrO_2, then carefully cleaning after each melt. Piping in the casting was minimized by chilling the mold wall by water cooling coils, vibrating the mold during metal solidification and constantly hot tapping. An aluminum bright etch was found for final cleaning for billet preparation. Supports to center the elements within the driver elements or process tube are being fabricated from 8001 aluminum sheet, with steel shoes to prevent aluminum rubbing onto the Zircaloy surfaces. Present goals call for fabrication of one tube of poison elements and two tubes of target elements for N-Reactor by November 15, followed by fabrication of two additional tubes of target elements by December 1.

Hot-Headed Closure Studies. Seventy hot-headed sections of N-inner fuel elements were prepared. These are being processed for end cap welding. Subsequent to welding they will be subjected to a series of evaluation tests.

Work on the development of projection welding methods for outer fuel elements similar to those applied to inner elements has continued. Tooling for welding has been designed and constructed and sections of N-outer elements have been hot-headed and prepared for end cap welding. Efforts to projection weld end caps to the elements are proceeding.

Cladding Deformation Studies. Thirty-six NaK capsules containing a total of 94 Zr-2 clad uranium rods have been irradiated to provide data on the strain capabilities of Zircaloy-2 cladding as a function of cladding thickness uniformity, temperature, and exposure. A total of 54 samples have been examined visually and diameter measurements completed. The data from these fuel rods are in agreement with previous data relating mean cladding temperature to total uniform cladding strain for samples with small (± 0.002 inch) cladding thickness variations. Below an average cladding temperature of 325-350 C, the total cladding strain limit, before plastic instability or necking occurs, is approximately 1.5%. Above cladding temperatures of 350 C, the strain limit increases. From this analysis of the data it is concluded that irradiation damage plays a significant role in cladding instability below 300-350 C.

From the data on the striated and unstriated samples with a nominal cladding thickness of 0.025 inch, an analysis was made of the effect of striation depth on cladding instability. Total uniform cladding strain decreases from 1.6% for an unstriated sample to approximately 0.5% for a sample with a single intentional striation of 0.005 inch depth.

Three of the capsules of this test contained U - 2 wt% Zr fuel cores coextrusion clad with Zircaloy-2. Four of these samples showed little or no localized strain even though the total cladding strains were in the range of 2.6 to 2.9%. This cladding performance is considerably better than for any of the unalloyed fuels.

Uranium Sulfur Alloys. An alternate scrap recovery process at NLO will use uranyl ammonium sulfite (UAS) to replace the uranyl ammonium phosphate (UAP) feed to the Winlo Process. No work has been done on the low end of the U-S diagram (0-1 wt% S) so the effects of sulfur additions to uranium are being studied. Two hundred grams of powdered uranium mono sulfide have been obtained from Argonne National Laboratory for investigation of the low sulfur end of the U-S diagram. The powder will be used to form uranium-sulfur alloys of 250-10,000 ppm sulfur.

Heat Treatment of Uranium-Carbon Alloys. The effects of heat treating uranium containing carbon in the high gamma uranium phase are being studied. Work by Blumenthal of Argonne National Laboratory has indicated carbon losses when uranium is held at temperatures in the gamma range. Ingot uranium (300-400 ppm carbon) specimens have been prepared and are now being analyzed for carbon. Specimens are also being heated to temperatures in the gamma range (800-1100 C) and will be reanalyzed for carbon and examined microscopically to observe the effects of heating.

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Stainless Steel Clad Niobium Tube Fabrication. A $\frac{1}{2}$" OD stainless steel clad Nb - 1% Zr tube was produced from a 2.085-inch OD extrusion billet by a combination of hot extrusion and cold swaging. The coextrusion billet was designed to produce a 0.020 inch thick stainless steel clad over a 0.040 inch wall Nb - 1% Zr tube containing sacrificial material of graphite cored 1020 steel. The billet components were cleaned and the niobium component was vacuum sealed between the stainless steel and 1020 steel by EBW to protect the stainless niobium interface from oxidation during preheat and extrusion.

The extrusion billet was preheated to 1250 C and extruded in a 0.030 inch thick graphite sleeve to 3/4 inch diameter. The extrusion was then cold swaged to $\frac{1}{2}$ inch diameter, the graphite drilled out and the 1020 steel chemically removed leaving the stainless steel clad Nb tube.

Examination of the tube revealed that little or no diffusion bond existed between the stainless and the Nb - 1% Zr. Oxide was present at the interface in the lead half of the extrusion and can be attributed to the difficulty experienced in making the weld on that end of the billet.

Creep-Penetration Apparatus. Detailed drawings for a device for determining the hot hardness of materials as a function of time have been received from Mallinckrodt. These plans have been slightly modified for use with available laboratory equipment. The device will be built and used for investigating various uranium alloys.

2. Corrosion and Water Quality Studies

Studies on the Release of Tritium from Aluminum-Lithium Alloys. The migration rate and containment of tritium in aluminum-lithium target materials is being investigated by means of the following studies:

Data on in-reactor tritium evolution rates in the first capsule indicated a much lower temperature coefficient (40 kcal/mole) than has been reported by duPont (140 kcal/mole) for ex-reactor tritium outgassing rates.

A second in-reactor capsule will be charged in KE Reactor during the current tube outage. The capsule will operate at 300 C to terminal GVR (gas volume ratio, i.e., cc T_2+ He per cc of target alloy); the capsule will be monitored periodically to see if measurable tritium is released at this or higher temperatures. Evolution rates also

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will be determined at a single temperature with the reactor up and down to see if neutron radiation levels cause detectable changes in tritium evolution rate.

Zircaloy Corrosion. The effects of beryllium and uranium additions on the corrosion of Zircaloy-2 were described in the August monthly report. It was shown that following 14 days of exposure in 360 C (680 F) pH-10 (LiOH) deoxygenated water, Zircaloy-2 corrosion rates increased proportionally on increasing beryllium concentration from 0 to 5%. There was no effect of 0-2200 ppm uranium concentrations on the Zircaloy-2 corrosion rates.

Except for the 1.5 to 2.5% Be-Zr-2 specimens, the previously noted effects are continuing following 28 days of exposure in 360 C pH-10 (LiOH) deoxygenated water. Zircaloy-2 samples containing no beryllium have weight gains of 31.3 mg/dm^2 as compared to 175 mg/dm^2 for samples with 3% Be, and 181 mg/dm^2 for samples with 5% Be.

The 1.5 to 2.5% Be-Zr-2 samples are undergoing a sloughing type corrosion so that weight losses rather than weight gains are observed. The sloughing corrosion appears to be entirely due to the beryllium content and is independent of the uranium concentrations which range from 0-2200 ppm in these particular samples.

Erosion Corrosion of Aluminum Alloys. Preliminary experiments were completed in an ex-reactor loop to determine whether aluminum alloys (X-8001 and X-8003) would be susceptible to erosion corrosion or other types of accelerated corrosion in process water at high velocity (42 fps) at 95 C. Samples were removed after 12 days and 26 days. The X-8003 alloy showed a smaller weight-loss. Some markings were found on the X-8003 which indicated possible erosion corrosion attack but was later found to be due to preparatory treatments.

Comparison of High and Low Alum Dosage. A test is in progress at F-Reactor to evaluate the effects of high alum dosage on corrosion and effluent activity. One half of the reactor is cooled by water treated with 18 ppm alum and the other is on water treated with the normal amount of alum (5-15 ppm). The pH is adjusted to 6.0 and sodium dichromate is 1.8 ppm.

Samples of aluminum coated with protective materials and samples of carbon steel were removed after 22 days of exposure. Several of the aluminum samples appear to have the protective coating still in place after exposure. The samples are being processed for film and corrosion measurements.

concentrations were low (0-3 ppm). Check samples obtained from the normally stagnant sampling system used during previous tests indicated slight amounts of turbidity and higher solids concentrations.

As another measure of the crudding tendency of the ammoniated coolant, the loop contains two instrumented crud detectors (one upstream and one downstream). The crud detectors are stainless steel clad UO_2 core elements with Zircaloy sleeves, each fitted with three thermocouples in the jacket. A fourth thermocouple is in the water adjacent to the element. It is presently planned to leave these elements in during the present outage and subsequent operation until they attain 1600 EFPH (about January).

The data from the thermocouples indicate there was no crud build-up, which agrees very well with the coolant quality during this period. All of the thermocouples appear to be functioning properly and, in general, both crud detectors have performed satisfactorily to date.

Additional ammoniated coolant tests are planned in K-4 Loop to determine the decomposition of ammonia at higher power levels. The K-4 Loop is now being modified for this testing.

The UO_2 in the crud detectors is packed to 65% theoretical density. Some concern has been expressed that because of the low density, the core might shift during irradiation and result in thermocouple readings which vary with position, and generally be difficult to interpret with respect to crudding. A new element is now being designed in which the core will be a cermet of UO_2 and/or PuO_2 in molybdenum. Similar elements have been irradiated with core temperatures almost to the melting point of molybdenum and post-irradiation examination revealed no reaction had occurred between the molybdenum and the oxide.

Evaluation of Decontaminants. Laboratory studies recently completed have shown that solutions of formic acid - citric acid and citric acid-hydroxyacetic acid efficiently decontaminate carbon steel with low corrosion losses. Mixtures of hydroxyacetic acid with either sulfamic acid or ammonium citrate were also effective but were more corrosive than desirable. All solutions were inhibited (with West Chemical inhibitor).

Further studies were made to improve the formulation of the peracetic solutions used to dissolve uranium and uranium oxides after a fuel element rupture. The variation of peracetic acid concen- [redacted]

[redacted]ion from 2.4 g/l to 20 g/l had no effect on the amount of

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Results from previous tests in 7.0 pH water with 1.0 ppm dichromate showed that corrosion of aluminum alloys 1100 and 6061 is approximately $1\frac{1}{2}$ to 2 times as great in water with 18 ppm alum dosage as in water with low alum dosage. Corrosion of aluminum alloy X-8001 and of carbon steel was about the same with both dosages.

Corrosion on Aluminum Heat Transfer Surfaces. The corrosion and film deposition on an X-8001 aluminum clad electrical heater was measured after 41 days in an ex-reactor loop (TF-20) operated with single-pass process water at 120 C. Calculated heat flux at the surface was 72,700 Btu/hr/ft^2 and the surface temperature was estimated at 150 C. A film of approximately 6 mils was formed and the penetration of the aluminum was 4.4 mils. The penetration of aluminum coupons (without heat transfer) for the same operating period in the same system was 1.2 mils. This study is continuing with similar X-8001 aluminum heat transfer surfaces being subjected to similar conditions, for longer exposures.

Resistance Temperature Detectors. A silver plated monel RTD, a copper plated monel RTD, and an uncoated monel RTD were removed from TF-20 after 4 weeks of exposure to pH 6.6, 120 C process water. The uncoated monel RTD withstood the conditions very satisfactorily showing no sign of unusual corrosive attack. The copper plated RTD had copper plating flake off in large areas, but the monel underneath was not visibly damaged. The silver plated RTD showed silver plating blisters broken through in some places. The money appeared to be attacked in these locations.

Evaluation of NH_4OH for N-Reactor Primary Coolant pH Adjustment. The program to evaluate NH_4OH as a substitute for LiOH in N-Reactor primary system is continuing. The K-1 Loop has been operating with NH_4OH since September 9, and at the time of the extended shutdown for tube replacement (October 8) had accumulated 450 effective full power hours (EFPH).

Coolant quality during the entire operating period (Sept. 9-Oct. 8) was satisfactory. In general, the ammonia concentration in the coolant was controlled at 12-15 gpm. The loop operated at a feed-and-bleed purification rate of 1 gpm, and it was necessary to add approximately 0.2 pound of ammonia per day to maintain the desired coolant concentration. The gas concentration present in the coolant was about 8-10 cm^3/liter (stp). The H_2/N_2 ratio observed was about 1-1.5.

None of the coolant samples obtained from the continuous flow sampling stream were visibly turbid, and the total solids

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oxides dissolved but did cause more precipitate to be formed. The use of 50 to 100 g/l peracetic acid without the peroxide gave results comparable to the solution with the peroxide. A test using this modification is planned for the IRP Loop.

N-Reactor Secondary System Cleaning. The N-Reactor secondary system was cleaned (by Dow Industrial Services) using a three-step procedure: (1) a hot water recirculation and rinse; (2) chemical cleaning with Vertan 690; and (3) passivation with Turco 4517. Scaled samples and weighed coupons were inserted in the piping to monitor the cleaning operation. Inspection of the samples and sections of temporary piping indicated the cleaning operation was successful and that an adequate passivation had been obtained. Corrosion penetrations of carbon steel coupons after the hot water rinse (10-12 hours at 150-180 F), after the Vertain 690 descaling (17 hours at 150-205 F), and after the Turco 4517 passivation were 0.04, 0.10, and 0.10 mil, respectively. Penetrations on silicon brass were less than 0.1 mil in all solutions.

Decontamination Tests of Tubing from N-Reactor Steam Generators. Nondestructive examinations have shown that many tubes in the N-Reactor steam generator have areas of intergranular attack, usually hemispherical in shape ranging in depth from 0 to over 80% penetration of the tubing wall. Tests were completed to determine what effects from proposed decontamination processes might be expected at these areas. One section of tubing was exposed to eight cycles of alkaline permanganate for two hours, followed by a two-hour exposure to a 10% bisulfate decontaminant at 60 C for two hours. A second sample was similarly exposed but employing a sulfamic acid decontaminant instead of the bisulfate. Chemical attack at the grain boundaries was found to have occurred deeply within the intergranular attacked areas and was widening the void space between the grains. Mild pitting was also occurring.

Two other tubing samples were exposed to 26 cycles and 20 cycles of the AP-bisulfate process. In these samples the grains had entirely fallen out of the intergranularly attacked regions. These samples are being given additional metallurgical examination.

In the TF-4 Loop, a test has been started to evaluate the effects of repeated decontaminations on N-Reactor heat exchanger tubing. Three cycles (of a projected eight) of alkaline permanganate followed by sulfamic acid (Turco 4306-D) have been completed.

Corrosion Tests of N-Reactor Steam Generator Components. Samples of defective steam generator tubing which have been pressurized to

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cause pinhole defects at the intergranularly attacked areas are being exposed in a high temperature loop (TF-7) to determine the leak rates of the defects, whether the defects increase in size with time, and whether any differences occur with lithiated water versus ammoniated water.

Tests are continuing to determine whether the fumes given off by charred purge balloons and charred and uncharred vapor phase inhibitor (VPI) will cause stress-corrosion cracking at room temperature. To date, no visible cracking has occurred on stressed specimens exposed $3\frac{1}{2}$ months in uncharred VPI or $1\frac{1}{2}$ months in charred purge balloon or VPI vapors. Localized pitting is occurring on the samples exposed to the balloon vapors. Similar tests at 80-90 C have been started. After seven days, no cracking has occurred.

The vapors given off in charring the purge balloon were found to be highly acidic and contained large amounts of chloride, indicating the gas given off is HCl. When VPI is heated, dense clouds of highly acidic brown nitrogen oxide gases are given off. Within a few seconds, the vapors become highly basic probably due to the presence of volatile amines.

Ultrasonic Decontamination of Sample Holders. A method of reclaiming sample holders used extensively in corrosion tests in radioactive, high temperature water environments was developed. These sample holders were previously discarded after use because it was impossible to decontaminate the numerous crevices, threaded holes, and joints. The method developed employs an ultrasonic cleaning unit. The cost of the new cleaning unit ($250) was saved during the first application when three sample holders were cleaned to background radiation levels. It is estimated that the use of this procedure for this one application will result in savings of several thousands of dollars per year. In addition, the unit is being employed in cleaning non-radioactive corrosion samples. In this use the unit has reduced the cleaning time by about a factor of five.

3. Gas-Atmosphere Studies

The second of two sets of graphite samples, intended to monitor the effect of CO_2 in B-Reactor atmosphere confirmed the results of the first measurement, showing burnout rates less than the allowed 2% per 1000 operating days (%/KOD). The second set was removed from Channel 1081 on September 23, 1963; the highest measured rate was 0.47%/KOD at a distance of about 115 inches into the front of the stack.

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Thus, B-Reactor monitors continue to show the absence of air in-leakage and the inconsequential effect of higher CO_2 concentrations. Consideration is being given to the operation of other reactors under this production test in the near future.

Monitors were also replaced in Channel 2577 in DR-Reactor. Test samples exposed in Channel 2577 of DR Reactor from November 18, 1962 to October 5, 1963, showed a maximum rate of 3.5%/KOD at about 80 inches into the stack.

Initial Oxidation Studies. The rate of reaction of TSX graphite with carbon dioxide at one atmosphere pressure has been determined in the temperature range from 750 to 900 C. The sample was in the form of a plate of dimensions 1.0 by 0.4 by 0.25 inch. The sample was preoxidized to a total burnoff of 5% and during the course of the experiments an additional 5% weight loss was sustained by the sample. At each experimental temperature a linear weight loss versus time was observed. The rate data obey the over-all equation:

$$\text{Rate (g/g/hr)} = 6.38 \times 10^9 \exp(-70.2 \times 10^3/RT) \qquad (1)$$

The purpose of this study with TSX graphite is for use in the N-Reactor atmosphere computer code. The values currently being used in that program were based on CSF graphite.

If, in the accepted mechanism

$$\text{Rate} = \frac{k_1 P_{CO_2}}{1 + k_2 P_{CO} + k_3 P_{CO_2}} \qquad (2)$$

and it is assumed that k_2 and k_3 are the same as those given for CSF graphite (HW-74663), then a new value for k_1 can be determined:

$$k_1 = 3.75 \times 10^6 \exp(-65.5/RT)\ (\text{hr}^{-1}\ \text{mm}^{-1}) \qquad (3).$$

4. Process Tube Development

Project CAH-922 Irradiated Burst Test. The construction of this facility has progressed favorably during the past month. The cost plus fixed fee contractor (J. A. Jones) is progressing favorably with the installation of the heating and cooling systems, the electrical system and the equipment enclosures. The furnace-containment vessels have not been received as yet (original delivery schedule was 9/15/63); however, they were inspected at the vendor's shop during the month. The vessels will be completed by the vendor; however, the furnaces must be rebuilt to conform to the design and this work will be done in the CPFF contractor's shop (J. A. Jones). The instrument panels were observed and found to be progressing favorably though the original delivery schedule of October 24, 1963, was not met. The anticipated delivery is November 15, 1963. The majority of the instruments have been received by the vendor thus permitting the wiring and piping hookups to be made on the panels. The project scheduled completion of 91% (10/31/63) still leads the estimated completion of 71% because of the delayed delivery of the containment vessels and instrument panels. The construction schedule has not as yet been affected by the delayed delivery dates of these items.

Stress Rupture Facility. This facility has continued to operate at a capacity during the past month. Testing of the N-Reactor steam generator tubing has continued. Originally, 25 sections of this tubing with flaw depths greater than 80% of the tube wall were installed on a common manifold for evaluation at test conditions of 3000 psig and 300 C (572 F) as compared with a maximum design pressure and temperature of 1800 psig and 315 C (600 F), respectively. To date, 13 sections of this material have failed and the total time at test conditions is now 845 hours. Five longer sections of the N-Reactor steam generator tubing were tested at 6000 psig and 300 C (572 F); four of these sections failed in less than one hour. A series of tests at 5000 psig and 300 C are being prepared.

347 SS Tubing From the ETR P-7 Loop. Room temperature tensile tests were run on sections cut from the P-7, 347 SS Loop. These tensile sections were cut from both the in-core and out-of-core portions of the tube. The average in-core section had an exposure of 4×10^{20} nvt (E > 1 Mev). This exposure produced an increase in ultimate strength and yield strength of 38 and 140%, respectively. The uniform elongation over the gage length was reduced 54% while the reduction in area indicated no change.

5. Thermal Hydraulic Studies

N-Reactor Studies. Experiments were continued to determine liquid-phase and two-phase pressure drops across an N-Reactor process tube, piping, and fittings during heat generating conditions. This information will be of particular value in analyzing hydraulic stability conditions for the reactor.

The experiments were conducted in the Thermal Hydraulics high pressure heat transfer apparatus using a full-scale, electrically-heated model of the downstream half of an N-Reactor fuel column with prototypic N-Reactor outlet connector and fittings. The experimental procedure consisted of varying the coolant flow, for specific active section powers and outlet riser pressures, to obtain a range of coolant outlet conditions in the subcooled and steam quality ranges. As the active section represents only the downstream half of a fuel column, the inlet temperature was adjusted as flow was changed to maintain outlet coolant enthalpies representative of those for a reactor tube operating at twice the total power of the laboratory test section. At each flow step measurements were made of pressure drops across the major systems components. Active section power, coolant temperatures and flow rates were measured also.

A total of 54 sets of measurements were made during the month. The ranges of experimental conditions covered are summarized below:

Outlet Riser Pressure psig	Heated Section Power (1) kw	Coolant Outlet Conditions
300	500, 750, 1000	100 F subcooled to 35% quality
1400	2500	60 F subcooled to 12% quality
1200	2000	15% quality
1200	2500	11% quality
1200	2600	11% quality

(1) Corresponding reactor tube power is two times this value.

The experiments at 300 psig were conducted to obtain information for pressures available from the reactor's emergency cooling system and also to obtain information applicable to reactor startup pressures. Substantial pressure drop increases occurred as net boiling was produced by reducing the flow at this pressure. For example, the pressure drop at 35% outlet quality was 3.3 to 3.5 times that at zero quality. Also, for the 35% outlet quality cases and one of the 30% outlet quality cases, the flow distribution between the three flow channels in the active section appeared to oscillate. Flow rate through the center channel showed a variation of ± 15% with a period of about two seconds. Minor flow oscillations were observed in the other two channels.

[redacted]

The 1400 psig experiments were conducted to obtain information at expected normal operating pressures. Experiments had been conducted previously at 1200 psig outlet riser pressure to determine flow versus pressure drop relationships in the neighborhood of the low pressure trip setting, the minimum possible pressure for sustained operation. The 1400 psig experiments at tube power of 2500 kw showed that the riser-to-riser pressure drop would have a net increase of 2% while the flow was decreased to change the outlet conditions from zero to 12% steam by weight. As expected, this increase was somewhat smaller than the 4-5% obtained at the same power and outlet quality with a 1200 psig outlet riser pressure.

One purpose of the three experiments conducted at 1200 psig this month was to verify burnout experiment results obtained previously with short heated sections. Conditions were designed to produce heat fluxes within 10% of the predicted burnout heat fluxes at certain locations in the heated section. No indications of burnout were observed in these experiments, indicating that conclusions drawn from the short section burnout experiments are not particularly unconservative when applied to the full-scale model.

The experimental results obtained in the past two months indicate a fairly flat riser-to-riser pressure drop versus flow curve in the normal operating pressure range with outlet qualities up to about 30%. For most types of incidents it appears that instability would not occur at these pressures. At 300 psig outlet pressure, however, the riser-to-riser pressure drop increases quite sharply with increasing outlet quality and detailed analysis of the experimental data will be required to determine whether instability could occur for certain types of incidents.

Present Reactors Studies. Further calculations were performed to analyze the heat transfer in special elements containing thorium dioxide. In order to reduce the costs of the irradiation, it may be desirable to use ThO_2 in the "as-received" condition for vibrational compaction into the fuel cans. Since the particle size distribution in the ThO_2 "as-received" is not ideal for high density vibrational compaction, rather low theoretical densities would be expected in the finished elements. It is estimated that the minimum theoretical density will be about 60%. Since the resulting voids will have a detrimental effect on the thermal conductivity of the material at temperatures below 2500 F, lower tube power limits must be established to prevent the fusing of the powdered ThO_2. Using physics data provided by P. A. Carlson, calculations were undertaken to calculate the maximum center temperature of the elements for the range of tube powers of interest and for the various element configurations proposed. Thermal conductivity-temperature values for 90% density ThO_2 were used with corrections to 60% density made by means of Eucken's rule. For a solid element 1.5 inch OD with 0.020-inch aluminum cladding, the following temperatures were calculated:

Specific Power kw/ft	Clad Temperature °F	Central Core Temperature °F
3	123	598
6	126	1400
9	129	2297
15	136	3361
21	142	4068
24	145	4344

Hydraulic Tests. Tests were continued on investigating hydraulic characteristics for rear face hardware modifications to B-D-F type and H Reactors. The purpose of the tests is to compare the pressure losses of the present rear face fitting assemblies against the pressure losses of various combinations of nozzle, adapter, and connector modifications. Preferred combinations (economically determined) appear to consist of a reamed front nozzle from K-Reactor (surplus from retubing of K-Reactors with zirconium tubes), a 0.543-inch diameter nozzle adapter, a 5/8-inch diameter connector, and either 0.543 or 0.469-inch diameter header adapter. If rear fitting modifications are made on the reactors, present Panellit trip limits and settings must be reviewed.

The flow characteristic equations were determined and reported for ten 0.419-inch venturis which were to be used in a production test on K-Reactor. The production test (IP-572-A) concerned the effect of eccentricity on the irradiation behavior of self-supported fuel elements.

The flow characteristics were determined for five 0.344-inch venturis (CG-558 type) which had been rejected because they were outside the dimensional tolerance which was specified during reaming of B-Reactor venturis from 0.310 to 0.344 inch. It was found that although the reject venturis showed flow characteristics which were slightly different than the standard (within tolerance) 0.344-inch venturis, all five of the reject venturis would indicate a flow rate within 1% of the actual flow rate when using the calibration equation of the standard venturis. This should be accurate enough for reactor usage of the "reject venturis." Since the cost of individual calibration on the rejects is only about one-sixth of the cost of replacement, signficiant savings could result if the B-Reactor rejects are calibrated and found acceptable for use in subsequent venturi reaming programs on other reactors.

6. Shielding Studies

N-Reactor Shielding Evaluation Program. Effort was centered around design of apparatus to facilitate measurement of gamma ray and neutron flux distributions in the N-Reactor shield plug. The shield plug was removed from the reactor for inspection and measurements were taken for the design of experimental equipment. Foil holders which fit the shield plugs have been fabricated. A thimble which will fit the shield flux opening in the reflector below the shield plug has been designed. This thimble will form a gas seal and allow access to the reflector without disturbing other reactor operation. A graphite rod to fit inside the thimble and hold the detectors also is being designed. Calculations to estimate neutron and gamma dose rates at various locations within the reflector and shield plug as a function of reactor power level are in progress to estimate foil exposure time and detector count rates. A special test report is being planned.

The instrumentation development program for the N-Reactor shield evaluation is proceeding. Considerable difficulty was encountered with instability in the neutron spectrometer equipment, but the $helium^3$ detector has now been operated with satisfactory count rates in the coincidence mode. However, additional work will be required to improve the detector resolution. The $lithium^6$ and plastic scintillation detectors are available and preparations are under way for experimentation with these devices.

A multiple thermocouple temperature recording system has been designed and assembly of this apparatus has been ordered.

An experiment in the PCTR to pretest the N-Reactor shield evaluation equipment is being planned which will involve exposing all of the detectors (both foils and electronic devices) to neutron fluxes of similar intensity to N-Reactor fluxes at the shield plug locations during zero power and power ascension tests. The detectors which will be exposed are (1) gold, sulphur, iron, titanium, nickel, and cobalt foils; (2) the $helium^3$, $lithium^6$, and plastic scintillation neutron spectral detectors; and (3) the gamma ion chambers.

7. Graphite Studies

Graphite Contraction. Measurements of filler-block and tube-block separations in the 2C Test at KW were obtained by IPD personnel in August. The dimension changes were compared with the average contractions inferred from vertical-height traverses of filler blocks in other parts of the central zone. The dimensional changes are in good agreement, and the ratio of the tube-block contraction to filler-block contraction is consistent with the variation in neutron flux and the average temperature differences. The hole in the tube block does not appear to have significantly altered the contraction rate of the tube block in relation to the contraction rate of a solid bar.

N-Reactor Graphite Irradiations. The long-term irradiations of N-Reactor graphite continue satisfactorily. The first third-generation capsule, H-4-3, was removed from the GETR on September 23, after successfully completing five cycles (108.0 effective full power days) of irradiation. Maximum neutron exposure to samples exposed in three consecutive capsules is estimated at 7×10^{21} nvt, $E > 0.18$ Mev. The capsule was disassembled on September 26 and 27 without incident and all capsule components were found in good condition.

The last of the third-generation capsules, H-6-3, was installed in the GETR on September 24 to replace H-4-3. Twenty of the samples in H-6-3 have been previously irradiated in two consecutive capsules and the remaining four have been previously irradiated in one capsule. The capsule is operating completely satisfactorily after two weeks of reactor operation.

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.

[redacted]

C. REACTOR DEVELOPMENT - 04 PROGRAM

1. Plutonium Recycle Program

Fuels Development

PRTR Fuel Element Inspection. Basin and FEEF examinations of PRTR fuel elements were conducted to assess the condition of selected PRTR fuel elements.

The black zirconium oxide film formed in-reactor on three non-autoclaved fuel elements was generally good, after exposure to approximately 67 MWD. The unusual rod shortening and wire wrap loosening associated with Al-Pu fuel rods was observed. Several areas where the original black oxide film had been scuffed had developed gray films. For the first time, indications were seen of relative movement between rod wire wraps and the fuel rod surfaces (as evidenced by gray surface scuff marks under the wires). This rubbing by the loose wires is perhaps more noticeable with this element because the in-reactor-formed film may be thinner than a conventional, higher temperature autoclave film.

The general condition of other elements examined (swaged and vibrationally compacted UO_2-PuO_2) was excellent.

Post-Irradiation Examination of Failed Vibrationally Compacted UO_2-PuO_2 PRTR Fuel Elements. A sample of Zircaloy cladding from the point of failure of a rod from a vibrationally compacted PRTR UO_2-PuO_2 fuel element (FE-5203 with 200 MWD/T) was identified by x-ray diffraction as zirconium hydride. This gross hydriding is the first positive characterization of this previously observed, structurally featureless material which does not display the usual needle-like appearance of zirconium hydride, and does not appear to be associated with heavy corrosion (i.e., oxidation) attack. Twelve and nine-tenths ml of gas was collected from a nonfailed fuel rod from this element. Mass spectrographic analysis revealed 36% hydrogen.

Cladding Failure Studies. Samples of Zr-2 chips machined from a variety of source materials were analyzed for F and Cl contamination. Although a few high results were obtained, no definite pattern of contamination was established. Samples analyzed included those from the inner surface of cladding used for swaged and vibrationally compacted fuel rods, from end caps used for both types of fuel rods, from Zircaloy bar used to fabricate the end caps, from off-site fabricated cladding components, from EBWR cladding

[redacted]

components, and from an NBS Zircaloy-2 spectrographic standard. Analyses were also conducted on subsurface layers. Various surface cleaning techniques were evaluated. Only four significantly high F analyses were recorded in more than 60 samples obtained. From this sampling it appears that while F contamination exists on some fuel rod cladding and end caps, the statistics do not support fluoride contamination as the principal cause of recent fuel failures. To eliminate any potential failures because of surface contamination, however, all tubing and end cap material is now routinely ultrasonically cleaned in a detergent solution, then rinsed with tap water and deionized water, dried, and stored in vacuum.

Fuel Element Refurbishing. Two UO_2-PuO_2 mixed oxide fuel elements (5117, 5119) with broken rod wire wraps were repaired in the PRTR basin. The fuel elements were then examined in the FEEF and charged into the reactor.

PRTR Fuel Fabrication Studies. Three impacted UO_2-1 wt% PuO_2 swage compacted fuel elements were prepared for the PRTR. Five similar elements are in process. The following measures were employed to minimize or eliminate halogen and moisture in fuel material and cladding: (1) ultrasonic cleaning of all components, (2) vacuum storage of components, (3) longer pump-down period in weld box, (4) shorter exposure of fuel to the hood atmosphere, and (5) increased inspection and cleanup. Impacted UO_2-PuO_2 is now prepared at a rate of 200 pounds per day.

The impaction of 15-pound containers of UO_2-1 wt% PuO_2 for PRTR fuel elements was successfully demonstrated on the newly installed Model 1220-B Dynapak machine in the 308 Building. More than 300 pounds of fuel > 99% TD were impacted on the first day of trial operations.

The container design for impacting fuels was modified to use a $\frac{1}{4}$-inch diameter off-gas tube instead of the 1/8-inch tube used previously. This change simplified handling of the loaded cans between the heating and impaction facilities. The punch protector - a mild steel disc used to protect the hardened face of the punch from the hot impaction can - was modified by slightly increasing the diameter and increasing the thickness. A tendency of the welded junction between the can wall and the lid to extrude between the punch and die wall was thus eliminated.

An improved method for removing material from Nupac containers of UO_2-PuO_2 was developed. Equipment procurement and installation in the plutonium fabrication pilot plant were completed. The new method replaces a manual hammer operation with an electric hammer operation.

Material preparation for the high energy impaction (Nupac) process involves roasting pulverized UO_2 in air at 125 ± 5 F overnight (~ 17 hours) to increase the O/U ratio for greater plasticity. The O/U ratio can become too high for reactor specifications if the time, temperature, and size distribution of the material are not optimum. A timer has been installed and the time limit set at 15 hours for PRTR material and 12 hours for EBWR material, the EBWR UO_2 being a finer size than the PRTR UO_2.

Tests were made to determine the efficiency of a new purification method for PuO_2, employing humidified air. The results indicate that conditions of temperature and humidity are of greater importance than extended time of treatment.

Several materials used in fuel fabrication processes were found to contain halides and were therefore eliminated. For example, cutting oil used in counterbore Zircaloy tubing contained 3.5 wt% chloride and traces of fluoride. Tap water used to rinse cladding contained 30 ppm chloride. One plastic "white" tape used to seal plastic bags to containers of fuel materials contained chloride on the "sticky" side; the tape material itself contained about 20 wt% chloride. Polyvinyl chloride plastic, Teflon, and tygon-type materials also are being removed from service wherever possible. The rapid screening of many such materials for halides was made possible by x-ray fluorescence and infrared techniques.

Compatibility Studies. A capsule for examining salt cycle UO_2-Zircaloy compatibility was completed. A 1/16-inch diameter tungsten electrode provides central heating, and will be monitored by a Pt - Pt-10% Rh thermocouple embedded in the UO_2. End closures are made by Lavite plugs which provide electrically-insulating, mechanically-strong, but not hermetrically-tight seals. Initial tests of the system, instrumentation, and safety features will be conducted after sample loading. The pressure jacket has already been room temperature tested to 1800 psi with no leaks.

Irradiation Testing of Prototypic EBWR Fuel Rods. Post-irradiation examination of two vibrationally compacted $U^{D*}O_2$-2.5 wt% PuO_2 capsules was continued. Fission gas release data are tabulated as follows:

*Superscript "D" signifies depleted uranium.

Capsule No.	UO_2-PuO_2 Fuel Type	Total Gas Collected,* (ml)	Percent Release of Xe + Kr **
GEH-14-422	Impacted	13.1	5-6
GEH-41423	Physically mixed	9.0	4-5

*At S.T.P.
**Based on exposure estimated by MTR personnel - 0.37 x 10^{20} fissions/ cm^3 (1300 MWD/T fuel).

Analyses of the collected gas revealed a high concentration of hydrogen (~ 72%) in capsule GEH-14-423. Less than 2% hydrogen was present in GEH-14-422. Ceramographic examination of the former capsule disclosed a corroded (internally) Zircaloy end cap, no U_4O_9, and inclusions presumed to be metallic uranium (or possibly iron). Impurity levels in archive samples of the fuel were: 28 ppm F, 50 ppm Cl, and H_2O (incomplete). A tentative explanation of the high hydrogen content in the irradiated capsule involves radiolytic decomposition of sorbed moisture in the fuel.

Analyses of archive specimens of the impacted UO_2-PuO_2 in GEH-14-422 revealed: 16 ppm F, 93 ppm Cl, and H_2O (incomplete). No internal corrosion of Zircaloy was noted.

Fuel structures indicate that both GEH-14-422 and GEH-14-423 operated somewhat below the design condition (Zircaloy surface heat flux of 158 w/cm^2 or 500,000 Btu/hr-ft^2) and that the latter operated at the lower rod power. Burnup and flux-monitor wire analyses are in process to establish actual operational conditions.

Capsules GEH-14-421 and -424 are tentatively scheduled for recharging into the ETR (Cycle 59) for irradiation to 1.43 x 10^{20} fissions/cm^3 (5000 MWD/T fuel).

Irradiation Behavior of Uranium-Plutonium Oxide. Microdrill samples of eleven irradiated UO_2-PuO_2 specimens were analyzed. Test specimens included high density (~ 90% TD) sintered pellets and low density (~ 65% TD) nonsintered pellets, with up to 7.45 mole% PuO_2 in each variety. These were irradiated to generate powers of 214-1160 w/cm (7-38 kw/ft), and to 0.14-4.15 x 10^{20} fissions/cm^3 (460-19,600 MWD/T fuel). The analytical results are being evaluated.

Preliminary data from high density UO_2-2.57 mole% PuO_2 GEH-14-85, irradiated at 916 w/cm or 30 kw/ft, indicate a relatively uniform

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concentration of Pu, Sr^{90}, and Zr^{95}-Nb^{95} throughout the cross section of the fuel. Three-fold variation in Ce^{144}-Pr^{144} concentration and 100-fold variation of Cs^{137} and Ru^{106} were determined, however,

Short Duration Irradiation Tests of UO_2-PuO_2. The transport vehicle for conducting short term irradiation tests of fuel in VH-4 facility during continuous MTR operation was redesigned to prevent accidental reccoupling of element and vehicle after release. Two, new model transporters were delivered to the MTR for test.

Plutonium Wire Enriched Fuel Element. A fuel element containing sintered UO_2 pellets enriched with coaxial, Pu-15 wt% Zr alloy wires was successfully irradiated to an exposure of 2 x 10^{19} fissions/cc (680 MWD/T_{fuel}) in the MTR. The test element (GEH-4-88) generated a maximum surface heat flux of 122 w/cm^2 (385,000 Btu/hr-ft^2). Concentration of plutonium at the axis of the fuel rods extends the reactivity lieftime of the fuel element because of plutonium self-shielding. This enrichment method may also permit separation of recycled plutonium and newly formed plutonium. Post-irradiation examination is expected to yield information on: (1) the interaction of the plutonium alloy and UO_2 during irradiation, (2) the extent of plutonium relocation [redacted] tested in-reactor). The UO_2 is being heat treated to determine whether the chloride content can be reduced sufficiently for cold testing and possible in-reactor experiments.

High Heat Flux Test Element. A UO_2 fuel rod (GEH-12-33) is being irradiated to generate 655 watt/cm^2 (2.1 x 10^6 Btu/hr-ft^2) surface heat flux. Calculations indicate that fuel within 83% of the radius is molten and the maximum fuel temperature is ~ 3700 C. The test element will be discharged from the ETR P-7 Loop on October 28, after accumulating approximately 7700 MWD/T_{fuel}.

Particulate UO_2 (enriched to 50 wt% U-235 in U) is contained in 1.18 cm (0.465 inch) OD x 0.099 cm (0.039 inch) thick molybdenum cladding that is externally roughened by threading (25.2 per cm, 0.023 cm deep) to improve heat transfer between the cladding and the coolant.

ThO_2-PuO_2 Irradiation Studies. Twenty capsules containing hydrogen sintered ThO_2-PuO_2 fuel pellets (five compositions ranging from 2.23 to 18.48 wt% PuO_2) are being irradiated in the MTR and ETR. The ThO_2-PuO_2 fuel pellets were prepared by wet ball milling ceramic grade ThO_2 and plutonium oxalate to obtain intimate mixtures. Autoradiographs (500X) made from polished surfaces of the sintered pellets show that the PuO_2 is homogeneously distributed. Duplicate pellets are being oxidized in air at 800 C to measure weight increases caused by oxidation of Pu_2O_3 to PuO_2. The oxidized ThO_2-PuO_2 pellets will be irradiated to compare their irradiation performance with that of hydrogen sintered pellets.

PuO_2-SS Cermets. The first group of six pneumatically impacted PuO_2-SS cermet pellets ($\frac{1}{2}$-inch diameter x 1 inch long, 20 vol% PuO_2 dispersed in 304 stainless steel) were made using a split die. The pellets were easily removed from the die and need only be ground to final dimension. The uniformity of PuO_2 dispersion will be determined by metallographic examination and autoradiography. Density measurements will be made on the finished pellets prior to their encapsulation for irradiation testing. An invention report was submitted on the design of a split die for fabricating fuel pellets by pneumatic impaction.

Concept for Fuel Grids. An invention report, HWIR-1659, was submitted describing the use of fuel grids for reactor core loadings.

Plutonium Handbook. The revised version of Chapter 2 of Section VI of the Plutonium Handbook, to be submitted for presentation at the 1964 Geneva Conference, was declassified.

Corrosion and Water Quality Studies

Out-of-Reactor Checkout of Zircaloy-2 Samples. An out-of-reactor system checkout was performed in which Zircaloy-2 samples were enclosed in a titanium capsule and exposed for 1000 hours to deionized water at a temperature of 360 C (680 F) and pressure of 3000 psi. The temperature was maintained steady by a current-adjusting type controller driving a magnetic amplifier; the water pressure was maintained by pumping water through the capsule at the rate of 65 milliliters per hour - approximately 100% refreshment per hour. No attempt was made to control the dissolved gases in the feed water; the dissolved oxygen concentration averaged about 6 ppm. Twenty Zircaloy-2 samples gained an average of 40 mg/dm^2. This value is higher than weight gains of 29 mg/dm^2 normally obtained for Zr-2 in stainless steel autoclaves under similar conditions. Variations in sample surface preparation, metal fabrication history, electrochemical effects between samples and vessel, or water chemistry could account for the difference.

Zirconium Concentration Data. The total zirconium concentration in the PRTR primary coolant is being monitored routinely as a means of detecting fretting or other types of accelerated corrosion of Zircaloy as it occurs. During the past month grab samples were collected, and emission spectrographic techniques were used to obtain the analytical data.

The results indicated that the zirconium concentration was within the 0-1 ppb range considered normal for this system, except on one occasion when a value of 2.4 ppb was observed. It is unlikely that any significant amount of fretting corrosion occurred during this period.

These results confirm the previously reported trend toward less frequent concentration excursions in this system. During the past three months there were only two occasions where abrupt concentration increases occurred that may have been indicative of fretting attack. One of these periods occurred during startup after an extended outage and could have been due to charge-discharge and/or tube inspection work. In any case the concentration excursion frequency is now much lower than it was during the first five months of 1963. This is presumably due to the larger fuel element support pads which are being employed in increasing number.

Ex-Reactor Fretting Corrosion Tests with PRTR Fuel Elements. The PRTR fuel element with the 360° ring support was discharged from TF-7 after testing for approximately 192 days with external vibration to induce fretting. Very little fretting occurred on the pressure tube. The rod wire wrap with the most severe fretting was removed and measured. The penetration was 16 mils (original diameter 71 mils). Measurements on the ring supports were inconclusive because of mechanical deformation. Accurate measurements will be obtained during the metallurgical examination for hydriding effects.

Reactor Components Development

PRTR Pressure Tubes. The eleventh Zircaloy-2 PRTR pressure tube discharged from the PRTR had a 20-mil deep flaw at 11 feet, 5 inches from the flanged end. A 20-inch long piece of the eleventh tube with this flaw at its mid-length has been burst tested at room temperature. The piece burst at a pressure of 9200 psig. The burst occurred at or very near the 20-mil flaw. The burst was classified as a "pinhole" type because, after penetrating the wall, it propagated only a short distance along the tube. The dimensions of the burst, as viewed from the OD, were estimated to be 0.25 inch long by 0.005 inch wide. The fast neutron exposure in the region of the failure was estimated to be 10^{18} nvt ($E > 1$ Mev).

This room temperature burst test demonstrated two things: (1) the burst pressure was more than eight times greater than PRTR operating pressures, and (2) the failure was ductile in nature.

The highlights of the twelfth meeting of the General Electric Technological Hazards Council contained an announcement of acceptance by the GETHC of a decrease in the rate of removal of Zircaloy-2 PRTR pressure tubes for destructive examination. The discharge rate of one tube each operating month will go to one tube each 90 operating days. This rate change was formally proposed in August 1963 through the medium of report HW-78456 entitled "PRTR Zircaloy-2 Pressure Tube Surveillance Program."

Pressure Tube Monitoring. Fourteen process tubes were inspected this month. While all of the process tubes exhibited new areas of fretting corrosion, none was very severe. Most of the tubes inspected this month were fringe tubes on the south side of the reactor core. A comparison of these tubes with center zone tubes over comparable operating periods indicates that the incidence of new fretting marks of all types may be slightly less in the fringe tubes. This may be due, in part, to the fact that there are generally more UO_2 type elements in the center zone tubes. There are, however, some exceptions to the above general conclusion, probably due to the fact that the number of times fuel elements are discharged and recharged into the reactor is not constant for all tubes.

The tube in process channel 1558 was again inspected this month to examine more fully a suspected defect. Detailed examination showed the suspected defect to be of the raised type. This defect is about 20 mils high, about 1/8 inch wide, and approximately 1 inch long. Because of the suspected defect, the tube was replaced.

Mark III monitoring equipment is now substantially completed, and plans are being made to move this equipment into the PRTR early next month.

Shim Rod Development. Final assembly of the second generation shim rod driving head is in progress. The cast aluminum heat sinks and three zirconium lead screws have been fabricated and shipped.

Fretting Corrosion Investigation. Operation of the EDEL-I Loop facility has been resumed after a shutdown for pump motor repair. No fretting action has been noted on the pressure tube or fuel element after 15 days of operation at PRTR coolant conditions. The pads on the existing fuel element have been reduced in area in an effort to induce fretting corrosion.

The instrumented fuel element, which is the basic tool to be used in defining the conditions under which fretting will occur in the PRTR geometry, has not been completed and apparently will not be available until the end of November.

An autoclave with a 2 cps striker impinging on a zirconium sample has been piped into the EDEL-I Loop to determine, together with the ones installed on PRTR and TF-7 Loop, the effect on water quality on fretting or impact corrosion.

Tests utilizing a new ultrasonic translator contact probe were negative for sounds of fuel element and pressure tube impact during operation.

Design Analysis

PRCF-EBWR Safety Studies. Refined calculations of the time delay in the Doppler temperature coefficient associated with mixed PuO_2-UO_2 fuels are being carried out. Measurements of the PuO_2 particle size distribution for -325 mesh material have been obtained from Fuels Testing and Analysis Operation. Additional autoradiographs and thermal conductivity measurements have been requested for the dyna-packed fuel. A formulation for calculation of the fission fragment energy transport into the UO_2 was also completed.

Thermal Hydraulic Studies

Calculation of Pressure Transient Following Loss of Pumping Power. A simplified thermodynamic model was developed to calculate pressure transients following a loss of primary pumping power with a scram at the high pressure trip. The model, as reported last month, assumes a uniform pressure increase in a constant volume system consisting of a subcooled liquid and saturated vapor. The effects of the pressurizer safety relief valve discharge can also be included.

Calculations have been made, assuming volume expansion at constant pressure, to compare with earlier calculations. Agreement of the total volume increase was within about 10%. The constant pressure calculation is unrealistic, however, because fluid properties change with pressure.

A pressure transient at 100 Mw and 11,600 gpm flow with a fast pump coastdown was considered. A reactor inlet temperature of 477 F was chosen to keep the coolant from the hot tube 5 F below the saturation temperature if the flow in the hot tube was reduced 10%. For these conditions the relief valves did not blow. The pressure increase was about 200 psi.

The thermodynamic model mentioned above has also been modified to calculate the pressure transient associated with boiling convection cooling following a complete power outage at the PRTR. Boiling is assumed to start where liquid-phase natural convection would normally occur. The system is assumed to become vapor bound and the natural convection flow stops. Using a minimum pump coastdown from 11,600 gpm at 100 Mw power and current operating procedure, the pressures and cooling requirements are well within acceptable limits.

Calculations to date indicate that, from a thermal hydraulic standpoint, the increase of the PRTR power from 70 to 100 Mw would probably be possible using current operating procedures and equipment.

Air Cooling in the PRTR Fuel Element Examination Facility. The Fuel Element Examination Facility (FEEF) of the PRTR is designed to provide for remote examination of irradiated fuel elements from the reactor. The cooling necessary to remove the fission product decay heat of 19-rod bundle fuel elements is provided by a transverse flow of air. Experimental laboratory tests were made which indicated that a fuel element generating 20 kw would not have a surface temperature exceeding the limit of 572 F.

Early tests of the FEEF showed some deficiencies in the distribution of air flow. An arbitrary limit of one kw heat generation was placed on any fuel element to be placed in the FEEF. Modifications were made to the FEEF to correct the maldistribution of coolant air flow.

Data, consisting of air velocity measurements downstream of a dummy fuel element, were taken after the modifications. The data were submitted to Thermal Hydraulics for evaluation with the intent to raise the maximum permitted heat generation rate to 10 kw. The data show that the flow distribution axially is adequate. However, the data do not allow evaluation of air leakage around the sides of the fuel element. Side baffles are provided to minimize side leakage. The laboratory tests used side baffles 1/8-inch from the fuel element. The distance between the fuel element and side baffles in the FEEF is not known, but it is estimated that it could be as much as 3/8 inch. Calculations were made which showed that a heat generation limit of 5 kw would be appropriate if the side gap is greater than 1/8-inch but no more than 3/8-inch.

<u>Boiling Burnout Experiments in the PRTR Rupture Loop</u>. A possible experiment in the PRTR Rupture Loop is one in which a fuel element is deliberately operated into boiling burnout. Such experiments would provide information concerning the applicability of boiling burnout data from electrically heated models to actual reactor operating conditions. Furthermore, they would provide valuable information concerning the consequences of boiling burnout.

A review of the possibility of in-reactor boiling burnout experiments shows that the use of temperature monitoring devices is not feasible to detect burnout. The mode of detection would have to be with the fuel element rupture detection equipment of the loop to detect fuel element failures resulting from boiling burnout.

However, calculations show that attaining boiling burnout in the reactor would be difficult. If the loop were operated at one-fourth the normal coolant flow rate, the flow rate for which electrically heated test section data are available, and if the loop inlet temperature were adjusted to just below saturation, the electrically heated data predict burnout would not occur at tube powers less than 1400 kw. Such a tube power cannot be attained in the rupture loop without a highly enriched fuel element.

2. Plutonium Ceramic Fuels Research

Plutonium Cermet Electron Microscopy. A compacted cermet specimen containing 40 vol% PuC in a tantalum matrix was successfully replicated for electron microscope examinations before and after heat treatment at 1400 C for 11 hours in vacuum. The PuC-Ta interfaces showed no evidence of interaction or diffusion between phases.

A three-step process using, successively, Faxfilm, polyvinyl alcohol, and evaporated carbon produced uncontaminated positive replicas of the polished surfaces.

Plutonium Ceramics Irradiations. Two capsules containing PuN, GEH-14-405 and 407, have been shipped to NRTS, Idaho Falls, for irradiation in the MTR and will be inserted into the reactor for MTR cycle 199.

The status of the Plutonium Ceramics irradiation program is as follows:

GEH-14-I	Material	Reactor	Exposure*(FPD)**	Goal Exposure(FPD)**
405	PuN	MTR	In for cycle 199	1000
406	PuN	MTR	48.5	100
407	PuN	MTR	In for cycle 199	1000
408	PuC	ETR	48.1 (out)	(out)
409	PuC	ETR	70	1000
411	PuO_2	MTR	48.5	1000
412	PuO_2	MTR	48.5	100
413	β-Pu_2O_3	MTR	48.5	1000

*As of end of MTR cycle 198 (10/21/63), and ETR cycle 58 (10/28/63).
**Full power days of reactor operation (each FPD is roughly equivalent to 1000 MWD/T exposure).

UN-PuN Fuel Tests. A stainless steel clad, irradiated capsule containing UN-20 wt% PuN pellets is being assembled for irradiation in the MTR. The compatibility of 304 stainless steel with UN-20 wt% PuN pellets was checked by heating the nitride to 1000 C in a stainless steel crucible for 4 hours in vacuum. No visual evidence of a reaction between the fuel material and stainless steel was seen.

Thermionic Emission Studies. The thermionic emission apparatus as originally designed has been completed and will be operated as soon as the high voltage power supplies are installed and checked out. This design is based on the Schottky plot technique and should be usable up to temperatures in the region of 2000 C, at field strengths up to 10,000 volts/cm, and under vacuum conditions of about 3×10^{-6} Torr.

A new experiment has been designed following the work by Webster at GERL using a constant potential difference technique. This system will be bakeable and operate at 10^{-10} Torr or less.

3. Ceramic (Uranium) Fuel Research

Thermal Conductivity of UO_2. Recent radial flow, thermal conductivity measurements for single crystal UO_2 confirm previously reported results that internal radiation contributes to heat transfer in large grain, stoichiometric UO_2. A maximum in thermal conductivity occurs between 1000-1200 C. Other preliminary data indicate a second increase in thermal conductivity above 1500 C.

Thermal Diffusivity Measurements on UO_2. Thermal diffusivity measurements were made in the temperature range from 600 C to 1200 C on a 30.4-mil single crystal UO_2 specimen using a pulsed energy method. The source of the thermal pulse is a ruby laser. The relationship between thermal conductivity and temperature shows a maximum in conductivity between 1000 C to 1125 C. Compared to the previous data, this peak is shifted to a slightly higher temperature.

Thermal diffusivity measurements were also made on a specimen of pneumatically impacted UO_2, 33.3-mil thick. For this specimen a (poorly resolved) maximum in conductivity may occur between 1000 C and 1050 C.

Electron Microscope for Ceramic Materials Research. Installation of a second electron microscope column operating from the original primary power supply has been completed. This instrument, a result of close cooperation of manufacturer's engineers and Hanford personnel, is the only known microscope of this design. For 50% additional cost, available instrument time has been nearly doubled by greatly reducing conflicts between reflection and transmission microscopy experiments on toxic and nontoxic ceramic materials.

Image Rectification for Reflection Electron Microscopy. A photogrammetric image rectifier has been installed. The image rectifier provides optical correction for the one dimensional foreshortening of micrographs that is inherent in reflection electron microscopy.

Induction Plasma Torch. The induction plasma torch, capable of operating in a controlled atmosphere, has been developed into a usable source for high temperature research. The plasma has been operated continuously for several hours with no apparent damage to the enclosure. Pilot experiments with coarse alumina powder passing through the plasma resulted in crystal growth of seed crystals 1/8 inch x 1/4 inch.

Materials and Information Exchange. Two single crystals of vapor deposited UO_2 were received from EURATOM for use in fundamental studies. Several single crystals of UN prepared by arc-melting were received from BMI.

Tungsten Cladding Evaluation. Evaluation of tungsten cladding tubing for ultra high temperature irradiation was begun. Three general types are available: deposited, spark discharge machined, and extruded tubing. Tubing made by hot swaging of electro-deposited material will become commercially available in the near future. Metallographic examination of various types and sizes of tubing revealed that tubing made by spark discharge machining from solid rod has a structure relatively unchanged from that of the original rod. In the smaller diameters, a small, equiaxed grain structure is available, but the grain size in rod diameters approaching 2 cm is excessive. Vapor deposited tubing has an effective wall thickness of only one grain, a factor that causes extreme brittleness. One vendor reports that a method for refining the grain structure of vapor deposited tubing has been developed.

High Temperature Irradiation Studies. Post-irradiation examination of a fully enriched UO_2-80 vol% tungsten cermet fuel plate revealed that very little UO_2 was lost from the plate during irradiation at 3000 C. X-ray diffraction analysis of a deposit on the interior surface of the outer container showed the deposit to be tungsten. No UO_2 lines were observed. Density and mass changes during irradiation indicate that 0.37 gram of tungsten was vaporized from the surface (cladding) of the plate and that apparent loss of UO_2 was nil.

Thermal expansion of the UO_2 during irradiation caused enlarging of the particle cavities. Pores developed in the interior of the irradiated UO_2 particles suggest that the UO_2 was above its melting temperature (2790 C) during irradiation.

Beryllium-Clad Fuel Elements. Metallographic examination of recent beryllium welds revealed apparently good bonds, with relatively small recrystallized grains in the weld area. However, detailed nondestructive testing indicates the joints are not yet adequate for in-reactor testing, because of microcracks in the upset metal. A preheat and postheat controller which is being installed may eliminate this problem. A transformer to supply additional power to the magnet of the magnetic force welder should also help.

Four fuel rods were vibrationally compacted to 87% of TD, using Nupac UO_2.

4. Basic Swelling Program

Irradiation Program. Two general swelling capsules have continued their irradiation at respective control temperatures of 500 and 575 C (932 and 1067 F). Difficulties have been encountered with one of the power control systems allowing one of the capsules to drop in temperature by as much as 50 C (90 F) on several occasions. Two capsules were bench tested which contained eight heat transfer fins rather than the customary six. Surprisingly little or no difference in the heater power-temperature relationship was obtained.

Four additional capsules are under construction which will contain high purity uranium specimens of various size and shape.

Post-Irradiation Examination. High purity uranium specimen 17-A-1 which was irradiated at 575 C (1067 F) to 0.5×10^{20} fissions/cc (0.1 at% B.U.) and specimen 18-A-1 irradiated at 525 C (927 F) to 0.15×10^{20} fissions/cc (0.03 at% B.U.) were post-irradiation annealed at 625 C (1157 F) for 100 hours. Optical and electron metallography show that no qualitative change has taken place in extent or distribution of porosity. No change in grain size or in the number of deformation twins was observed as a result of the annealing. Density is yet to be measured, but in view of the metallographic examinations no change in density is expected.

Thorium. Density measurements made on the two irradiated thorium specimens annealed at 950 C (1742 F) for 100 hours reveal a volume change of about 1% for the low burnup specimen (0.6×10^{20} fissions/cc - 0.18 at% B.U.) and of about 22% for the high burnup specimen (2.5×10^{20} fissions/cc - 0.92 at% B.U.). These values agree qualitatively with the optical metallography. Replicas from the etched surfaces are being processed for electron metallography.

Thorium is observed to resist volume change to a higher temperature than does uranium. This is in general agreement with other reported observations. It is believed by many that the cubic crystal structure possessed by thorium is responsible for its enhanced resistance to swelling, as compared with that of the anisotropic, orthorhombic alpha uranium. However, if results with the high burnup thorium specimen are compared with typical uranium data obtained by post-irradiation annealing in the proper frame of reference, it is concluded that both metals show the same fission gas-induced volume increase. The 950 C (1742 F) annealing temperature employed here with thorium is the same homologous temperature (i.e., fraction of the absolute melting point) as is 575 C (1067 F) for uranium. When the data at the same homologous temperature are normalized to unit burnup respective "R" values for thorium and uranium are 24 and 21. Thus, it is the high melting point of thorium (approximately 1700 C - 3092 F) rather than the isotropic crystal structure that is responsible for its apparent superior swelling resistance. A complication arising with uranium, of course, is the fact that volume changes can occur in-reactor which are not caused by fission gas porosity but rather as a consequence of the fission event itself. Thorium apparently does not exhibit this behavior.

5. Irradiation Damage to Reactor Metals

Alloy Selection. The last shipment of materials procured for test specimens for the Irradiation Effects on Reactor Structural Materials Program was received this month. Compilation during the past month of the material fabrication history indicated that some information was yet to be obtained from the vendor. Contacts to obtain this information are presently being made.

Tensile specimens of Hastelloy N, Inconel 625, Inconel 702, Inconel 718, R-235 and R-27 from the ETR hot water loop facility are presently being examined by Radiometallurgy Laboratory. Additional specimens of R-235, R-27, Hastelloy N, Inconel 625, Cb - 1 Zr, Cb-752, and Ta - 10 W irradiated in the ETR G-6 cold water facility are also being examined. To date, each specimen has been weighed and hardness measurements taken. Tensile tests on these materials are now being initiated and will be followed by metallographic and electron microscopy examination.

Four graphite boats containing specimens of Inconel 600, Incoloy 800, Inconel 625, AISI 406, Hastelloy X-280, AISI 348, R-235, Hastelloy N, Inconel 702, Inconel 718, and TD nickel were charged into the reactor for irradiation. After reactor startup, temperature monitors indicated a specimen temperature of 690 C (1274 F). These specimens will be irradiated in a gaseous environment to exposures reaching 5×10^{20} nvt (>1 Mev).

<u>In-Reactor Measurement of Mechanical Properties</u>. During the last month two in-reactor creep capsules in which steady state data had been obtained were used to verify the in-reactor activation energies for 20% cold worked Zircaloy-2. During a reactor outage activation energies were measured in both capsules. In one capsule four determinations were made by temperature cycling between 395 C and 415 C (743 F and 779 F) at 25,000 psi stress. In the other capsule only one determination at 375 C (707 F) and 35,000 psi was conducted. All activation energies found during the outage averaged close to the self-diffusion value of 60,000 cal/mole. About four days after the reactor startup identical experiments were performed during irradiation. In both capsules the measured activation energy was in good agreement with the previously determined "during irradiation" value of 86,000 cal/mole. The fact that different activation energies are observed with neutrons off and with neutrons on in the same capsule proves neutron irradiation does affect activation energies.

It has been suggested that the activation energies during irradiation were due to the aging phenomena sometimes observed ex-reactor. If aging were producing the high activation energy during irradiation, one would observe the high activation energy with the reactor off, also. This is not the case; therefore, aging does not produce the high activation energy. For a consistent explanation of the high activation energy, one must return to the theory proposed earlier, which involves motion of dislocations past irradiation produced obstacles. The number of the obstacles depends on temperature.

The theory used to rationalize the 86,000 cal/mole activation energy requires that a certain temperature exists where the 86,000 cal/mole value can no longer be observed. This temperature corresponds to the condition where the time required to change irradiation produced defect levels by annealing is less than the time required to change temperature in a temperature cycle activation energy determination. From previous data this temperature was believed to be around 325 C (617 F). This temperature was verified during the month by conducting activation energy studies around 325 C (617 F). Below 325 C (617 F), 60,000 cal/mole was observed; above 325 C (617 F), 86,000 cal/mole was observed.

Another in-reactor capsule containing 20% cold worked Zircaloy-2 was charged during the month. A test was started at 375 C (707 F) and 20,000 psi and has accumulated about 100 hours. The creep rate at 100 hours is 7.1×10^{-6}/hr. The above conditions were selected to verify the position of the 20,000 psi, log $\dot{\epsilon}$ vs 1/T plot. Earlier studies leave some doubt as to the true position of this line.

Two creep capsules with annealed 304 stainless steel specimens are under construction and will be ready for charging during the next reactor outage. Metallographic studies and tensile tests are being conducted on three 5/8-inch diameter bars randomly picked from a test stock of 304 stainless steel. Comparisons are being made between the as-received material and material that has been annealed at 2000 F (1093 C) for 15 minutes. Metallographic examination disclosed little difference between the as-received and annealed material. Tensile tests were performed using 0.185-inch diameter by 1.20-inch gage length KAPL type specimens. The yield strength for six annealed specimens gave an average of 32,650 psi, while three as-received specimens had a yield strength of 38,300 psi. The vendor quoted a yield strength of 50,000 psi for the as-received material based on tests conducted with modified ASTM 0.505-inch diameter tensile specimens. Two tests were conducted on site using this specimen goemetry. The average yield strength for these tests was 49,650 psi. The difference in the yield strength of the as-received material using KAPL specimen versus the modified specimen is attributed to surface cold work put into the rod by the vendor after it was annealed.

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

At the conclusion of ETR Cycle 58, six capsules containing notched beams of Zircaloy-2 were discharged from the K-7 position. Three capsules containing tensile specimens of refractory alloys were discharged from the G-6 position and one quadrant containing tensile specimens of AISI 406 was discharged from the G-7 hot water loop. The notched beams will be used in fracture studies currently in progress.

Four Linear Variable Differential Transformers (LVDT's), which were specially designed and constructed for use at 300 C (570 F), were received from Datronics Corporation. These LVDT's were mounted on Baldwin B-3M clip-on type extensometer frames. This assembly was further modified to provide a means of remotely detaching the extensometer from the specimen during a test. This extensometer provides an accurate measurement of strain in constant temperature tests up to 300 C (570 F), which is the approximate irradiation temperature for specimens exposed in the ETR G-7 hot water loop.

Load maintaining and load cycling controls have been added to both the 60,000 and 200,000 pound capacity Baldwin Hydraulic Testing machines. These features provide control to within ± 1% of the full scale load for the range selected. Loading rate is controlled manually by adjustment of a needle valve on the hydraulic system.

In order to determine the structural stability and effect of neutron irradiation on phase transformations which occur in metastable austenite, magnetic measurements were made on several tensile blanks and cold rolled strips of AISI 304 and 348 stainless steels. An Aminco-Brenner Magna Gage was used to measure the force required to lift a standard magnet from the specimen. Measurements on materials containing from zero (annealed) to 70% cold work show a considerable increase in magnetic susceptibility with increasing amounts of cold work. These measurements were used to establish relationships which predict the amount of austenite that will be transformed by cold working and subsequent neutron irradiation. Preliminary results on material cold worked to various levels and annealed in vacuum at 300 C (570 F) for various time periods indicate a significant change in the martensite content as a result of annealing at this temperature. Measurements of magnetic susceptibility are being made on both the annealed and cold worked AISI 304 and 348 irradiated at approximately 300 C (570 F). Considerable difference in the magnitude of response to the annealing was noted between the unirradiated cold worked AISI 304 and 348, the greater absolute change being exhibited by the 304. Based on previously reported experiments, this difference in response can be attributed to the difference in chemical composition of the two materials. Since cold work, temperature, and chemical composition strongly influence the amount of metastable austenite which will be transformed, it is not too surprising to find considerable scatter in the test data reported for these materials.

A study to determine the optimum condition of several nickel base alloys for subsequent service in a nuclear environment was initiated. Typical high temperature corrosion resistant alloys such as Inconel 600, Inconel X-750, and Hastelloy X-280 will be examined. A variety of pre-irradiation aging treatments varying from 1000 to 1700 F (538 to 927 C) and for time at temperature intervals up to 100 hours were performed on Hastelloy X-280 during the past month. Strength increases and reduced ductility were noted as the aging temperature was increased. There were no significant changes noted for aging at different periods of time at temperatures to 3100 F (705 C). Above 1300 F (705 C), Hastelloy X-280 appears to develop some notch sensitivity.

Below about 1300 F (705 C), the matrix and grain boundaries were relatively free of carbide precipitates. Above this temperature, precipitates of the M_6C type began to form in both the matrix and grain boundaries with increasing abundance. Fracture traces of the alloy aged at the higher temperatures were definitely of an intergranular nature, propagating from one carbide particle to another along the grain boundaries. Evidence of this type of phenomenon was also found in some of the specimens tested at a lower temperature.

Zircaloy-2 specimens irradiated in the G-7 hot water loop at a temperature of 280 C (540 F) to an exposure of 3.0×10^{21} nvt (fast) have been discharged and tested. Irradiation of these specimens was begun in August 1961 and completed in June 1963. A number of experiments containing both Zircaloy-2 and stainless steel were charged at the beginning of ETR Cycle 39 (August 1961) and are still being irradiated.

Preliminary results on the Zircaloy-2 tensile tests show the appearance of a yield point in this material tested in the rolling direction after an exposure of 3.0×10^{21} nvt (fast). A yield point appeared in testing the transverse specimens after exposures of 1.0×10^{21} nvt (fast). Neither the transverse nor longitudinal specimens exhibited a yield point prior to irradiation or after exposures up to approximately 1.0×10^{21} nvt. The strength of the material continues to increase and the ductility tends to decrease for these longer exposures.

A cooperative effort between the Reactor Metals Research and Physical Measurement groups was initiated to determine applicability and sensitivity limits of current concepts and equipment used in determining the onset of fatigue damage in metals. One goal of this study is to evaluate the magnitude of that fraction of fatigue life that elapses before the damage becomes detectable. It is of considerable interest to determine those differences which occur when a structure is deformed by conditions of fatigue as opposed to a single cycle or "static" loading condition. This type of information is necessary in arriving at a method for determining fatigue damage on a structure which is in service.

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

Past studies of radiation damage in iron have led to conflicting conclusions about the defect responsible for the modification in properties. For example, studies based on measurement of resistivity reveal damage effects not seen in measurements of mechanical properties and vice versa. Since this study is primarily concerned with the changes induced in the mechanical properties, it is imperative that a method be devised to identify and characterize this specific type of defect. It is hoped that such a method has been found in the application of rate theory to the process of plastic deformation. This method combines chemical kinetics with dislocation theory to describe plastic deformation by the rate of movement of dislocations over barriers to their movement. At low temperatures where the picture is not clouded by recovery and rearrangement of the dislocations and obstacles during deformation, the deformation process can be described in terms of the stress applied to the dislocations by the tensile machine and the thermal fluctuations superposed on this stress which control their rate of movement over the barrier. Thus, the deformation rate is described by an Arrhenius equation, $\dot{\epsilon} = A \exp(-U/kT)$, where $\dot{\epsilon}$ is the strain rate, A is a constant, and U is the activation energy for thermal activation over the barrier. U can also be described as $U = U_o - V\tau^*$, where U_o is the total activation energy for surmounting the barrier (applied stress plus thermal), V is the activation volume describing the barrier, and τ^* is the effective stress on the dislocation.

These quantities can be derived from low temperature tensile tests in which the change in stress associated with a change in strain rate and the change in stress associated with a change in temperature are measured.

Equipment for performing these tests has been built and the initial tests to evaluate their capabilities completed. A dummy sample was strain gauged to determine the eccentricity in loading for the tensile fixtures. Strain gauge readings from points 90 degrees apart on the specimen were within 10% of each other indicating satisfactory alignment. All specimens pulled to date have shown a sharp yield point, also indicative of good alignment and a "hard" fixture. Use of the zero suppression circuit of the Instron permitted a stress resolution of 20 psi during strain rate change tests.

A series of experiments to determine the effect of initial dislocation configurations on the activation volume was performed. One sample was pre-strained 3.5% at room temperature at a strain rate of 0.002 min^{-1}, another at 2.0 min^{-1}, and still another was deformed at 77 K at a rate of 0.002 min^{-1}. Following these pre-strains, the samples were then cycled between strain rates of 0.002 and 0.02 min^{-1} in a liquid nitrogen bath. The various pre-strain treatments had little or no effect on the activation volumes but did influence the mode of deformation. Twinning was observed in the sample pre-strained at 77 K, and also at the higher strain rate (0.02 min^{-1}) for the sample pre-strained at two inches min^{-1}. If twinning occurred in the sample pre-strained at 0.002 min^{-1} at room temperature, it did not produce serrations in the stress-strain curve characteristic of twinning. All subsequent experiments will be standardized at a pre-strain of 3.5% at room temperature at a strain rate of 0.002 min^{-1}.

A test was also run in an ethyl alcohol-dry ice bath (200 K) to determine activation volumes at this temperature and stress level. Using the data obtained from these tests and from the published results of Basinski and Christian, a total activation energy to surmount the barrier (U_o) of 0.4-0.5 ev was calculated for both the deformation at 77 K and 200 K. Above 300 K, however, the deformation process does not appear to be a thermally activated process. The calculations are only estimates because of certain approximations used, but they are sufficiently accurate to indicate one mechanism is controlling the thermally activated movement of dislocations over this temperature range. Additional experiments will be confined to the temperature range 77-300 K.

ATR Gas Loop Studies

Test Section. A report, "The Model High Temperature Gas Loop Test Section - An Interim Report," HW-79159, has been written to relate the development of the design of the test section and describe several areas where results from testing and operating experience can supply information applicable to the ATR test section. General features now known for the ATR test section which have been incorporated into the model gas loop test section are the same diameters and materials, possible tube closures, and insulations compatible with attaining and maintaining ultra-pure helium in the coolant system.

The test section consists of the following: a tube of type 347 stainless steel to bear the internal pressure, insulation at the lower and middle sections of the tube to maintain a relatively low pressure tube temperature, and a Haynes alloy 25 flow buide tube at the top of the test section for attemperation. Access to the top of the test section is provided by a Grayloc-type blind flange.

Construction has continued on the Mark I test section for the model gas loop. The Haynes alloy 25 attemperator guide tube has been completed and machining has started on the transition joints for the various sizes of tubing used in the test section. Materials yet to be received are the metallic foil insulation and the bellows for sealing between the top of the shell and attemperator flow guide tube.

Heat Transfer Studies. The possibility of using water cooling instead of gas attemperation to reduce the temperature of the 1093 C (2000 F) gas leaving the test section was examined. The somewhat cursory appraisal indicates that water cooling has sufficient merit to warrant consideration for the ATR gas loop by the architect engineers.

Information was supplied to Babcock and Wilcox to permit determination of the film coefficient correction factors for cooling helium.

Analysis of the tensile specimen holder by both the Washington State University consultants and Babcock and Wilcox indicates that the pressure drop will be about a factor of 10 greater if the coolant flow around the specimens acts as a series of highly turbulent constricted orifices rather than as a series of simple expansions and contractions. Since analytical results do not show which flow mode will prevail, apparatus has been set up to experimentally measure the pressure drop with air used in place of helium. Initial results will be available the last part of the month.

Helium Purification. A document, HW-79275, entitled "Design Criteria for Helium Purification Plant for the ATR Gas Loop," was prepared and issued by J. H. Hoage, R. E. Westerman, and D. C. Kaulitz. The document was prepared following consultation with Dr. Frank C. Wood and Hans F. Enzmann, of O.E.C.D. Dragon Project, England, in a conference held at Hanford on September 30, October 1 and 2, 1963. Other offsite visitors attending the conference were: K. A. Krauss, Ebasco Services, New York, N.Y.; W. D. Ennis, Idaho Operations Office, AEC, Idaho Falls, Idaho; and R. J. Daly, LASL, Group K-4, Los Alamos, New Mexico.

Frequency Changer. The frequency of the electrical power produced by the frequency changer is controlled by manually adjusting a variable resistor. Some of the experiments that will be performed using the model loop require that the flow of helium in the loop respond automatically to temperature changes in the loop. Such response to temperature might be accomplished by cascade control, using a change in temperature signal to automatically vary the frequency set point. The degree of coupling between the induction motor and the synchronous generator provided by the eddy current coupler would then be controlled by comparing the actual electrical power frequency with the desired electrical power frequency which is set manually and is modified automatically in response to a change in temperature signal. A scheme to accomplish this cascade control has been designed.

Stress-Rupture Tests. A program has been started to evaluate the stress rupture properties of Haynes 25 at 2100 F (1149 C). This material is being considered for use in the ATR as a structural material for the heater exit ports.

The induction heating furnace described in a previous quarterly report, HW-77954, has been adapted for these tests. The temperature is automatically controlled and recorded using a magnetic amplifier and Brown recorder. The load is applied directly to the specimen by the use of a pan of weights attached to the bottom pull rod. Elongation is measured with a compression type extensometer.

The following results have been obtained to date:

Stress (psi)	Time-to-Rupture (hrs)	Elongation (%)
3230	34.0	--
4157	8.5	6.9
4350	7.0	--
4914	5.8	7.9
5498	2.8	7.2
6010	0.7	7.2

Long term tests are now in progress to predict rupture life to 1000 hours.

Gas Chromatograph. Evaluation of a gas chromatograph utilizing a new micro ion cross-section detector continued this month. It was determined that this detector responds very well to H_2, N_2, O_2, CO, CO_2, and CH_4 in helium. After replacing the molecular sieve column furnished by the vendor with a new one and adjusting the flow rate it was possible to achieve a sensitivity of about 0.5 ppm per chart division for all six gases with a lower detection limit of at least one or two ppm. No extremely dilute standard gases are yet available to test below 10 ppm.

The good results with the detector do not extend to the balance of the instrument, however. The temperature programming feature for elution of CO_2 from a molecular sieve packed column analysis furnished by the vendor proved unsatisfactory for trace levels of CO_2 producing severe base line shifts and poor reproducibility. Modification to permit CO_2 analysis off of a silica gel column is more satisfactory. The original valving system for gas sampling is not sufficiently leak tight for trace analysis making the instrument inaccurate for O_2 and N_2. The valves have recently been modified to reduce air in-leakage. Operation of the sample valving system produces base line shifts probably due to variable pressure drops within the piping. Addition of a pressure regulating flow controller may help this problem. Difficulties in the valve drive mechanism were eliminated by remachining the brass bearings for teflon inserts. It has also been determined that the column temperature measuring system is highly inaccurate although close temperature control is not required with the ion cross section detector.

Corrosion Studies. Samples of Haynes 25, Hastelloy X, and Inconel 600 are being tested in environments of various oxygen pressures, at 1200 C (2192 F). Hastelloy X is much more oxidation resistant than Haynes 25 at oxygen pressures > 1 mm O_2. However, if the pressure is allowed to drop in the system during the course of a run, both materials begin to lose weight at ~ 0.3 mm O_2, presumably due to a combination of oxide spalling and volatilization of oxide and/or metal. If the system pressure is maintained at 0.002 mm Hg or less during the entire course of the run, both Haynes 25 and Hastelloy X continuously lose weight, at a rate of about 0.8 mg/cm^2/hr for short times (~ 5 hrs), or a metal loss rate of about 0.035 mil/hr. Long-term metal loss data for Haynes 25 (~ 50 hrs) show an over-all penetration rate of 0.028 mil/hr under these conditions.

Continuous Total Impurity Analysis. The micro ion cross section detector used in the new CMO gas chromatograph has been found to be very sensitive to contaminants in helium. This suggests its application to the continuous analysis of total impurities in the ATR gas loop helium coolant. Without the use of a chromatographic column, no qualitative analysis would be possible; however, continuous quantitative analysis of total impurities including water vapor should be possible with at least 10 ppm sensitivity, perhaps better. This compares with 100-200 ppm for more conventional thermal conductivity detectors.

6. Gas-Cooled Reactor Studies

Inhibition of Graphite Oxidation. Studies are being conducted to determine the effect of small amounts of impurities on the effectiveness of the oxidation inhibitor, CF_2Cl_2. The EGCR graphite has been compared with CSF graphite. Because the latter is purer, it oxidizes at a slower rate.

It has been found that the inhibiting effect on CSF and EGCR graphite is not greatly different. The addition of 1/2% CF_2Cl_2 reduces the air oxidation rate of CSF and EGCR graphites by a factor of 3 and 4, respectively. Oxidation to 10% weight loss results in a large increase in the BET surface areas as indicated below.

CHANGES IN BET SURFACE AREA FOR OXIDATION AT 615 C, m^2/g

Graphite	Initial	Air	Air + 1/2% CF_2Cl_2
CSF	0.33	1.45	2.92
EGCR	0.38	2.06	5.06

When the instantaneous oxidation rates are adjusted to unit BET surface area, the specific oxidation rates at 10% weight loss for CSF and EGCR graphite are in good agreement, both in air and in air plus 1/2% CF_2Cl_2.

A change in oxidation temperature from 550 and 650 C for oxidation to 10% loss in air with 1/2% CF_2Cl_2 had no significant effect (± 10%) on the BET area. Consequently the activation energies on the basis of either instantaneous rates or specific rates per unit surface area are comparable. The activation energy for oxidation in air with 1/2% CF_2Cl_2 is 42 to 43 kcal/mole for EGCR or CSF graphite. This agrees within experimental uncertainty with the values found for air oxidation. A reduction of rate without changing the activation energy is consistent with the assumption that active oxidation sites are blocked by adsorption of the inhibitor as described in HW-67255.

The Radiation-Induced Reaction of Carbon Monoxide with Water. The rate of the reaction of carbon monoxide with water in a Co-60 gamma field was measured. Known quantities of redistilled C.P. carbon monoxide and outgassed distilled water were placed into an evacuated capsule which was sealed off and irradiated at constant temperature until 1 to 5% of the carbon monoxide had reacted.

After removal of water by cold trapping at 78 C, the volatile gases were analyzed on the mass spectrometer. Carbon dioxide and hydrogen were the only gaseous products detected.

The energy absorbed by the gases was calculated from the dose rate (determined by $Ce(SO_4)_2$ dosimetry) assuming an absorption coefficient proportional to the electron density.

The 100 ev yields (g values) for the formation of hydrogen and carbon dioxide are tabulated below:

T, °C	g_{H_2}	g_{CO_2}	T, °C	g_{H_2}	g_{CO_2}
115	2.6	6.7	350	12.3	13.7
150	2.4	7.3	385	22.1	23.1
200	5.1	8.0	405	11.2(?)	11.0(?)
250	7.3	8.7	425	36.3	--
300	9.0	9.6	450	50.8	51.4

Below 350 C log g is a linear function of 1/T but deviates in a positive direction above that temperature. The 100-ev yields of H_2 and CO_2 at temperatures below 350 C are:

$$g_{H_2} = 170e^{-3500/RT} \text{ Molecules/100 ev}$$

$$g_{CO_2} = 20e^{-850/RT} \text{ Molecules/100 ev}$$

The observation that CO_2 and H_2 are not formed in equal quantities below 350 C indicates that there are at least two competing reactions, one of which forms CO_2 but not H_2. At higher temperatures the yields of CO_2 and H_2 are essentially equal. The relatively high activation energy for hydrogen formation is also an indication of two or more reactions competing for the energy. The increased rate of the reaction at temperatures above 350 C is probably due to contributions from the thermally induced reaction.

The effect of inert gases on this reaction is currently being investigated. Preliminary experiments indicate that the energy absorbed by the inert gas is transferred to and induces reaction in the reactant mixture.

Thermal Oxidation of Large Graphite Samples by Water Vapor. To determine the effect of the graphite sample size on its rate of oxidation under conditions where diffusion of the oxidant into the sample may influence the rate, relatively large graphite samples are being oxidized in a flowing oxygen-free helium stream containing about 7000 ppm water vapor. A TSX graphite sample, 2 inches in diameter, 12 inches long, and weighing about 1060 grams was used. Oxidation rates were determined in the temperature range 730 to 840 C and for time periods of 50 to 75 hours duration.

The activation energy under these conditions is 48.6 kcal/mole. The complete rate expression is:

$$\text{Rate } (\text{hr}^{-1}) = 5.34 \times 10^{4} e^{-48.6 \times 10^{3}/RT}.$$

These results are in excellent accord with those obtained on a TSX graphite sample 12 inches long but only 1.5 inches in diameter and weighing about 600 grams. The rate expression determined for this latter sample under very similar conditions was

$$\text{Rate } (\text{hr}^{-1}) = 4.82 \times 10^{4} e^{-48.5 \times 10^{3} RT}.$$

It therefore appears that under the present conditions of oxidation the rate is nearly the same for 1.5 and 2-inch diameter samples. These measurements will be extended to samples whose diameters range from 0.5 to 3.5 inches.

<u>Annealing of High-Temperature Radiation-Induced Contraction</u>. A series of six transverse and six parallel samples of CSF graphite are being irradiated at the ETR at approximately 700 C to check the effect of high-temperature annealing on the contraction rate in subsequent irradiations. This study was prompted by the effects noted in similar experiments performed with lampblack-based samples.

The first irradiation of three reactor cycles resulted in a contraction of 0.1% in the transverse samples and 0.2% in the parallel samples. Three samples of each orientation were annealed and these, together with the unannealed samples will be re-irradiated. All of the annealed samples displayed an additional contraction of approximately 0.05% after annealing one hour at 1500 C. Samples were annealed to a maximum temperature of 2300 C, but only minor changes occurred above 1500 C.

The effect of annealing on these samples was significantly different from that previously reported. Previously it had been found that annealing caused an additional contraction in the transverse direction but a slight expansion in the parallel direction. No apparent reason for the difference has been discovered.

Coefficient-of-Thermal-Expansion Equipment. Current coefficient-of-thermal-expansion (CTE) measurements are time consuming because only one sample can be measured at a time and establishment of stable thermal conditions is slow. A new CTE apparatus has been built to measure three graphite specimens simultaneously from 25 to 800 C. An experiment was performed to compare the CTE's for several graphites between 25 and 425 C in the new and old apparatus. The results from both instruments agree within the standard deviations of the measurements; however, for five of six samples the average values from the old apparatus were slightly less than for the new equipment. One sample was the same.

Thermal Diffusivity Measurements at Low Temperatures. An apparatus was designed and fabricated to measure the thermal diffusivity of graphite samples from -195 C to 25 C by the flash method. A 0.4-inch diameter sample was mounted in an aluminum block which was attached to a liquid-nitrogen-cooled thimble. An evacuated glass envelope surrounded the thimble. The sample block was heated externally by a 150-watt projection lamp.

Thermal diffusivity measurements were completed on unirradiated types TSGBF and TSX graphite. The thermal conductivity of TSGBF graphite in the transverse direction increased from 0.077 cal/cm-sec-°C at -188 C to a maximum of 0.247 cal/cm-sec-°C at 25 C and then decreased to 0.175 cal/cm-sec-°C at 450 C. The change in thermal conductivity with temperature can be expressed by the following equation where T is the temperature in degrees Kelvin

$$K = -0.0917 + 2.45 \times 10^{-3}T - 5.53 \times 10^{-6}T^2 + 3.73 \times 10^{-9}T^3.$$

The change in thermal conductivity of the TSX transverse sample was similar and increased from 0.13 cal/cm-sec-°C at -195 C to a maximum of 0.36 at -25 C and then decreased to 0.19 cal/cm-sec-°C at 504 C.

EGCR Graphite Irradiations. The series of long-term irradiations of EGCR graphite is progressing satisfactorily. The seventh capsule, H-3-7, was installed in the GETR on September 24.

7. Boronated-Graphite Studies

Boronated-Graphite Irradiations. Work continued on schedule during the month to determine heating rates of boronated-graphite samples. Two capsules of identical construction were used to irradiate graphite containing 6 wt% and 8 wt% boron, respectively, for the purpose of establishing quantitatively heating rates of the two

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materials. The capsules were similar to earlier models but were modified to give a more uniform temperature profile.

The capsules were designed to operate at sample temperatures of 650 ± 150 F. In actual test, individual sample temperatures were well within the above range and averaged about 650 F. The relative heat generation rates of 6 and 8 wt% boronated samples compare well with the calculated values reported last period. Those calculations indicated 5% less heating in the 6 wt% samples.

The design of the capsule for the long-term test is complete with the exception of the final sizing of the inner heat-transfer assembly.

8. Metallic Fuel Element Development

Dry Abrasive Cleaner Installation. A dry abrasive cleaning machine has been procured and installed. The machine is a modified Vacu-Blast Model 200P Dry Honer in which the exhaust has been fitted with a filter of the absolute type. The machine has a work cabinet 24" x 28" x 36" wide. This machine will be used primarily for the abrasive cleaning of toxic material, especially thorium and its alloys.

Thorium Fuel Element Development. Defect testing of Zircaloy-2 clad, Th - 2.5 wt% U - 1 wt% Zr core, fuel rod specimens in 300 C water is continuing. Test specimens 5.1 cm long (2 inches) are cut from the 1.3 cm (0.523 inch) diameter, 0.063 cm (0.025 inch) clad, coextruded rod stock. The ends are recessed chemically and fitted with Zircaloy-2 end caps that are electron beam welded to complete the closure. The specimens are tested in a windowed autoclave that permits visual observation and time-lapse photography during the test.

Two samples taken from near the lead end of the extrusion exhibited behavior different from that of material taken from the middle or rear of the extrusion. In previous tests a small blister formed around the defect shortly after the specimen reached temperature, and the blister continued to grow slowly reaching a size of approximately 1 x 1.5 x 0.2 cm high after 5.5 hours. However, both lead-end samples developed a longitudinal split in the clad initiating when the blister was about the size of a BB. The corrosion rate, as determined by gas collected from the effluent water, was approximately twice that of previous samples due to the increased exposure of core material associated with the splitting of the clad. The samples are being sectioned for closer examination and metallography. The lead-end specimens were in the as-extruded condition except for being autoclaved in steam at 400 C for 72 hours prior to defecting.

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

Nine additional alloys (tabulated below) are currently being double vacuum arc melted into 7.4 cm (2.9 in.) diameter x 22.8 cm (9 in.) long ingots weighing approximately 11.3 Kg(25 lbs). These will be machined into extrusion billets and coextruded to 1.33 cm (0.525 in.) diameter rods with 0.063 cm (0.025 in.) thick Zircaloy-2 cladding. The rod stock will be used for defect corrosion testing.

Alloy No.	% U	% Zr	Alloy No.	% U	% Zr
1	2.5	--	2	5.0	--
3	2.5	1	4	5.0	1
5	2.5	2	6	5.0	2
7	2.5	5	8	5.0	5
9	2.5	10	10	5.0	10

Electrode stock has been prepared from 3.4 cm (1.342 in.) diameter x 15.2 cm (6 in.) long thorium slugs. Drilled longitudinal holes are provided for the zirconium alloy addition along with longitudinal grooves in the side for the normal uranium addition.

To date, alloys Nos. 1, 2, 9 and 10 have been single arc melted using 25-26 volts and 2800 amperes. The 10.0% Zr addition to both the 2.5 and 5.0% uranium binary alloys has a considerable effect on the melting characteristics and the resulting ingot side wall quality. The 10% Zr alloys appear during melting to have a more diffuse arc with the surface of the molten pool only faintly visible. The resulting ingot side wall is darker and more porous than the binary Th - U alloy ingots.

Dispersed Phase Thorium Alloys. One phase of the metallic fuel development program is the study of quarternary additions to thorium base, solid solution strengthened, Th - U - Zr ternary alloys for development of finely dispersed intermetallic compounds in the structure. The dispersed phases are expected to contribute to additional irradiation swelling resistance in these high strength, corrosion resistant alloys.

Alloys were prepared by arc button melting with aluminum or beryllium additions (500 to 2000 ppm) to thorium base alloys containing 2.5 or 5.0 wt% U and 2.0 or 5.0 wt% Zr. Th - 2.5 wt% U - 1 wt% Zr alloy buttons prepared by either prior vacuum melting or argon melting only were hot rolled from 900 C preheat in neutral salt to 0.318 cm (0.125 inch) thickness without cracking. Attempts to hot roll the alloys of higher Zr or U content have not been successful.

Submicron Uranium Carbide Dispersion in Metallic Uranium. An additional quantity of uranium shot, -10 +30 mesh, containing from 500 to 650 ppm carbon, was received and examined. Conditions used to prepare this material resulted in slower quenching speeds. The uranium carbide was thus precipitated in a dendritic pattern in most cases and only the edges of the particles contained the carbide distribution resulting from precipitation from enforced solid solution. Billet assemblies have been prepared using this material and the finer shot, -10 +250 mesh, to determine the effect of shot size on densification during extrusion.

Compatibility of Structural Materials with the HTLTR Environment. The evaluation of cladding and structural materials for use in the High Temperature Lattice Test Reactor (HTLTR) was continued. The materials being evaluated are Nickel A, TD Nickel, Hastelloy B, molybdenum, Hastelloy X, Inconel 625, and Inconel 600. In the second test of this series, specimens of these materials were exposed for 200 hours to a nitrogen gas-graphite environment at 1200 C. The gas was passed through graphite at 1200 C, then over specimens in contact with graphite and finally over specimens held in metal racks not in contact with graphite. In this test all of the materials were found to have undergone some weight gain. The alloys which did not contain chromium, i.e., Nickel A, TD Nickel, Hastelloy B, and molybdenum had weight gains of 200 to 700 mg/dm^2. The highest weight gains were for samples in contact with graphite. Bend tests showed that the chromium-bearing alloys and the molybdenum, but not the Nickel A, TD Nickel or Hastelloy B, were embrittled in this test.

Additional mechanical tests and metallographic data will be obtained on these specimens in order to complete this preliminary evaluation. A 1000-hour test will be run in the simulated reactor atmosphere on selected alloys for a more rigorous evaluation.

9. Aluminum Corrosion and Alloy Development

C-1 Loop. The C-1 Loop was charged on September 26 with X-8001 alloy aluminum dummies having a geometry identical to that of the proposed fuel charge. In the period before reactor startup, the coolant flow (single-pass process water) to the loop gradually dropped to zero. (The reason for this loss of flow is not known; perhaps the flow control valve, which had been valved down to 4 gpm vibrated closed.) When the reactor was started up, the test section became overheated which resulted in accelerated corrosion of the aluminum and almost complete blockage of the flow channel. Cooling was not restored until a few hours after startup; at the present time, flow is sufficient to maintain adequate cooling despite the partial flow blockage. The inner tube of the test section will be pulled out and replaced at the next outage to remove the dummies. There is no reason to suspect that the in-reactor pressure tube was damaged. This incident will probably delay the C-1 test program at least one month.

At the last reactor outage, a new dump system was installed on the C-1 Loop. This system will completely condense all the loop steam if the dump valves are opened or a rupture disc should fail.

Two new items have to be installed in the loop as a result of preliminary hazard analysis. The first of these is an automatic valve to be placed in series with the present flow control valve. The new valve will be a backup for the flow control valve in case it doesn't close during the dump cycle. Plans are being made to put a hydraulic operator on the presently installed manual valve.

The second item is an inter-tie line between the backup accumulator and the loop accumulator. This should allow a smooth flow after the backup accumulator has been emptied and before the process water starts flowing through the tube. Plans are being made to install this system.

At the last outage, additional shielding was installed to decrease the neutron leakage to a safe level.

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

Thermal Hydraulic Studies. A partial analysis was made of data from experiments investigating the cooling of 19-rod bundles and a rough draft interim report was written. The partial analysis consisted of abstracting those data concerned with boiling burnout and subjecting them to abbreviated hand calculations. This was done to present reasonably firm conclusions concerning boiling burnout, the most important information of the experiments, at this time rather than waiting for the results of the more complete and detailed analysis.

New information from this analysis which has not been reported before is concerned with operation at heat fluxes higher than those which give the first indication of boiling burnout. The importance of this information lies in the fact that it gives a clue to the magnitude of the problem which may exist when the first indication of boiling burnout is exceeded. The analysis is predicated on data from several typical experimental runs. A discussion of how the experimental data were analyzed follows.

In general, thermocouple readings of average inside tube wall temperatures were of the order of 600-625 F just prior to the start of boiling burnout. Heat fluxes were increased until average inside wall temperatures rose to a maximum of about 750 to 850 F. The wall temperatures "wandered" in a random manner after the start of boiling burnout. Temperature wander was pronounced at low flow rates and low outlet quality conditions and ranged from about 10 F at some of the least pronounced conditions to over 100 F. In this analysis two simplifications were introduced: first, a time average was used for the rod wall temperature, though the random nature of the wander made accurate averaging difficult. Second, the temperatures used in the calculations were average circumferential temperatures. An 800 F reported temperature may have come from an average of 600 F around part of the circumference of the rod and a temperature in excess of 800 F over the remainder of the circumference.

The temperatures used in this analysis were inside wall temperatures. The outside surface temperatures would be less due to the temperature drop across the tube wall. At an average heat flux of 700,000 Btu/hr-sq ft, the temperature drop across the wall would be about 27 F and 38 F for the inner seven and outer 12 rods, respectively. For an average heat flux of 300,000, the corresponding temperature drops would be about 10 F and 15 F.

Based on these considerations, the data show that an increase in heat flux of about 70,000 Btu/hr-sq ft would cause an inside rod wall temperature increase from about 600 to 800 F. This appeared to be independent of flow rate and outlet quality. At conditions of 5 to 10% outlet quality, the start of boiling burnout occurred at heat fluxes around 700,000 Btu/hr-sq ft. Thus, an increase in heat flux of about 10% caused the rod inside wall temperature to increase by 200 F. At higher outlet qualities a larger percentage in heat flux was required to cause the same increase in surface temperature. At a mass flow rate of 500,000 lb/hr-sq ft and an outlet quality of about 50%, a 70,000 Btu/hr-sq ft increase in heat flux also caused the rod inside wall temperature to reach 800 F. This, however, was about 30% the boiling burnout heat flux at these conditions.

From this analysis it is concluded that the occurrence of boiling burnout at high qualities does not necessarily have drastic consequences; instead, a sharp but limited temperature increase may be experienced.

11. Advanced Reactor Concept Studies

Fast Supercritical Pressure Power Reactor. Burnup in the 300 Mwe FSPPR has been reevaluated for a shorter fuel exposure. The average fuel exposure is now about 80,000 MWD/T. Some corrections were made in the calculations, giving a breeding ratio of about 1.14 at this exposure.

Calculations are in progress to obtain fuel reaction rates for the 1000 Mwe unit to be used for economic comparisons with the 300 Mwe reactor. The only remaining calculations for the 300 Mwe design were those relative to neutron kinetics, which have now been completed. Results obtained yield a neutron lifetime of 8.7×10^{-7} sec. The effective delayed neutron fraction (and hence the dollar value) was computed to be 0.00414. Fast fissions in U-238 approximately double the effective delayed fraction associated with plutonium fissions alone.

Fuel cycle cost calculations were completed for the 300 Mwe FSPPR. Using the AEC Cost Evaluation Handbook Methods, with minor changes associated with the use of plutonium fuel instead of U-235 and assuming the new plutonium prices of $10/g for Pu-239 and Pu-241, the fuel cycle direct cost is 0.75 mill/kwhr, and the working capital cost for fuel fabrication is 0.25 mill/kwhr.

Preliminary layout of the FSPPR power plant was completed. Total volume of the proposed design is around 1,700,000 cu ft. The design calls for a standard all-steel building. A six-inch protective concrete wall separates power plant area from office and shop area. A cost estimate for this building has not yet been prepared.

Plutonium-Fueled Reactors. A review of Hanford studies on plutonium-fueled fast compact reactors and a description of spacecraft and rocket reactor concepts was given before a group of interested AEC-DRD personnel in Washington.

Military Compact Reactor - Plutonium Fueling Studies. At the request of the USAEC a brief study is being made of the potential size and weight reduction which might be achieved in the Military Compact Reactor (MCR) by using plutonium instead of U-235 fuel. Preliminary results of this study indicated that by substituting plutonium fuel for uranium in the MCR, without altering the basic reactor core design (except for size) for the same power and life (15 Mwt and 3600 Mwd), the reactor core could be reduced from an original 15 inches diameter to about 8.5 to 9.5 inches, depending on the fuel material selected. A comparison of the core designs considered is given in the following table.

Shielding calculations were performed to evaluate the percent reduction in weight of the fixed and removable shields for the plutonium-fueled Military Compact Reactor based on reducing core size to nine inches. Although the reference uranium-fueled MCR core physics calculations were based on a 5-inch BeO reflector, the reference shield design was based on a composite reflector of 2.2" BeO and 3.3" stainless steel. The reference shield is thus already somewhat "light" for the reference core, so that the weight comparisons made here should be conservative. For the plutonium-fueled cases, a reflector composed of 3 inches of BeO and 2 inches of tungsten was substituted for the previous reflector. Several assumptions were utilized in doing the shielding weight reduction calculations: (1) reduction in self-shielding by the core due to reduction of core diameter was assumed insignificant, (2) the new reflector configuration was assumed to attenuate neutrons as efficiently as the reference "shielding" reflector concept, and (3) the increase in density of the new reflector concept compared to the old was used to reduce the cross-sectional density of the fixed inner lead shield. Based on these assumptions, the radii of the shields were scaled down and weight savings were calculated based on reduction of shield region volumes. No reduction in axial shield dimension was made, although it appears that some additional savings could be made by doing so. The total fixed shield was found to be 33% lighter. The total removable shield was 13.75% lighter and the total shield weight was reduced by 8900 lb, or 21.2%.

Further studies have indicated that by using an advanced fuel structure such as a cermet-based "gridplate" fuel (a honeycomb structured fuel), core size could be further reduced to 7 to 8 inches diameter. Alternatively, a reactor the same size as the MCR could be approximately doubled in power without sacrifice of useful life, or at the same power could have a several-fold multiplication in endurance, using plutonium fuel. Such reactors would represent an advanced technology and are predicated upon the achievement of high burnup in the fuel.

To take full advantage of the potential reduction in core size made possible with plutonium fuel, it is undesirable to use large inventories of U-238 as a means of obtaining a negative Doppler coefficient as is apparently done in the MCR design. A prompt negative temperature coefficient could probably be achieved by the use of cermet fuel, or by special design arrangements in a ceramic fueled core such as providing a fueled-cermet cage structure for mounting ceramic elements in a split-half array. Such designs would require development efforts which are beyond the scope of further consideration at this time.

Reducing core size inevitably increases the magnitude of many design and operating problems, such as control span required to compensate for burnup, fuel burnup, fission gas pressure buildup, heat flux, and coolant pressure drop. The cores described in the following table are based upon what is considered to be reasonable development of the technology of pin-type fast reactor cores. In the smaller cores, the larger control span requirements appear to preclude the use of heavy metal reflectors with rotating drum controls, but composite reflectors such as used here appear feasible. Ceramic fuel designs would require reservoirs for fission gas retention and a special support design (for prompt negative temperature coefficient) as previously mentioned. Cermet fuel cores may offer more flexibility in design of the smaller plutonium fuel reactors, if burnup capabilities and reactivity coefficient characteristics prove to be promising.

Control Region in the PFSR. Initial two-dimensional calculations using the ANGIE Code indicate that the control span of the "spectral shift" control region is less than was initially estimated. The control strength was reduced by about 13% $\Delta k/k$ when a cylinder of finite length is compared with an infinite cylinder.

These are the first calculations of a series dealing with a variety of control region configurations necessary to determine the maximum control span available with a control region.

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MCR CORE COMPARISONS

		Reference	Pu Core I	Pu Core II	Pu Core III
Type		Ceramic	70/30 Cermet	Ceramic	70/30 Cermet
Fuel		UO_2	PuO_2 or Pu_2O_3/Nb	PuO_2	PuO or PuN/Nb
Enrichment		47% U-235 53% U-238	95% Pu-239 5% Pu-240	95% Pu-239 5% Pu-240	95% Pu-239 5% Pu-240
Core size (Cylinder dia.)		15"	9.5"	9"	8.5"
Reflector		5" BeO	3" BeO 2" W	3" BeO 2" W	3" BeO 2" W
Core Composition vol%	Fuel	61	53	43	48
	Coolant	22	30	40	35
	Structure	17	17	17	17
No. of Pins		2520	870	3113	674
Pen Diameter, in.		0.264	0.264	0.125	0.264
Heat Flux B/hr-ft^2		230,000	1,000,000	660,000	1,500,000
Burnup, Mwd/T		13,000	99,000	98,000	117,000

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SNAP-8 Irradiations in C-Reactor Shield Test Facility. Calculations were performed with MAC to evaluate the C-Reactor shield experiment well for use as an irradiation facility for SNAP-8 instrument package tests. In the proposed irradiations, it is desired to achieve a dose of 10^{13} nvt and 10^8 roentgens of gamma in 100 hours. Neutron and gamma distributions were calculated for three cases: the reactor shield, the shield well filled with water, and the well filled with lithium hydride. It appears possible to achieve the desired doses in a longer irradiation time with water, or to closely approximate them in the desired irradiation time using a judicious combination of materials. A list of equipment requirements and a cost estimate are being prepared jointly with IPD Irradiation Testing.

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.

Dislocation channeling in deformed irradiated molybdenum has been discussed in previous reports. A total of five electron micrographs and accompanying diffraction patterns have now been analyzed, representing two as-irradiated foils and three foils annealed at 750 C (1382 F). Two carbon levels, < 10 ppm and 150 ppm carbon, were represented among the five foil specimens. The electron diffraction patterns were not complete, usually consisting of only a few spots. In each case one or more orders of (110) reflections were observed, with at least one reflection from a different plane. These reflections sufficed to determine the foil orientation.

In every case the direction of the channels observed in the electron micrographs coincided with a trace of a $\{112\}$ or $\{110\}$ plane, both of which are commonly reported as slip planes in molybdenum. These observations are in agreement with the hypothesis that the channels result from the movement of dislocations through the matrix by a simple glide mechanism. Insofar as can be determined, the amount of carbon present and the extent of annealing have no effect on the channeling process. Molybdenum microtensile specimens, suitable for straining in the electron microscope, have been irradiated to 1×10^{19} nvt (E > 1 Mev). These specimens will be strained while under observation in an effort to observe the nature of the dislocation reaction leading to the formation of the channels.

A detailed study of radiation-induced defects in molybdenum is in progress. Foils containing < 10 ppm and 150 ppm carbon have been irradiated to 1×10^{18}, 3×10^{18}, and 6×10^{18} nvt (E > 1 Mev). These specimens will be studied by transmission electron microscopy in the as-irradiated state to determine the variation in the density and nature of defects with fast neutron exposure. Annealing studies similar to those reported previously will also be carried out.

Additional x-ray diffraction micrographs have been obtained from molybdenum single crystals. Attention has been turned to the study of deformed crystals. Irradiated and unirradiated crystals deformed to fracture in tension have been sectioned approximately parallel to (110) and (100) planes. Diffraction micrographs obtained to date with crystals sectioned parallel to a (110) plane have been of two types: (1) with irradiated crystals, a pattern of parallel slip traces at approximately 45 degrees to the tensile axis, and (2) in both irradiated and unirradiated crystals, a pattern of traces intersecting in a complex manner. The observation of both micrographs types in the case of irradiated crystals may be a consequence of variation in location of the scanned area of the specimen with respect to the fracture surface.

Investigation of the strain-rate dependence of the 0.2% offset yield stress in annealed polycrystalline molybdenum at intermediate strain rates has verified the existence of four distinct regions in the curve of room temperature yield stress versus log strain rate. The range of strain rates investigated is from 8.33×10^{-6} sec^{-1} to 8.33×10^{-3} sec^{-1}. The strain-hardening coefficient over this range of strain rates and at room temperature is given by the relationship

$$n = -0.02 \ln \dot{\epsilon} + 0.2$$

where n is the strain-hardening coefficient and $\dot{\epsilon}$ is the strain rate in units of min^{-1}.

Results of strain-rate sensitivity experiments conducted at room temperature are still being evaluated. Of primary concern is the observation of a parabolic relationship between $\Delta\sigma$ and σ. This may arise from cycling between two regions which have a different stress-strain rate dependence. Further experiments are planned, in particular, a series of strain-rate factor change tests.

Design and fabrication of a prototypical capsule for irradiation experiments at elevated temperatures is continuing; final assembly has been delayed pending arrival of thermocouples and heaters. Dummy capsules are being prepared and will be charged into a reactor for the purpose of determining the magnitude of nuclear heating which will be encountered and whether the heat transfer properties of the capsule are sufficient to maintain a minimum temperature of 150 C in the nuclear environment. These tests will also indicate the maximum temperature which can be reached in a capsule designed for operation below 150 C with no power input to the heaters. It may be necessary to utilize one capsule design for irradiation between 150 C and 500 C (302 F and 932 F), and a second design for temperatures above 500 C.

2. Plutonium Physical Metallurgy

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

The steady state creep characteristics of the alpha and beta phases of plutonium were determined as a function of temperature in the range of 100 to 160 C (212 to 320 F) at compressive stresses of 150 to 3100 psi. The creep strain of the alpha phase at 120 C (248 F) and a stress of 3100 psi was found to be less than 0.001% per hour, while under a similar stress the steady state creep rate of the beta phase decreased from 1.2% per hour at 157 C (314.6 F) to 0.054% at 120 C (248 F). The activation energy for creep of the beta phase at this stress was computed to be 30,000 calories per mole. A similar activation energy was calculated at a stress of 2150 psi.

The deformation during the alpha to beta and beta to alpha transformations at a transformation rate of approximately 1% per minute was determined using 0.38-inch diameter specimens under compressive stresses of 100 to 1600 psi. This permitted the use of longer specimens (0.75 inch) resulting in greater accuracy in determining the strain data than was possible with the previously used 0.25-inch diameter specimens. The results, however, were very similar to those obtained from the smaller specimens. The strain during both transformations increased from zero at 100 psi to 2.5% at 1600 psi at an average transformation rate of one to two percent per minute. The transformation strain was very nearly a linear function of applied stress.

One of the problems encountered in the phase transformation studies has been the presence of microcracks which form during the beta to alpha transformation while casting. The volume of these microcracks can be decreased by hydrostatic pressing in the beta phase and cooling under pressure. Metal prepared in this manner has obvious disadvantages as starting metal. Rapid cooling from the beta phase during casting provides another, and more convenient, method of reducing the volume of microcracks.

A casting consisting of four rods, 0.38-inch diameter and 6 inches long, was cooled from the melt to the beta phase at approximately 10 C (50 F) per minute. Two of the rods were cooled through the beta to alpha transformation at approximately one degree Centigrade per minute and the other two were quenched to room temperature in a Freon bath. Twenty-four hours after casting the slowly cooled rods exhibited a density of 19.5 g/cm^3. No density change was observed after ten thermal cycles between -80 C (-112 F) and +100 C (212 F). The density of a specimen from the quenched rod was 19.58 g/cm^3 twenty-four hours after casting. This value increased to 19.68 g/cm^3 after a similar cycling schedule. The density increase can be attributed to the transformation of retained beta. The lower density of the slow cooled metal is explainable on the basis of the presence of a greater volume of microcracks. This was confirmed by metallography, which indicated that the quenched metal had fewer and smaller microcracks.

It has been established that the transformation of alpha-rolled plutonium significantly alters the texture induced by the original working. A secondary texture, however, results which yields a diffraction pattern significantly different from that typical of a completely random orientation. This phenomenon provides a sensitive method of determining whether or not transformation has occurred during an annealing treatment at, or near, the transformation temperature. Recent data from Argonne suggest that alpha plutonium recrystallizes at about 120 C (248 F). Since this temperature is above the equilibrium transformation temperature, there is some question as to whether or not the alpha to beta transformation contributed to the apparent recrystallization. Work is under way to clarify this point if possible. Material of comparable purity will not be available, however, until the LASL electro-refined plutonium is received. It is expected that the recrystallization temperature is quite sensitive to the impurity content and should be directly related to the purity level.

Work is continuing on the identification of the crystallographic orientation produced in alpha plutonium by rolling. The primary effort at present is directed toward establishing the consistency of the results obtained from various specimens which have been similarly treated. To date it has not been possible to satisfactorily verify the anisotropic changes in lattice parameter which had been indicated in the earlier work. This may be a function of the length of the time interval between the actual working and the diffraction evaluation. The detailed derivation of the rolling texture is awaiting the availability of a pole figure goniometer. This equipment is scheduled for delivery next month.

Decontamination of cellulose acetate replicas of plutonium for electron metallography involves ultrasonic cleaning of the replicas in HCl. The fact that most acids, including HCl, are known to attack cellulose acetate has discouraged some investigators from utilizing it in replicating radioactive materials. However, it has been determined that by using HCl concentrations of 0.5N or less, no detectable artifacts are introduced, even after a seven-hour treatment in the acid. Ultrasonic cleaning for 15 minutes in alternate baths of 0.5N HCl and water (two cycles) will generally reduce the smearable contamination on the acetate to less than 5000 D/M.

Metallography of plutonium is complicated, not only by the extremely high toxicity of the metal, but also by its high oxidation rate. After preparation, only a few hours are available for examination of metallographic specimens before oxidation is noticeable. A metallographic technique which may prove successful in both eliminating the toxicity problem and reducing the oxidation rate, and yet not sacrifice image quality, is to coat the sample with a thin plastic film. Preliminary observations of a specimen of etched Zircaloy-2 which was coated with a $< 0.5\mu$ thick layer of Formvar indicate no loss of contrast or resolution due to the Formvar film, even at magnifications greater than 1000X. It is not possible to visually distinguish in the microscope between coated and uncoated areas of a metallographic specimen. The Formvar film will next be evaluated for its effectiveness in (1) retarding plutonium oxidation, and (2) containing the contamination.

E. CUSTOMER WORK

1. Radiometallurgy Laboratory

Examinations and Measurements. Routine examinations and measurements are or will be reported as part of the sponsoring research and development programs.

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

a. Metallography

Samples Processed	71
Photomosaics	8
Autoradiographs	21

b. Chemistry

Burnup Dissolutions	31
Decladding Dissolutions	5
Fission and other Gas Collections	20

c. Physical & Mechanical Testing

Tensile Tests (Room Temperature)	60
Hardness Tests	234
Density Measurements	17

d. General

Negatives Processed	741

High Temperature Tensile Testing Machine. Modification of the high temperature tensile testing machine furnace is scheduled for completion by the end of the month. Final testing of the completely assembled testing equipment at the vendor's plant is scheduled for November.

"E" Cell Metallographic Facilities. Redesign of the cathodic etcher for installation in "E" cell is about 75% complete.

Pinhole Autoradiographic Camera. The resolution of the pinhole camera was tested by making gamma autoradiographs of a special target sample. The target sample was prepared by making a grooved grid on the surface of a steel disk and filling the grooves with a mixture of irradiated UO_2 and epoxy resin. These radiographs showed good resolution of the sample grid.

Scintillation Scanning Equipment. Testing of gamma scanning equipment continued by scanning wafers sectioned from a Pu-Al alloy PRTR cluster rod. Test results show good reproducibility of fission product distribution curves and ability to differentiate minor differences in radial fission product distribution.

Fission Gas Collection. The first tritium bearing gases were successfully collected and quantities measured in the fission gas collection system. Studies are being made to determine whether or not significant amounts of tritium were lost through isotopic separation from samples collected in the metal collecting system.

High Temperature Extensometer. Initial tests conducted on an electromechanical transducer or distance detector at a temperature of 300 C show that a temperature variation of ± 5 C will cause a corresponding measuring error of ± 0.0002 inch. Reproducible measurements within ± 0.0001 inch can be made with this device at room temperature.

Surface Replication of Al-Li Alloy. A procedure for surface replication of Al-Li alloy samples which utilizes vacuum cathodic etching was established this month. The surface produced is suitable for both second phase and gas void detection.

2. Metallography Laboratories

During the report month 332 samples were processed, a total of 464 macrographs and micrographs taken, 1828 negatives printed and 6186 prints processed.

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

A two-week laboratory training class was held for two technicians from Physical Metallurgy for instruction in general metallographic techniques. The material included laboratory techniques and sufficient theoretical background to permit use of metallographic equipment. This included study and practical experience with abrasives, polishing powders, acids and mixtures of acids, microscopes, metallographs, photographic films and procedures, and photomicrography.

3. N-Reactor Design Testing

N-Reactor Magazines. An N-Reactor charging magazine was successfully modified to prevent liner shrinkage by removing the tapered end, milling slots in the periphery of the carbon steel tube and molding the polyethylene liner around the protrusions with heat and pressure. This method of liner restraint was tested with no adverse results in a series of tests, including a fuel charge with 700-pound back force, a fuel charge with 1900-pound back force, ten charges against a water column of 700-pound back force, and ten charges without back force. As a result of these tests, 100 magazines will be modified in like manner to the one tested.

4. High Temperature Lattice Test Reactor Prototype. Detailed layout of the core insulation and structural design of the shell are 85% and 30% complete, respectively. Machining of graphite for the core mockup is in progress.

Fabrication of the safety rod drop and braking mechanism has been completed, including successful preliminary tests. A 42-pound weight which was used in the tests was stopped in 18 inches after a free fall of $7\frac{1}{2}$ feet.

5. EBWR Fuel Elements

EBWR Plutonium Fuel Element Fabrication. Construction of the new vibrational compaction facility in the Plutonium Fabrication Pilot Plant was completed. Except for several minor alterations and/or improvements, the facility is now ready for use in fabricating plutonium-bearing fuel elements.

The two 1500-pound force vibrators and their respective consoles were checked by a factory representative. All necessary calibration was performed and standard operating procedures were established.

Air gauging equipment is being set up for quality control, dimensional inspection of either EBWR or PRTR size tubing.

Fused depleted uranium dioxide is being ground, screened and roasted for EBWR fuel; 250 kg have been finished and 200 kg are in process. (1600 kg will be needed.)

Cladding Procurement. The last two groups of EBWR cladding tubes to be received on-site have minute longitudinal scratches on some inner surfaces of the tubes. Associated with these scratches are tiny burrs which are reported as defect indications by the ultrasonic testing equipment. Thus, adequate testing of these tubes now requires reworking to remove the burrs.

An order has been placed for the fabrication of a portion of the tubes needed for Mark II nested tubular PRTR fuel elements. The remainder of the tubing will be ordered after receipt of revised bids.

Burnable Poison Fuel in UO_2. Photomicrographs of 2 wt% and 10 wt% Sm_2O_3 in UO_2 revealed uniform distribution of the Sm_2O_3.

6. Other Off-site Customer Work

Fuel Fabrication. A 1/8-inch hexagonal honeycomb grid was fabricated directly by pneumatic impaction of an 80 vol% tungsten-UO_2 mixture. The particle sizes of the tungsten and UO_2 were 5 microns and 44 to 62 microns, respectively. One-eighth-inch mild steel hexagonal bars were used as mandrels in the 37-hole grid. They were removed chemically after impaction.

Thoria Test Elements. Thorium oxide was pneumatically impacted, crushed, sized and Vipacked into aluminum cans for reactor testing purposes.

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PHYSICS AND INSTRUMENTS LABORATORY

MONTHLY REPORT

OCTOBER 1963

FISSIONABLE MATERIALS - 02 PROGRAM

REACTOR

K-Lattice PCTR Experiments

Absolute k_∞ values have been determined for KVNS fuel pieces in Zr-2 process tubing in the K-lattice. This work has included both wet and dry cases. An informal report, HW-79325, "K-Lattice Parameters from PCTR Measurements," is almost complete.

N-Reactor Lattice Parameter and Spectral Measurement Tests at Startup

Procurement and fabrication of materials are under way for startup tests to measure the lattice parameters ϵ, p, and f, the neutron temperature, and the r-value of the epithermal flux in the lattice. Similar tests are planned for both cold and hot core tests.

The fuel elements required for cold tests are nearly complete. Pins and foils of many materials are being fabricated for placement in these cold elements. Fuel elements required for hot tests are being machined, and similar pins will be used. At this time there appears to be no major problems.

Scattering Law for Water

A report describing the improvement in scattering law calculations by treating exactly two resonances of the spectral function rather than one resonance has been written for the "Physics Research Quarterly Report - July, August, September, 1963."

Modified Heavy Gas Model

Studies of graphite-moderated uranium and plutonium systems with the Horowitz modified gas model have been compared with studies of such systems with the more exact neutron balance equation. The modified gas equation produces reasonable results over large ranges of uranium concentration and moderate ranges of plutonium concentrations. A report of this work has been written for the "Physics Research Quarterly Report - July, August, September, 1963."

Instrumentation

Instrumentation development, procurement and fabrication are being accelerated in preparation of N Reactor startup and physics tests. Thimbles for water-cooling several of the detectors, gas seals, and clamps are being fabricated. Measurements have been carried out in the 305 Reactor which confirm neutron flux-streaming estimates at perforations in the N Reactor biological shield. A majority of the developed items has now been completed. Progress was made on a special detector system to be used to monitor filtered samples of effluent water for traces of uranium during the physics tests. Progress on both the electronic and mechanical portions of the complete physics test instrumentation system is moving ahead on schedule.

Operational testing was continued at KE Reactor on one N Reactor gamma energy spectrometer of the 24 to be used in the N Reactor fuel rupture monitoring system. Data obtained in cooperation with Research and Engineering, NRD, following water sampling system changes, showed a definite increase in N-16 activity; however, multichannel analyzer results did not provide any definite photopeaks, even after considerable lead shielding was added around the detectors to reduce the background. At present, the spectrometer is being used to monitor one test loop for possible ruptures in some target fuel elements. Three production spectrometers were received from General Electric (APED) and final circuit drawings were also received. Work is being accelerated to complete the necessary test procedures (DT-1133) to provide information for general testing of the production units.

Installation and fabrication work progressed satisfactorily on the new reactor fuels testing loop being installed at PRTR. General rack cabinet installation is nearly complete, and both regulated and unregulated power lines are installed. Flow control equipment has been mounted and connected. Electronic instrumentation was received for both the dual channel gamma spectrometer and the delayed neutron system. Tests and necessary modifications were carried out on both systems. Major installation of the electronic instrumentation was started. Fabrication of special preamplifiers was completed and both were tested with the gamma spectrometer to determine operational capabilities.

Investigations were started to determine appropriate materials for use in fresh fuel storage casks, which are being developed to fill a need for economic storage of enriched fuel in nuclear-safe containers. Two sample casks have been fabricated to previously established specifications. A suitable neutron counting system is being assembled at present. Initial materials to be employed in the experimental tests will be water and cadmium.

System Studies

The N Reactor primary coolant system simulation was checked out and preliminary runs completed. These tests establish the approximate controller settings required for good stability and performance, and the relative sensitivity of these settings. It was found that best over-all system performance is not obtained by optimizing the individual loop flow controller settings, as might be expected. Data defining controller settings for use in forthcoming fields tests were obtained.

Maximum primary coolant flow fluctuations accompanying loss of one and two primary pumps were also determined as were coolant temperature transients. These runs were made at initial flow rates corresponding to both the fifty percent and one-hundred percent of design flow cases.

The pump drop-out simulation studies have disclosed that an error is introduced in the computations as the check valve closes and reduces coolant flow through the disabled pump to zero. Although these errors are so small as to have a negligible effect on data obtained to date, a more accurate simulation model has been devised which eliminates this deficiency.

A MIDAS program for the primary coolant simulation was prepared for use on the IBM 7090 computer. It was used to check the accuracy of the analog model under normal operating conditions. Limited results compared favorably.

The secondary loop simulation work done by Bailey Meter Company is being used in setting up separate steam generator and secondary loop simulations for use on the HAPO computers. This is resulting in a considerable savings in time because the necessary scaling and circuitry is, in many cases, already worked out. Changes are necessary, however, due to differences in the available equipment, the data desired, and the advisability of using a more versatile model. In addition, certain areas of the Bailey report on this work are not well explained. These areas are being reworked to assure accuracy and understanding.

The steam generator simulation model is essentially ready for debugging. The model is patched on a problem board and calculation of the initial computer settings is nearly complete.

The previous difficulty in simulating two of the steam generator models with the MIDAS system was overcome. The MIDAS systems for these steam generator models (they are different from those currently used for the analog simulation) have been completely debugged for the MIDAS system. The non-linear model requires 50 seconds of 7090 time for simulation with the MIDAS system.

The simulation model for the balance of the secondary loop is approximately eighty percent complete. It consists of one dump condenser model, one surge tank model, the steam header, one condensate pump model and associated control circuits.

A request for additional power excursion data on N Reactor was received from N Reactor Department during the month. The startup kinetics model developed earlier this year by Systems Research Operation was programmed on the analog computer and was used to determine maximum power level and reactor water and fuel temperatures resulting from relatively slow reactivity ramp inputs. The ramp rates ranged from 0.5 to 10 cmk/seconds. Heat transfer coefficients 0.5, 1.0, and 1.5 times the most probable value were used to show the relative sensitivity of the results to errors in the calculated coefficient.

Three Bailey control systems were used in the simulation runs made during the month on the N Reactor primary coolant system. During the calibration and early debugging stage, preparatory to making these runs, it was found that the controller time constants did not correspond to the calibration dials, and, further, that the settings appeared to vary with zero adjustment of the controller operational amplifiers. The trouble was traced to an error in the wiring of the fixed signal limiters used with the controllers. The required wiring changes have been completed.

The first portion of a three part Optimum Control problem was completed which solves a set of two differential equations simultaneously to establish optimum values of two variable parameters. Solutions to parts two and three are now proceeding.

SEPARATIONS

Critical Experiments with PuO_2-Plastic Mixtures

Critical mass experiments were continued with PuO_2-polystyrene compacts using the Remote Split-Table Machine. Seven critical core configurations, consisting of rectangular prisms, or slabs, were assembled during the month. The experiments provided information on the reflector savings of the Lucite reflectors, and on the effect of core shape on criticality. The plutonium concentration in the compacts was 1.14 g/cc (2.2% Pu^{240}), with an H/Pu atomic ratio of 15. The critical assemblies were comprised as follows:

1. A rectangular parallelopiped of cross sectional dimensions 6-in. x 7-in.

   The critical length of the parallelopiped when reflected with 6-in. thick Lucite on all surfaces was 23.51 inches; the critical mass of Pu was 19.94 kg. The critical length for the case in which the ends were unreflected was known from a previous measurement to be 26.89 inches. The difference in core length with and without these reflectors gives a value for the reflector savings of the Lucite at the ends; the savings at each end is 4.29 cm.

2. A slab of 4-in. thickness and 24-in. width.

   With 6-in. thick Lucite reflectors on the top and bottom surfaces, the critical length was 16.35 inches (critical mass 29.33 kg Pu). When the sides of the slab were also reflected with Lucite, the critical length was reduced to 15.38 inches and the critical mass to 27.58 kg Pu.

3. A slab of 4-in. thickness and 8-in. width.

   The critical length with Lucite reflectors on all surfaces except the ends was 19.64 inches (critical mass 23.49 kg Pu). On reflecting the ends of the slab, the critical length was reduced to 16.08 inches and the critical mass to 19.22 kg Pu. The reflector savings of the Lucite at each end surface is, therefore, 4.53 cm.

4. A rectangular prism with 12-in. x 12-in. base dimensions.

   With 6-in. thick Lucite reflectors on the top and bottom surfaces, the critical height was 9.34 inches and the critical mass was 25.13 kg Pu. A similar non-reflected rectangular prism was previously found to be critical at a height of 12.54 inches. Thus, the reflector savings at each of the reflected surfaces is 4.06 cm.

5. A rectangular prism 8-in. thick and 12-in. wide reflected with Lucite on the top, bottom, and ends.

   The critical length was measured as 9.85 inches with the critical mass being 17.66 kg Pu.

In reflected assemblies, it is necessary to remove a portion of the reflector to permit insertion of the fuel bearing control and safety rods. Also, the fuel material making up the rods is encased in a steel jacket and inserted in a steel sleeve within the core. The combined effect of the control and safety rod sleeves, etc., and the effect due to the absence of reflector behind the channels was determined to be about 11% in mass for the case of a fully reflected prism with 9-in. x 9-in. base dimensions. Thus, in a reflected "clean" system, the critical mass would be about 11% smaller. Most of the effect is attributed to the removal of reflector in the placement of the control and safety rods.

Pulsed Neutron Source Experiments

One of the purposes of the pulsed neutron source experiments is to examine the feasibility of using this technique for k_{eff} measurements on in-plant equipment. Potentially the method may have wide applicability in assessing the degree of subcriticality of process vessels, and of storage arrays of fissile materials in the plant.

On October 7, the first attempt was made to perform such measurements in the field. The pulsatron (pulsed neutron source) and time analyzer equipment were mounted in the back of a sedan delivery, and with the use of a portable motor generator set for power, several experiments were performed.

The Z-9 underground waste crib, which is believed to contain a significant amount of Pu, was pulsed. The neutron detector and pulse tube were lowered to the earth floor of the crib through holes cut in the concrete roof at three different locations. The prompt neutron decay rate was measured, following the injection of neutrons into the system, at each of these locations.

The pulse generator equipment was then transported to Army ammunition storage igloo T-103 (now used for storage of plutonium scrap). The pulser was placed in several locations within the igloo and neutron decay rates measured. The data from these measurements are currently being analyzed.

The major electronic problem encountered in setting up these experiments was the loss of ion source pumping in the neutron accelerator head. This was due to voltage drop in the line from the control unit to the accelerator head. This distance had to be at least 30 feet and preferably 100 feet--a restriction imposed by the remote area. A cable about 50 feet in length with twice the diameter as the original worked satisfactorily for these experiments.

Analysis of Plutonium Solution Experiments and Minimum Critical Mass of Plutonium

A study of the critical data obtained from solution experiments in the Plutonium Critical Mass Laboratory and those reported by F. Kreusi, et al, has resulted in a slightly different value for the minimum critical mass of Pu^{239} in water. The new value is 520 g Pu as opposed to 509 g. The difference appears to be due to the methods used in correcting the critical data for the effect of the nitrate in the plutonium solutions. Although the minimum critical mass is only slightly different, the conditions under which it is obtained are more significantly changed. The results are compared below.

	New Evaluation	Previous Evaluation
Minimum Critical Mass in Water Reflected Sphere	520 g	509 g
Pu Concentration	~ 28 g/ℓ	~ 33 g/ℓ
Critical Volume	~ 18.6 ℓ	~ 15.4 ℓ
Sphere Diameter	~ 12.9 inches	~ 12 inches

The current results were corrected for the effect of the thin stainless steel spheres on criticality in the experiments, whereas the previous values are appropriate to a thin stainless steel sphere. The principal difference is in the concentration and volume at which the minimum critical mass is obtained; this difference is attributed chiefly to the corrections applied for nitrate, and not to the effects of the stainless steel shell.

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

The effect Pu^{240} has on the criticality of Pu^{239} depends on the H/Pu ratio, or degree of moderation. For dilute Pu solutions (for example, 30 g Pu/ℓ) the effect is such that for each percent Pu^{240} added, the critical concentration of Pu^{239} will be increased by about three percent. Calculations indicate Pu^{240} to have its greatest effect on critical mass for H/Pu ratios in the range of 20-60, where the neutron energy spectrum is faster, being intermediate between that of either a predominately thermal or fast system. In an unmoderated or fast system, the effect becomes small; it has been reported by G. A. Graves and H. C. Paxton that Pu^{240} present in Pu^{239} metal to less than ~ 10 percent increases the critical mass slightly.

Since larger quantities of high exposure plutonium will be coming available, it is interesting to speculate on the criticality of Pu^{240} by itself in a fast system. Pu^{240} is similar to U^{238} in that both nuclei contain even numbers of protons and neutrons in their nucleus. The threshold energy for fission in U^{238} is about 1.2-1.4 Mev. A large quantity of U^{238}

will not chain react by itself because of the energy losses from inelastic scattering of fission neutrons in the uranium, but Pu^{240} has a higher charge (Z) number and is more fissionable with a smaller threshold energy.

A series of Monte Carlo calculations have been made for systems containing only one of the isotopes U^{235}, U^{238}, Pu^{239}, or Pu^{240}. The calculations give the value for the reproduction factor in an infinite system of the pure metal with no moderator present. The data are presented in the table below:

MONTE CARLO CALCULATIONS OF REPRODUCTION FACTOR IN INFINITE SYSTEM

Isotope Present	Neutron Histories	Fissions	Absorptions	k_∞ of Metal
U^{235}	6000	5477.6	5988.8	2.382 ± .004
U^{238}	4000	424.49	4058.1	0.298 ± .008
Pu^{239}	6000	5465.4	5961.7	2.785 ± .006
Pu^{240}	6000	4702.9	6080.4	2.561 ± .038

It is interesting to note that k_∞ for an unmoderated Pu^{240} system is larger than k_∞ for a similar unmoderated system of U^{235}. This being the case, these calculations indicate the critical mass for a bare Pu^{240} sphere would be less than that for U^{235} metal. However, it is important to note that these results are very sensitive to the cross sections used in the energy range between 0.05 MeV and 5.0 MeV, where most of the fissions and absorptions take place in the fast system.

Truncated Sphere Buckling

The two-dimensional quadratic fit approximation for the diffusion equation has been coded and debugged. Calculated fluxes are very nearly consistent with expected values for almost all points. There are one or two points, however, where irregular values occur. Since these points also have a large coefficient coupling them to the unique boundary point at the junction of the curved and straight boundaries, it is plausible that the treatment of the contribution of this unique point is incorrect even though the values

Instrumentation

Rod drop tests to determine the worth of the rods were continued on the split-table assembly. A simultaneous comparison of a fission counter and a scintillation detector was made. The fission counter showed a larger drop in reactivity or a greater rod worth for the safety rod.

Requisitions were written for the purchase of an additional television camera for the close viewing of the split-table assembly, and for an X-Y recorder for use in plotting the pulsed neutron data.

Consulting Services on Nuclear Safety--Criticality Hazards

1. Nuclear Safety in HL

The "J" series of specifications for the Plutonium Metallurgy Operation was revised as follows:

J-1 - Plutonium Handling in Dry Glove Boxes

CCl_4 and Freon were added as permissible liquids.

J-3 - Plutonium Transporting and Storage

Rules to cover metallographic samples were added.

J-4 - Rules for Metallographic Samples

The size of the sample block was reduced to 8 ml., which permits the allowable number in a unit area to be increased to 180.

J-6 - Processing Thin Walled Castings (Formerly J-7)

The unit mass was increased from 1.0 kg up to 2.8 kg.

A new specification (K-6) was issued for the Specialty Shops. This specification covers the special handling and storing of 1.25 w/o U^{235} enriched uranium billets.

2. Nuclear Safety in NRD

A study is being made to determine safe limits for criticality control in handling enriched fuels that may be used in NPR. These limits will be used in regard to process evaluation studies in the 333 Building and 105-N Building, and will also be used to establish safe processing limits for the fuel. To date, safe nuclear parameters have been derived for the following:

1.25 w/o U^{235} Ingots and Billets
1.6 w/o U^{235} Ingots and Billets
1.25 w/o U^{235} Outer NPR Tubes
0.95 w/o Inner - 1.25 w/o Outer NPR Fuel Elements

This study will continue until a nuclear safety review has been completed of the 333 Building for processing the higher enrichment and formal nuclear safety specifications have been issued.

3. Nuclear Safety of Offsite Shipments

An addendum was made to the Redwood Car specification to permit the shipment of two UO_2-PuO_2 irradiation test fuel elements on the car. The shipment of 44 kg of 0.95 w/o U^{235} enriched uranium on the Redwood Car was also approved.

NEUTRON CROSS SECTION PROGRAM

Scattering-Law Measurements for H_2O at 95°C

Only a minor amount of data was obtainable on the triple-axis spectrometer during the month because of continued reactor outages. A few measurements were made to attempt to observe multiple-scattering effects in the scattering of neutrons from 95°C water. Measurements made with a water sample of one-half the thickness previously used showed differences in scattering which were about equal to the accuracy of the data. The source of previously discovered systematic errors in the scattering from the calibration standard of vanadium was determined to be due to the inaccurate positioning of the vanadium sample. Some measurements were made to determine the magnitude of any remaining inconsistencies in the scattering from vanadium but these measurements are incomplete.

Multiple-Scattering Correction for Slow-Neutron Scattering

The problem of calculating multiple-scattering effects in slow-neutron scattering-law measurements has been studied in some detail. Three methods of calculation have been considered: 1) a combined analytical-numerical scheme, based on the integral formulation of transport theory, which gives accurately and rapidly only the double-scattering effects; 2) a strictly numerical solution of the integral equation, giving all orders of scattering with reasonable accuracy and speed; and 3) a Monte Carlo calculation of all orders with reasonable accuracy and speed. Unfortunately the limited size of fast memory (32 K) restricts the second method to cases having the incident beam perpendicular to the sample slab. However, at the cost of greater running time, the Monte Carlo method can handle any angle of incidence. Since experiments at various angles of incidence are planned, the Monte Carlo method has been adopted, and schemes to minimize running time are being developed.

Time-of-Flight Spectroscopy for Slow Neutrons

Work continued on the development of equipment for a rotating-crystal spectrometer for the measurement of slow-neutron inelastic scattering by time-of-flight. The beam-shutter step plug for this facility was laboratory tested under expected operating temperature conditions. The step plug has now been installed in the 4-B hole of 105 KE reactor adjacent to the triple-axis spectrometer facility. Considerable modification of the shielding of the triple-axis spectrometer will be required before the new beam facility can be utilized. A study of the shielding of slow-neutron detectors on the experimental levels of the 105 DR and 105 KE reactors was completed. This study gives the information necessary to optimize the shielding of detectors for the rotating-crystal spectrometer. Reflectivity measurements were made on three commercial copper single crystals obtained for test. One crystal was chosen which shows some promise as a neutron monochromator. The indicated reflectivity of this crystal in the reflection mode was greater than the Al crystal sphere which is a candidate for the monochromator for the rotating-crystal spectrometer. The apparent mosaic of theCu crystal was less than 0.25 degrees.

Fast-Neutron Cross Sections

A review was made of the status of the data which have been obtained in the program of total cross section measurements for neutrons with energies of 2.5 to 15 MeV. As a result of this review the measured cross-sections of 38 elements were compiled and sent off-site for inclusion in cross-section compilation reports now in progress at BNL and URCL and for the use of physicists working on theoretical models of nuclear reactions. Samples of separated isotopes of Cr^{53} and Ca^{42} were received during the month and samples of Ru, Rh, Os, Ir, Pt, Cr^{53}, W^{182}, and W^{186} were prepared for total cross section measurement.

The Mark-III transistorized vernier chronotron prototype being developed for fast neutron time-of-flight measurements has been tested on the bench. The differential linearity is better than one percent, and the channel width stability is better than ± 2 picoseconds per channel over a period of 48 hours. These figures are much better than the present Mark II-vacuum tube model. In addition, the new instrument will have no duty cycle limitation, and will be built almost exclusively of commercial digital modules. The simplicity of construction and adjustment compared to the previous model will be attractive to commercial instrument manufacturers.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Approach-to-Critical Experiments Using High Exposure PuAl Fuel

Measurements were completed on the 2.0 wt % Pu-Al 16% Pu^{240} fuel rods and the results have been completed for the July, August, September Quarterly Report (HW-79054).

The following are results measured in the 0.90 inch lattice spacing, containing 835 fuel rods, that will not appear in the Quarterly Report.

	Worth (measured in numbers of fuel rods in outer ring)
Center fuel rod replaced by water	- 8.42
Cd safety rod displacing water in center position	-31.7
Copper tube (176.5 gm, 36" x 3/8" O.D.)* filled with water	-13.0
Copper rod (1030 gm, 36" x 1/2" O.D.)*	-26.0

* Same position and conditions as the Cd safety rod.

A radial traverse of the fuel worth was made by removing six fuel rods, symmetrically spaced at a given radius, and allowing water to fill the void.

Lattice units from center position	Worth (measured in numbers of fuel rods in outer ring)
2	-49.95
4	-43.66
6	-35.63
8	-27.53
10	-18.18
12	-10.43
13	- 7.375
14	- 5.285
15	- 6.185
16	- 0.147

The radius of the fuel assembly is $R - \sqrt{N\sqrt{3}/2\pi} = 15.17$ lattice units, where N is 835 fuel rods. The critical number of rods for this lattice spacing is 869.8.

The value of the rod worths measured above show some interaction between the six fuel rods at the positions near the center.

Experiments in D_2O Moderator in the PRCF

Two reports have been submitted for inclusion in the July-September Physics Research Quarterly Report (HW-79054). The titles are "Neutron Spectrum Measurements in the Plutonium Recycle Critical Facility," and "Delayed Neutron Parameters and Reactivity Measurements in the PRCF."

Reactivity measurements of 3 Pu-Al (initially 6 w/o Pu^{240}) fuel elements have been made in the PRCF to determine the loss in reactivity after irradiation in the PRTR. The elements were irradiated for 30, 56, and 90 MWD. The reactivity measurements were made relative to an unirradiated Pu-Al element. The reactivities measured were 63%, 31%, and 15% of the reactivities of the unirradiated elements for the 30, 56, and 90 MWD elements, respectively.

Measurements of the reactivity coefficient for the changes in moderator level have been made at six moderator levels between 70 and 100 inches. Data obtained earlier were for levels between 90-105 inches. The results of these measurements will yield the migration area for the PRCF and will also be used in the analysis of previous experiments.

The relative worths of a Pu-Al fuel element and four mixed oxide fuel elements in the center of the PRCF were determined from positive period data. The results are useful in that they provide a reactivity measurement of the mixed oxide elements before they are irradiated in the PRTR.

Three of the mixed oxide fuel elements (Mark I-M 1 w/o PuO_2) will go into the PRTR and the fourth will be retained as a standard. Future measurements will be made of the mixed oxide elements as a function of exposure in the PRTR. These measurements will be made relative to the standard.

The results indicate that the three mixed oxide elements (1 w/o PuO_2) are each about 7% more reactive than a Pu-Al element. The standard (which contains approximately 8% less PuO_2 than the other mixed oxide elements) was measured to be about 4% less reactive than Pu-Al.

Thus, the results are useful also for PRTR operation, for they indicate the worth of the mixed oxide elements with respect to Pu-Al.

Reactor Noise Experiments and Temperature Coefficient Measurements in the PRCF

Data for both of these experiments have been taken. A preliminary number of 0.25 mk/$^{\circ}$C, between the temperatures of 28°C and 41°C, has been obtained.

Phoenix Fuel Program

1. Mark I Parameter Survey

   Work on the Mark I parameter survey is continuing. Major emphasis at present is being devoted to more careful burnup studies and various alternative treatments of the effective Pu-240 absorption cross section.

2. Beryllium-Plutonium Cores

   Some multi-group calculations for Be-Pu systems have been started.

3. Phoenix Fuel Program Proposal

   A reasonably detailed Phoenix fuel program proposal is presently in preparation.

4. Use of MTR for Possible Phoenix Burnup Experiment

   Evaluation of MTR as a possible irradiation facility for a plutonium core is continuing. Studies completed this past month have indicated several important factors which should be recognized in more detailed future analyses. The present study is concerned with the statics and burnup of MTR cycle 108, a 1958 experiment in which plutonium-aluminum elements containing 4.4 isotopic percent Pu-240 was burned. Future studies will treat cores which contain more effective concentrations of Pu-240.

The status of the present design analysis is sufficiently advanced to be able to conclude that, for Phoenix action to be discernible, not only high Pu-240 concentrations (20%) will be necessary, but in addition, the spectrum must be made harder to increase the importance of resonance effects.

In the present study, difficulties in obtaining enough information to specify closely the actual experimental conditions of cycle 108 have been encountered. This mainly concerns the perturbing effect of hardware and penetrations immediately surrounding the MTR core. It is primarily for this reason that little confidence can be placed in the calculated multiplication values.

SNAP-50

An alternate fuel loading scheme for a SNAP-50 type reactor was investigated at the request of the Design Analysis group. The results of the study indicated that a 15 percent reduction of core weight without initial reduction of reactivity is feasible with the proposed fuel scheme. No fuel life comparisons were made.

Thermalization and the RBU Model

Since RBU uses a modified gas model in which the mass and temperature of the gas are functions of neutron energy, it is possible to calculate analytically the zeroth moment scattering kernel for such a model. In doing this, upscattering must be treated directly rather than derived from downscattering by detailed balance. Consequently, the IBM 7090 subroutine SIGGAS was modified to calculate neutron upscattering for a gas as well as downscattering and to print out the results in an array form.

In addition, SIGGAS is being modified to calculate scattering with varying temperature and mass as a function of initial neutron energy. When these modifications are complete, we will be able to compare directly the approximate model used in RBU with more exact and detailed scattering models such as the Egelstaff and Nelkin models.

PRTR Burnup Experiments

The design and construction of a gamma scan at the Radiometallurgy Laboratory has given additional data from the destructive testing of the three graded exposure Pu-Al fuel elements reported in the September monthly report. These data are from the gamma scan analysis of 1/16" wafers that were cut from the center of each fuel rod. These wafers were cut adjacent to the 1/2" section from which the macrodrill samples were obtained (see September monthly report). The plutonium isotopic analysis done on the macrodrill samples indicate a radial variation of isotopic composition of the Pu but the variation is small and does not exceed two percent. To detect this small variation of fission products by gamma scanning, a large portion of the present data received will have to be retaken with longer counting periods in order to achieve the desired accuracy.

Analysis of PRTR Burnup Experiments

A four-group set of cross sections has been obtained for use with 9-ANGIE in the analysis of the Δk measurements which have taken place in the PRCF. The upper three groups were obtained from HRG with resonance calculations performed on the Pu isotopes 239, 240, and 241. The thermal group cross sections were obtained from THERMOS problems in which the 19-rod cluster was homogenized into one large cylinder having a surface equal to 1.23 times the rubber band surface of the cluster.

Cross section ratios have been derived from the experimental burnup data assuming a value of α^{41} = .33 rather than assuming the value of $\sigma_f^{41}/\sigma_a^{49}$ = .84. The assumed value (α^{41} or $\sigma_f^{41}/\sigma_a^{49}$) is held constant throughout burnup. It would appear that α^{41} would more nearly satisfy this requirement than $\sigma_f^{41}/\sigma_a^{49}$, since the spectrum changes would affect σ_f^{41} and σ_c^{41} similarly. The derived cross section ratios using α^{41} - .33 (constant) are very similar to those derived using $\sigma_f^{41}/\sigma_a^{49}$ = .84 (constant) with the exception that α^{49}, $\sigma_c^{49}/\sigma_a^{49}$, $\sigma_a^{41}/\sigma_a^{49}$ show slightly more variation over the burnup, but approximately the same average value.

The Effect of Variation of Physics Parameters on the Accuracy of Fuel Cycle Analysis

Perturbation coefficients (PC's) have been calculated for a typical H_2O moderated power reactor. PC is defined as the change in the dependent variable caused by a unit change in an independent variable. The extent to which the PC's depend upon the particular reactor chosen for study and the model chosen for their determination has not been determined; however, since most burnup calculations are performed with ALTHAEA, the constants have been determined for the above reactor using ALTHAEA. The case selected is from the "Three Reactor Plutonium Optimization Study," by Programming Staff, HW-77082, and is briefly described below.

A typical moxtyl-fueled water-moderated reactor might operate at a fissile enrichment of 2.88% (including the naturally occurring U-235 in the UO_2), and would go to about 15,000 Mwd/t in a rather large reactor, say, 1000 Mw thermal, operated batch mode. The ALTHAEA model is used for the analysis.

The initial composition of the fuel in the core is shown below:

Isotope	Concentration in Nuclei/Barn cm
U-235	1.63850 E-04
U-238	2.26491 E-02
Pu-239	4.43178 E-04
Pu-240	1.13499 E-04
Pu-241	6.90788 E-05
Pu-242	6.25118 E-06

Some of the more interesting results are:

1. The PC's for η and α, of the fissile species, for initial reactivity are approximately three times as large as those obtained for the absolute value of the cross sections.

2. The effects of uncertainties in η and α are even more important when core life is considered. Whereas, the PC (η-239) for initial reactivity is 0.759 (change in k_{eff} per unit change in η), it is 5.251 for the expected core life.

3. Fission product yields are approximately twice as important as the absorption cross section with respect to their effect on fuel exposure and fuel cycle cost (not including Xe or SM).

The total effect on initial reactivity or other dependent variables is often quite different from what would be expected from a single perturbation coefficient. This can be seen from the approximate total derivative for the ALTHAEA model:

$$\frac{dL_j}{dq_i} = \left[\left[\frac{\partial L_j}{\partial r}\left(\frac{\partial r}{\partial \Sigma_{az}}\right) + \frac{\partial L_j}{\partial T_n}\left(\frac{\partial T_r}{\partial \Sigma_{az}}\right) + \frac{\partial L_j}{\partial F}\left(\frac{\partial F}{\partial \Sigma_{az}}\right)\right]\frac{\partial \Sigma_{az}}{\partial q_i} + \frac{\partial L_j}{\partial r}\left(\frac{\partial r}{\partial \Sigma_{al}}\right)\frac{\partial \Sigma_{al}}{\partial q_i} + \frac{\partial L_j}{\partial q_i}\right], \quad \begin{matrix} i = 1,2\ldots\ldots n \\ j = 1,2\ldots\ldots m \end{matrix}$$

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where:

r = Westcott spectral index,

T_n = neutron temperature,

F = thermal flux depression factor,

Σ_{a2} = macroscopic thermal absorption cross section,

Σ_{a1} = macroscopic epithermal absorption cross section,

L_j = j'th dependent variable,

q_i = i'th independent variable (see Table I).

Table I

Perturbation Coefficients for Initial Reactivity, Core Life, and Fuel Cycle Costs, in an H_2O Moderated Power Reactor

Independent Variable	Dependent Variables: Reactivity	Exposure	UPC - TFC Fuel Cycle Cost
U-235 Cross Section Set	0.0362	0.2625	.2851
Pu-239 Cross Section Set	0.2016	0.7358	.3258
Pu-241 Cross Section Set	0.0528	0.4714	.244
Pu-240 Cross Section	0.1056	0.0766	.1018
Pu-242 Cross Section	0.002	0.0000	--
U-238 Cross Section	0.2243	0.9967	.3658
U-235 η	0.1092	0.8432	.5296
Pu-239 η	0.7596	5.251	2.434
Pu-241 η	0.1386	2.036	1.110
Fission Product Yield	--	0.996	0.600
Fission Product Cross Section	--	0.4577	0.300
Tn °C Neutron Temperature	0.0378	0.0573	0.031
r (Spectral Index)	0.3077	1.292	0.362
F (Thermal Flux Depression Factor)	0.2366	0.977	0.2577

Note: The UPC values in this table are subject to further refinement, and are preliminary in nature and should be used for the purpose of indicating magnitudes only.

Code Development

ZODIAC Chain

The burnup version of the Physics Chain (Zero-dimensional burnup, One-DImensional core Average Criticality) is apparently operative for one-pass cases. That is, it is possible to make a complete reactor fuel burnup calculation at one time step, going through GAM, TEMPEST, SIGMA-3-H, HFN, CLERKII, and DUAL NORMAL MODES in one machine pass. Two more links, partially written, must yet be added to the ZODIAC Chain before the information from DUAL NORMAL MODES can be fed back in for computation of a new time step.

Complete documentation covering input and operating instructions for the tape now available has been distributed to ZODIAC Chain users.

SUMFUN Chain

It was necessary to overcome under pressure some of the shortcomings of SUMMIT, the GA code for calculating neutron scattering kernels in graphite. SUMMIT and two existing auxiliary codes, SUMDUM and D2BKER, have been combined in a chain with a short monitor program, SUMFUN. The SUMFUN Chain may be used to run SUMMIT (producing BCD kernels), SUMDUM (producing BCD kernels from SUMMIT intermediate output), or D2BKER (binary kernels from BCD kernels); or any combination of the three. The SUMFUN Chain thus makes it possible to obtain binary kernels in the same machine run in which BCD kernels are generated by SUMMIT, or recovered by SUMDUM. The SUMFUN Chain is operative and being used.

Documentation of input and operating instructions for the SUMFUN Chain is completed; copies will be distributed immediately.

HFN-II

A few modifications to HFN appear to be in order to keep this widely used code up-to-date with the needs of users. The modified code will be known as HFN-II to distinguish it from the version now in the SPL. Requests for comments and suggestions were distributed last month and several worthwhile suggestions have been received. The HFN-II deck, now being debugged, will incorporate the following items:

1. Search calculations will be made using upper and lower limits on the variable parameter, rather than an initial guess and estimated rate of change of k with the parameter. This will allow an automatic bolt if the desired k lies outside the parameter range. New output formats have also been designed to aid in visualizing the convergence.

2. An additional option in the boundary variation procedure previously present in one special deck has been included. In this option, when the outer boundary of a given region is incremented, the boundaries of all regions further from the origin are incremented by the same amount so as to preserve region widths. The previous option of holding all other boundaries fixed will still be available.

3. A section is being added which spacially homogenizes the reactor using the critical flux obtained by the search routine.

4. Other changes including the possiblity of checking input for data field overrun, have also been considered, but have not yet been included.

The UPDATE Subroutine has been removed from HFN-II to yield a few hundred locations for these calculations. UPDATE simply allows the user to update a data tape, which should be possible to do by other means.

SIGMA 3-H Modification

SIGMA 3-H, the code to combine the outputs of GAM and TEMPEST and produce group cross sections for input to HFN, has been modified to extend the maximum number of groups which can be treated from the previous limit of 9 to the limit of 20 imposed by HFN. The revised SIGMA 3-H is now on the Physics Chain tape.

GAM - TEMPEST Input Preparation

An auxiliary code, COMBO, designed to eliminate some of the redundancy in input data now required in the GAM - TEMPEST combination, has been written and is being debugged. Input to COMBO is the basic information required by GAM and TEMPEST, but using the SIGMA 3 library designations for the nuclides. The LISTIN input format is used. COMBO obtains the proper GAM and TEMPEST material numbers from a table and prepares a BCD input tape in the proper format for a Physics Chain run involving these codes. As no changes are made in GAM and TEMPEST themselves, they can, of course, still be operated in the usual manner if COMBO is not used.

BARNS

The errors in the subroutine calculating the inelastic transfer matrices for BARNS have been corrected. A new 180 nuclide HRG data tape has been prepared using the revised BARNS subroutines. Also reflected in this tape are the recent revisions in the RBU data for deuterium and U-235.

HRG

The investigation of the resonance integral evaluation procedure of GAM-I and HRG has continued. The time for resonance integral calculations was cut approximately in half by relaxing the convergence requirement on the integration procedure used in determining the contribution from the unresolved resonances. Since this contribution amounts to only a few percent of the total, yet its calculation requires the major fraction of the time, the small possible loss of accuracy is more than offset by the gain in computing time.

CLERK

The CLERK code document, HW-75521, is being typed. The latest revisions allow for variation of fission energy for different possible isotopes in the determination at a burnup flux, and the recoding of the NORMAL MODES input data preparation, to assure non-equal elements in the diagonal of the cross section matrix. In addition, several output options have been added.

CRAM

The CRAM code (#105 in the reactor code abstracts), a 2-D, 1-D multi-group diffusion calculation for the IBM 7090 has been ordered from the Argonne Code Center. The program is currently in use at APDA for fast reactor calculations. It should be particularly useful for fast reactor calculations of FERMI-type cores.

Isotopic Analysis of PRTR Samples

Isotopic analyses were provided on 80 samples of PRTR-irradiated fuel elements in support of the Plutonium Recycle Program. Of these 34 were uranium samples from PRTR fuel element No. 1006, 21 were uranium samples from element No. 1041, 4 were plutonium samples from element No. 1501, and there were 7 uranium and 14 plutonium samples from element No. 1101. Analyses were also provided on 3 uranium and 6 plutonium samples of pre-irradiation fuel element material and two blank samples which were tests to determine the uranium content of chemical processing reagents. Some difficulty was again experienced in maintaining stable ion-emission for plutonium samples derived from uranium-oxide fuel elements. The difficulty occurs when the uranium to plutonium ratio of the sample is about 0.1 or greater.

Instrumentation and System Studies

Minor circuit modifications have been completed on the three installed PRTR liquid effluent gamma monitors. All units have now been modified excepting the spare, and this will be done as time permits. Performance of the converted units has been satisfactory.

Electronic instrumentation for the PRTR underwater gamma scanner for fuel rods and activated wires is now assembled and ready for detailed testing at PRTR. General function and calibration tests have been scheduled.

Discussions were held for the purpose of defining operating limits for proposed aural monitoring instrument to be employed in conjunction with the PRTR automatic reactor controller. The type of audible signal required must be established before circuit development proceeds.

The PRTR automatic controller test (PRTR Test #70) was run in which a small analog computer was connected to the PRTR controller and a feedback signal from the moderator liquid level was supplied to the computer. After the proper safety circuit jumpers were installed and the controller checked, the system was placed in the automatic mode. With some adjustment the controller was made to control the simulated reactor at power levels from 10 to 70 megawatts. Using a transfer function analyzer, frequency response data were taken at 10, 20, 40 and 70 megawatts and at moderator levels of 85, 90, and 95 inches. No unusual peaks or resonances were noticed in either the valve or the moderator.

Considerable interest was shown by the PRTR operating and supervisory personnel in both the test equipment and in the automatic operation of the controller. The controller behaved very well during the entire test. After the optimum settings were obtained, transients were purposely injected into the system to determine how well the controller operated. Cursory examination of the results indicates the PRTR should be very easy to control. One problem which might give trouble is the presence of a two cycle per second oscillation on the moderator level. This oscillation has been observed earlier in PRTR Test #35.

Constants and parameters for an analog simulation of the PRCF with light water moderator and either uranium or plutonium core were received and programmed on the EASE analog computer. Results for four different cases were obtained and forwarded to the customer.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

Two independent, high-temperature test runs have now been completed in which the behavior of various materials in a nitrogen atmosphere was studied. Each test lasted 200 hours. Two sets of samples of nickel A, TD nickel, Hastelloy B, Hastelloy X, Inconel 625, Inconel 600, and molybdenum were studied in each run. One set was loosely in contact with graphite, the other was held in a metal tray of Hastelloy X. The samples were weighed before and after each run. In the first run, at 1000°C, the first three of the listed materials showed little or no weight gains. In the second run, at 1200°C, the weight gains were in the range 5 to 8 mg cm^{-2}. The next three materials, all chromium-bearing, showed weight gains of 4 to 7 mg cm^{-2} in the 1000°C run and, at 1200°C, gains of 16 to 32 mg cm^{-2} when in contact with graphite. The first three materials, all of which survived bend tests that were made following the runs, can be considered for use in the HTLTR wherever metallic members are required. Although the molybdenum became brittle in each test, it showed low weight gains and may be useful in some applications in the HTLTR. The metallographic and chemical examinations of the samples are in progress. An additional test run of 1000 hours duration is being planned.

Design drawings of the HTLTR molckup in which reactor components will be tested have been completed for the graphite, insulating brick, and containment shell. The type of metal to be used in some of the supporting parts may be changed depending upon the outcome of further high temperature tests.

For the purpose of completing Title I design of the HTLTR some tentative locations in the reactor were agreed upon for the heating elements, control rods, thermocouples, traveling wire flux monitors, and the neutron chopper including its collimator, flight tube, and detectors. In addition to the cooling rate calculations that have been made for the reactor in a nitrogen atmosphere, another set will be made which would be applicable if helium were to be used in place of nitrogen.

A three group, one dimensional diffusion theory calculation has been made in order to estimate the feasibility of reducing the over-all temperature coefficient of reactivity of the reactor throughout its operating range of temperature. The calculation indicates that, for a 30 element loading of UO_2 fuel containing 5% U^{235} by weight of U, the temperature coefficient could be reduced by a factor of 10 by the addition of an amount of gadolinium that is about 0.01% of the weight of U in the fuel. This would permit the reactor as a whole to be sensitive to changes in reactivity of the experimental region as small as a few times 10^{-7} in k.

The time-of-flight neutron spectrometer for the HTLTR will employ a conventional type of rotating shutter to pulse the neutron beam. In a survey of existing designs, none was found that completely suited the HTLTR requirements. The most suitable was the spectrometer used at KAPL. Drawings of portions of this machine were obtained and the design will be adapted to our needs. An analysis has therefore been made of the resolution and rate of data accumulation as a function of the physical dimensions of the spectrometer. This analysis will form the basis for the design of the spectrometer. Performance estimates indicate that sufficient data can be taken in eight hours to analyze an energy spectrum between 0.0025 eV and 5 eV with a resolution of 5% or better. A design and performance specification is being prepared.

NEUTRON FLUX MONITORS

Chemical separation of the fission products from three irradiated regenerating (Phoenix) detectors is in progress to prepare the samples for mass spectrometer analyses. These data should establish the validity of the experimental and calculational technique for establishing initial detector composition. A fourth detector will be discharged from KE Reactor in the near future; however, extended reactor outages are continuing to delay the exposure and discharge of subsequent samples. Arrangements are being made to have several complete regenerating detector-chambers fabricated offsite to Hanford specifications.

Experimental work continued on the B-11 beta current neutron flux monitor concept, and a new prototype is being fabricated in an effort to reduce the cable-generated noise encountered in earlier tests. The new design employs very small diameter conductors with air insulation replacing solid materials so far as possible. In addition, provisions are being made to test other beta current generating isotopes such as Al-27, Tc-99 and Li-7 which have been successfully acquired in sufficient quantities for experimental use. Fabrication of the new assemblies is nearly complete.

A purchase order was approved for a new 240 milliwatt klystron oscillator to be used in the microwave in-core neutron flux monitor experiments. The necessary production test form was nearly completed to permit experiments to be carried out in the B-Test Hole at the DR Reactor test facility.

NONDESTRUCTIVE TESTING RESEARCH

Electromagnetic Testing

The multiparameter eddy current nondestructive testing device which was previously demonstrated to be capable of separating several test parameters in a three layer test specimen is being modified for evaluation as a tubing tester. In the first test in this new application the test variables will be 1) size of irregularity, 2) location of the irregularity; for example, inner or outer surface of tubing, and 3) test probe wobble. The tester must be operated at a higher sensitivity than previously and instrument drift is excessive. A major source of this drift has been traced to the bridge drive circuit.

A search for means to extend the frequency range of a commercial vector impedance locus plotter (vectorscope) resulted in the consideration of a heterodyne technique.

A minor modification was made in the plotter (vectorscope) which permits its use as a complex voltage plotter (vectorscope) in addition to its normal use as an impedance plotter.

Heat Transfer Testing

Analytical studies of non steady-state heat flow have revealed some methods for new application to heat transfer testing of both metals and insulators. The relationships between time variable heat flow and temperature, as a function of time, thickness, and thermal properties, have been studied for the specific cases of a finite plate, and a finite plate bonded to a semi-infinite region having a different thermal diffusivity. Interesting techniques for analyzing the data from time variable heat flow tests appear to be possible, based on the theoretical results. Experimental studies to test the theoretical results are planned.

Zircaloy-2 Hydride Detection

Signal output from the experimental eddy current hydride detection equipment was previously shown to correlate with the distribution of a number of small particles in the microstructure of a short section of N Reactor process tube. Vacuum outgassing analysis of samples from the tube section showed that it contained only 20 to 30 ppm hydrogen. Further analysis is under way in an attempt to determine whether the particle distribution correlates with the concentration of iron or tin. Comparison of eddy current test results before and after the samples are exposed to hydriding conditions will be necessary to eliminate the effect of variations in the concentrations of elements other than hydrogen. It is believed that the effect of elements other than hydrogen should remain constant, provided the process tubes do not greatly exceed their normal operating temperatures, since other elements do not diffuse as readily as hydrogen.

Maps were made on six N Reactor tube samples with the eddy current equipment. These samples are being hydrided by a new low temperature process being developed by Chemical Metallurgy, HL. They will be eddy current tested again after hydriding, and then given to Materials Engineering for their brittle fracture and burst tests. Metallographic examination should be possible after the burst tests.

A completely transistorized 44 kc eddy current comparator circuit has been designed. This circuit should provide better over-all stability than the previous prototype since less external amplification will be required. Drift in the external amplifiers contributes appreciably to the over-all drift in the present 44 kc instrument.

Fundamental Ultrasonic Studies

Preliminary critical angle tests, using boundary waves, show that the effects of Zircaloy hydride is readily detectable down to concentration levels of about 300 ppm. For this concentration, the signal amplitude from the hydrided area was about twice that for the non-hydrided area. Greater concentration levels gave, in general, larger signal changes. These critical angle tests are sensitive in depth of only a few wavelengths of Zircaloy surface, however, since this is the domain of boundary wave propagation. The approximate surface area covered by these initial tests was a circular spot having diameter of about 1/16 inch. Detection of hydride in smaller areas may be possible by suitable modifications to the experimental equipment. An additional effect was observed during the hydride measurements. The presence of oxide on the Zircaloy samples also gave pronounced signal changes indicating that a critical angle test may be useful in determining changes in oxide layer depth and/or quality. The boundary wave energy exchange phenomena, which was observed during liquid interface studies, was also observed during the hydride tests.

The basic theoretical studies being conducted by Applied Mathematics Operation continued. The difficulty in obtaining attenuating wave solutions which have proper directions of energy loss has been resolved. The solutions to the Voigt model now have decreasing loss in the direction of propagation which agrees with the actual physical situation. However, the Voigt model accounts only for a linear dependence of attenuation with acoustic frequency. Most liquids and solids do not have attenuation coefficients which vary linearly with frequency and, therefore, the Voigt model is not strictly valid. A review of the literature has indicated that complete solutions to the non-linear cases of wave propagation are not yet available. In order to fully explain the propagation behavior of pulses and beams, it may be necessary to include non-linear effects. However, since the linear theory is solvable by straightforward methods and since most ultrasonic sources are not too broad in frequency band, a complete study of the linear theory appears to be beneficial. These studies are under way. The rough draft report on liquid-liquid interfaces was completed.

USAEC-AECL COOPERATIVE PROGRAM

Nondestructive Testing of Sheath Tubing

Utilization of the prototype 17 mil wall tubing tester has continued. By fitting the mechanical system with different size mandrels and O-ring seals, tubes of different diameters and lengths were tested. A total of 500 tubes have been tested to date without appreciable wear on the mechanical system. Lengths of tubing from 19 inches to 96 inches have been accurately scanned at 1800 RPM rotation speed. No vibration or translation problems were encountered. Based on this experience the basic mechanical design was firmly established. Future mechanical design efforts will be limited to changes which would facilitate operation and automation of the station. Both the accuracy and stability of the electronics were also monitored periodically while the test was in progress. The standards were detected at amplitudes which varied less than ten percent upon successive inspections. The exceptional stability of the system was evidenced by the fact that nearly identical responses to the standards were obtained from day to day without equipment adjustments of any kind.

Two separate groups of tubing have been tested to date. Group one was 25 mil wall, 425 mil OD Zircaloy tubing. This tubing was tested using a one mil notch as the reject point. At this level approximately fifteen percent of the tubing was rejected. Less than five percent of these were rejected by the circumferential channel (sound beam propagating circumferentially around the tube) with the remaining ten percent being axial channel rejects. Of these, the majority were borderline rejects. In general, signal indications obtained on the circumferential channel were definite rejects, whereas excessive background noise was evident in the axial test. Visual examination confirmed that the source of this noise was due to preferentially oriented machining marks. All indications over two mils in amplitude correlated with destructive examination.

The second group of tubing consisted of 31 pieces of 17 mil wall Zircaloy previously examined ultrasonically in Canada. The tubing was fabricated by four manufacturers and contained both good and reject tubing. Three tubes were rejected on both channels in agreement with previous AECL results. The Hanford test, however, accepted many tubes which had been rejected by the Canadian tests. In each instance of disagreement the suspect defect proved to be a long, shallow scratch and not a bonafide reject. The ability to distinguish defects of this type from the more serious short but deep discontinuities is further evidence of the advantage of smaller, focused transducers employed by the Hanford system.

Crystal evaluation tests are also continuing. A large number of crystals are being evaluated in order to gain a better understanding of which crystal parameters are of major significance and which parameters exhibit the largest variation. Of the crystals measured to date, the largest inconsistencies in as-received transducers appear to be variations in transducer Q, or damping factor. Damping is related to sensitivity, and is therefore an important transducer parameter. Large variations have been measured between separate crystals and also from point to point. In some crystals, for instance, damping was observed to vary markedly from one side of the crystal to the other, such that the resulting ultrasonic beam was grossly non-uniform.

Fabrication of standards with which the final report data will be taken has been completed. Punched notches were inserted in the Zircaloy, stainless, and aluminum tubes without difficulty. The notches appear very clean and uniform, both visually and ultrasonically. Preliminary data show the notches were detected with equal sensitivity in the 25 mil wall sections of Zircaloy, stainless, and aluminum tubes. These data also point up the need for an even smaller ultrasonic beam with which to test 10 mil wall tubing. 1.5 mil defects were detected in the 10 mil tubing; however, the information received is not so clearly defined as in the 17 mil wall tubing. Crystals which should produce a smaller ultrasonic beam have been ordered.

Attempts to derive an analytical expression depicting the behavior of Lamb waves propagating circumferentially in a hollow cylinder are continuing. Recent developments have succeeded in writing the frequency equation in a form equivalent to Lamb's flat plate equation, except additional terms involving tube radii are included to account for the curvature of the tube. As the radius is allowed to go to infinity this expression reduces to Lamb's original equation, as it should. Hopefully, this equation will become useful with completion of remaining laborious but straightforward algebraic details.

WASHINGTON DESIGNATED PROGRAM

Isotopic Analysis Program

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

Studies continued on the scintillation-type ion detector for the mass spectrometer. A model of the detector was developed which maintained an adequately-low background noise. The principal change from previous models was the use of an insulator which did not have a straight cylindrical surface as the support for the secondary-electron producing electrode. The signal to noise characgeristics were further improved by using an E.M.I. 6097 S photomultiplier. Measurements indicate that this detector counts uranium ions with an efficiency greater than 99.5 percent. A working model of this detector has now been placed on the mass spectrometer of the Isotopic Analysis Program. Tests are now in progress to determine its operating characteristics in the spectrometer and to provide suitable data-recording methods.

BIOLOGY AND MEDICINE - 06 PROGRAM

Atmospheric Physics

The field model of the zinc sulfide real time sampler was successfully operated during two diffusion experiments. This sampler system detects and continuously records the concentration of zinc sulfide in the air yielding data pertinent to acute toxicity problems. Adequate signals were recorded even at a distance of one mile downwind from the source in windy, unstable atmospheric conditions associated with large dilution.

A rough calibration of the equipment was made showing the detection limit to be approximately 4×10^{-7} grams per cubic meter, an order of magnitude better than the sampler prototype which was tested earlier in the year. One chart division represents a concentration of about 2×10^{-7} grams per cubic meter. A more precise calibration is to be determined when the power supply equipment that will be used with the sampler arrives. Since a borrowed power source was used in these tests, some modifications in the calibration can be expected.

Data from the real time sampler permit an analysis of air concentrations and their variation with time. These studies could not be pursued with the bulk sampling system which has been used for exposure measurements. The table below presents the percent of time the ratio of the instantaneous concentration and average concentration are equal to or greater than a given value, A. These data were obtained by sampling 600 meters downwind for a nineteen-minute period during a field test where zinc sulfide was released at 200 feet height in an unstable atmosphere.

PERCENT OF TIME CONCENTRATION/MEAN CONCENTRATION $\geq$ A

A	% of Time	A	% of Time
0	100.0	10	2.2
1	15.8	20	0.14
2	9.4	30	0.036
5	5.3	34.6	0.0000

Three successful atmospheric diffusion experiments were completed in October. One test was conducted in stable conditions with a release of material from the 23-foot short stack. Two tests were during unstable conditions with zinc sulfide being released from the 200-foot level of the Hanford Tower. A total of 28 successful tests have been completed this year yielding valuable data pertinent to stack releases.

Radiological Physics

The studies in Alaska in 1962 showed that the food eaten daily by some Eskimos contained twice as much Cs^{137} as they eliminated per day. This would mean that their body burdens were increasing, but there were reasons for believing that they were not changing. These statements would be consistent if all the Cs^{137} in the food either did not enter the body or were not retained by it. Measurements were made on loss of Cs^{137} from reindeer meat for different methods of cooking. Boiling reindeer meat causes up to 75% of the Cs^{137} to be transferred to the broth; on roasting about 40% appears in the juice. These liquids, however, are also consumed by the Eskimos and so cannot account for the difference noted above. Measurements were made of reindeer meat and of people before and after eating the meat and of their excreta afterwards. These indicated that almost all of the Cs^{137} in the meat is retained by the body. The conclusion reached is that the body burdens of the Eskimos must have been increasing after all. (The results on removal of Cs^{137} from meat by cooking suggest ways in which the Eskimos might be helped to reduce their body burdens.)

Chromium-5_1 and Zn^{65} were found in recent feces samples of Richland residents. Sodium-24, Cr^{51}, and Zn^{65} were observed in Richland drinking water. These facts correlate with the observation of Na^{24} in non-100-Area workers at the whole body counter and are probably due to introduction of Columbia River water into the Richland water supply.

A pooled urine sample from 14 Eskimos from Kotzebue was interpreted as from a population whose average body burden was 140 nc. The average of the 14 by whole body counting was 200 nc. This is not as good agreement as obtained lately and is probably due to non-uniform individual sample sizes. Daily urine excretion from one individual is being studied to see how constant it is.

A beta ray counter was fabricated that requires a coincidence between a thin proportional counter and a thin scintillation crystal. It is being studied for application in P^{32} counting.

Phantoms filled with potassium solutions and people with no known plutonium body burdens were counted in the plutonium counter in a study of the background of the counter. Considerable variation was found among the people. However, it was also found that their backgrounds in the pulse height range used for plutonium analysis was correlated with that in other ranges. This helps to eliminate the effects of the variation.

The positive ion Van de Graaff was operated all month at reduced voltage (below 1.5 MeV) to avoid breakdowns. Consulting with High Voltage Engineering Corporation led to a program that we will follow to try to eliminate this recurrent problem.

Some special BF_3 counter tubes were purchased this summer because they showed promise of minimizing the effects of exposure to gamma radiation. It has been found, however, that the tubes are not stable. Their resolution has doubled since this summer.

A proportional counter spectrometer for low energy fast neutrons is being fabricated. In the meantime, mathematical study of end and wall effect corrections are being made and other equipment is being readied.

More calorimetric measurements were made on Pm^{147} to establish its half-life. Calibration measurements are being made.

A pulsed X-ray machine developed by the Linfield Research Institute for the AEC Division of Biology and Medicine was received. A special power source for it will be installed before the machine is put into operation.

Instrumentation

An entomological species counter was developed for use at Radioecology, Biology Laboratory. The solid state circuited instrument employs a miniature focused light bulb with a sensitive photocell as a detector. The light shines through a tiny glass tube, through which insects move. Passage in front of the light causes a suitable signal to be provided by the photocell and the following solid state circuits drive an electromechanical register to indicate the occurrence. Testing is being done at Radioecology.

Progress was achieved on the experimental animal biological function telemetry instrumentation with the completion of working bench-model circuits for both the transmitter and receiver. All circuitry is solid state. The transmitter is battery powered and operates on a carrier frequency of 46 kc/second. Both frequency and amplitude modulation is incorporated to convey temperature, respiration, and pulse rate data. The receiver includes amplifiers and demodulators to separate the three signals for measurement. Bench testing with synthetic function inputs was satisfactory.

An inexpensive solid state timing circuit was developed and tested for use in various types of radiation monitoring instrumentation, specifically hand and shoe counters. The heart of the circuit is a General Electric Unijunction transistor, which controls a series of relays to provide control and indication of the timing cycle. Cycle times from 0.2 to 50 seconds can be achieved with the present circuit parameters.

The experimental, pocket-size, recharging dosemeter was completed in a form suitable for use by Radiation Protection Operation, HL. Laboratory tests were satisfactory and the unit will now be used for preparation of electromechanical drawings and specifications. Tentative plans are to have a number of dosemeters fabricated offsite.

Final model fabrication was partly completed on the miniature solid state integrator developed for portable neutron survey meter use. The experimental circuits have all operated correctly, and the circuits are being packaged for direct application to the neutron detection instruments.

Three solid state circuits were developed and tested in bread-board fashion. They included a discriminator and anti-coinidence unit, an improved signaling or alarm circuit, and a pulse mixer circuit. These will be applied to various radiation measurement instruments now being developed. Descriptive memoranda reports were prepared and issued.

HAPO Radiotelemetry circuitry improvements have been made with the development of a new identification oscillator. Some commercial portions of the over-all system continue to cause sporadic trouble at temperature extremes and replacement of certain components is being considered. Portions of the slave station monitor have been fabricated and tested. With the exceptions of the commercial radio-portion temperature problem and minor problems with the old stepping switches, which are being replaced with new ones, the modified remote data station is functioning satisfactorily.

Solid state circuit development was started on a special radiation detection instrument for use in field analyses at the Experimental Animal Farm, Biology Laboratory. The instrument is to provide multiple operation with various detection and readout modes.

Experimental testing of the instrumentation being developed for the measurement of air filter entrapped beryllium will probably have to be terminated since the thin nickel-foil cover over the three curie Po-210 alpha source has degenerated to the point where a health hazard exists. Test results obtained to date are promising; however, the large source required for reasonable sensitivity would pose serious questions regarding its practical use.

Field tests carried out by Atmospheric Physics, HL, with the recently developed real-time zinc sulfide airborne particle monitors have been satisfactory. It appears that airborne zinc sulfide concentrations as low as 0.17 $\mu g/m^3$ can be measured. This is an order of magnitude better than was achieved with the original experimental instrument.

Modifications to the improved sense amplifier for the memory of the 400 channel analyzer were made during the month. Tests of the amplifiers were conducted that demonstrate reliable operation to about 100°F.

TEST REACTOR OPERATIONS

The PCTR was operated routinely. There was one unscheduled shutdown due to faulty bypassing technique. The K5N experiment was completed during the month. The experiment to measure the flux enhancement in a special column, in the NPR, over the flux to be found in a normal column was started and is continuing.

The TTR was operated on a two nights a week basis for the University of Washington Graduate Center. There was one unscheduled shutdown caused by electric failure.

The 0.95 inch lattice spacing measurements in the subcritical facility were completed during the month. All fuel pieces were 1/2 inch 2 w/o Pu-Al rods, 16 w/o Pu^{240}.

CUSTOMER WORK

Weather Forecasting and Meteorological Services

Consultation service continued on meteorological and climatological aspects of oxides of nitrogen release in the 300 Area. Considerable time was spent on continued work toward preparing environmental hazard information relative to N-Reactor. Rainout and deposition calculations were made. Methods were applied for calculation of exposures from protracted releases.

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

Weather Summary

Type of Forecast	Number Made	% Reliability
8-Hour Production	93	84.2
24-Hour General	62	86.6
Special	172	87.8

October marked the third straight warm and dry month. Precipitation amounted to only 0.04 inch and brought the total since July 7 to only 0.07 inch.

Mass Spectrometry

Isotopic analyses were provided on eight samples of uranium in support of HAPO U^{233} production studies and six analyses were provided in support of research in Nucleonic Instrumentation.

Instrumentation and System Studies

Digital computer analysis of the C-column simulation is proceeding with completion and checkout of a) the data processing program for automatically determining multiplier scale factors and calculating all potentiometer settings, b) the MIDAS program for the complex mathematical model, and c) the MIDAS program for the simplified mathematical model. The MIDAS program requires only about 90 seconds of 7090 time for one complete solution. All function generator settings for the 52 runs were prepared in an attempt to minimize additional digital computer time which might be required to check the analog results.

The creep data logging and control system is now fully integrated and checkout of the system is in progress. Noise pickup problems in the solid-state logic and drive circuits are being reduced by filtering and isolation techniques.

Further analysis and experimentation on the electro-optical high temperature extensometer system have further substantiated early predictions that light diffraction at the flags should not limit resolution so long as the collimator light source approaches a point source. A point source can be approximated with conventional optics, but only with a loss in intensity. A laser source is being considered as an alternate source.

Circuitry has been designed, built and tested for use in controlling the temperature of metal calorimeter specimens. A 2N1906 power transistor and 2N2219 driver are activated by a thermocouple output voltage such that an electrical heater current is varied inversely and proportionally to changes in the thermocouple voltage.

Development of an analog simulation of ground water conditions in the Hanford area is continuing at the request of Chemical Effluents Technology, CPD. A frame to house the taper pin blocks and other necessary mechanical components are being assembled. When completed, the system will be moved to 200-W for electrical wiring.

Laboratory testing was carried out on the soil moisture measuring probe that is being developed for Advance Technical Planning, CPD, and Chemical Effluents, HL. Encouraging results were achieved by lowering the unit into a large sand-filled container and adding water to the sand. Moderated neutron detection by the BF_3 proportional counters increased in proportion to the water added. Remaining work is to provide a suitable water-tight aluminum housing for the probe.

Most of the required commercial instruments for the U-235 fuel enrichment monitor have now been ordered. The monitor is being developed for Metal Fabrication Development, Plutonium Metallurgy, HL.

Satisfactory tests were achieved with the plutonium waste container monitor developed for Plutonium Process Engineering, CPD. The project is now complete and a technical report issued.

Most Hanford developed and commercial components have been assembled for the three coincidence-count alpha air filter counters being developed for Control Operation, CPD, and for Fission Products Processing Operation, CPD. Tests on portions of the instruments have been satisfactory to date.

Optics

Modifications to newly purchased projection comparator have increased its flexibility by providing a minimum of three inches working distance at magnifications ranging from 8.5X to 100X. These changes allow the equipment to be used for gaging radioactive parts which must be examined inside a hood.

A borescope movie camera was also modified to double image size. The system is used to photograph reactor VSR channels. The design of the two-inch diameter borescope has been completed and approved. This borescope will be used for VSR channel photography and other reactor inspection work.

N Reactor seismograph system has been reviewed and recommendation made whereby its ease of adjustment and accuracy could be substantially improved. A suggestion was also proposed for incorporating changes in the system which would automatically indicate if any one of the three seismographs became inoperative.

Recommendations have been prepared for the design of a fuel element camera which will effectively "peel off" the surface of a 5/8 inch diameter by 56 inch long cylinder. Eight-foot long fuel elements will be photographed by using two overlapping shots. The image on the film will be the same size as the fuel element.

A study was made of the requirements for putting a periscope back in service for viewing waste storage tanks. Parts from two damaged periscopes are being used; new parts are being fabricated as required.

During the four-week period (September 15-October 13) included in this report, 448 man hours shop work were performed. The work included:

1. Completion of a set of goniometers and manipulators to be used in ultrasonic research by Physical Measurements.
2. Fabrication of 10 lucite prisms and 24 small glass lenses for use by External Dosimetry in an automatic badge film densitometer.
3. Reconditioning and adjustment of an internal tube surface camera for Testing Methods.
4. Servicing an underwater periscope at 105-D Area.
5. Repair of two crane periscope heads for Purex and two for Redox.
6. Repair of three camera shutters for Metallurgy Labs.
7. Fabrication of one quartz ultrasonic coupling rod for Physical Measurements.
8. Polishing and aluminizing of one large plastic scintillator for Radiological Physics.
9. Repair of a metallograph.
10. Reconditioning of two laser heads.

Physical Testing

The development of an eddy current test to obtain an initial signal trace for each of the thousand process tubes in the N Reactor was accomplished on an accelerated schedule. A Model 1004, graphical null, eddy current tester was adapted for the test and several probe assemblies were designed and fabricated. The purpose of this test is to obtain pre-irradiation data on each process tube which can be referenced to subsequent tests performed after reactor startup. The tester will operate at 200 kc with a sensitivity responsive to discontinuities occurring within the inner .040

inches of the tube wall. The test is scheduled to begin as the front face elevator becomes available.

Twelve samples of Incoloy-800 tubing have been examined in support of the N Reactor Steam Generator repair program. The effectiveness of the ultrasonic and eddy current tests was somewhat marginal due to the excessively large grain of this particular tubing, although one serious defect was detected and confirmed destructively. This material is being replaced by tubing of finer grain size.

Final modifications and alignment of all six channels are completed on the Eddy Current Motion Analyzer (ECMA-1) being developed for study of PRTR fuel vibration. The program is presently awaiting fabrication of a prototypical, instrumented fuel rod suitable for reactor irradiation.

The ultrasonic leak detector is continuing to find wide applications to plant problems. Recent tests have shown the equipment useful for the detection of leaks in overhead, lead-covered, telephone cables and for the location of very small leaks in vacuum systems. An unusual but potentially rewarding use of this instrument is being investigated in which sounds generated by pumps, gears, bearings, etc., on large machinery are periodically monitored for early detection of abnormal operation.

An ultrasonic method was developed to inspect the thread root area of main autoclave heads in use at the N Fuel Production Process Facility. Seven of fifteen units examined have not revealed defective threads.

Time required to calibrate production line ultrasonic transducers in the N Fuel Production Process Facility has been markedly reduced as a result of pre-tests now being given the transducers in 306 Building prior to their production line use. These tests experimentally determine the operating characteristics of candidate transducers and thus eliminate time consuming attempts to set up a faulty unit on the production line. Success of the program is evidenced by the heavy N Reactor Department demand for these services which now exists. Recently, modifications to the test procedure which essentially consists of replacing a ball bearing reflector with a strand of fine wire reduces the time necessary to properly examine a transducer from two hours to about twenty minutes.

ANALOG COMPUTER FACILITY OPERATION

Problems considered during the month were:

1. N Reactor Injection System.
2. N Reactor Startup Kinetics.

3. C-Column.
4. HTR (Two-Dimensional Model).
5. PRCF Hazards.

Computer utilization was as follows:

GEDA	EASE	
162	314	Hours Up
22	30	Hours Scheduled Downtime
0	8	Hours Unscheduled Downtime
168	0	Hours Idle
352	352	Hours Total

The manufacturer is approximately on schedule in the construction of the new computer. The actual construction is expected to be completed by October 29, 1963. However, due to checkout and testing procedures, shipment will not be made before November 15, 1963.

All necessary acceptance tests have been developed and checked. The step-by-step procedures are approximately 60% completed.

The old GEDA analog computer was excessed and shipped to Phillips Petroleum at Idaho Falls to make room for the new EASE computer.

INSTRUMENT EVALUATION

Acceptance testing, with a concurrent instrument maintenance technician training course, is about fifty percent completed on seven combination alpha-beta-gamma hand, shoe, and clothing contamination counters which were fabricated offsite. The training course is requiring considerable time and effort. Operation of the instruments accepted to date has been satisfactory. Three more instruments are now being fabricated offsite.

Acceptance test results were reviewed for approximately 375 self-reading pencil dosimeters, most of which were of the 200 mr full scale type with some of 10 r and 600 r full scale. About 10% of the 200 mr units were rejected due to poor response and excessive leakage. The high-range units were all acceptable.

Evaluation testing was continued at 105-B on the experimental gamma background compensated beta-gamma hand and shoe counter. Operation was satisfactory except for one simply repaired minor light leak in one scintillation shoe pr be cover.

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CHEMICAL LABORATORY

RESEARCH AND ENGINEERING

FISSIONABLE MATERIALS - 02 PROGRAM

IRRADIATION PROCESSES

Water Treatment Pilot Plant Studies

The test of the substitution of aluminum nitrate as flocculating agent and nitric acid to control water pH instead of aluminum sulfate and sulfuric acid in reactor process water treatment has been nearly completed. Water treated by the nitrate system in the Water Treatment Pilot Plant was fed to one aluminum and to one zirconium process tube. A control tube which was fed the standard sulfate system treated water was used for comparison. In the cases of both of the nitrate treated tubes the P-32 concentration in effluent water was about one-half that of the sulfate-treated control tube. This tends to confirm the sulfur as a parent of the reactor effluent P-32 by the n,p reaction. Reductions were also noted in Mn-56, As-76, Np-239 and La-140. An increase in Na-24 was noted. The reductions observed may be due to the higher purity of chemicals used in the test, the high level of control of the Water Treatment Pilot Plant, or some film or water quality change. Measurements of Ga-72 and Ni-65 which reflect aluminum corrosion rates did not indicate any increase in corrosion as a result of the nitrate system of water treatment. Use of the nitrate system appears to be a practical process for reducing the P-32 content of reactor effluent water a factor of two.

Deionized Water Studies

Because of reactor outages little firm data can be reported on the addition of 0.2 ppm sulfur as potassium sulfate to the deionized water being fed two tubes. Preliminary data seem to indicate that the sulfur results in more P-32 than would be expected if the sulfur were added in the presence of other salts. This finding was expected and the data obtained should lead to greater understanding of the adsorption-activation-desorption processes occurring in the reactor tube film. Some increases in Na-24, Ga-72 and Ni-65 were observed which may indicate an increased corrosion rate (over the deionized water rate) due to the potassium sulfate addition.

SEPARATIONS PROCESSES

Uranium-233 Production Studies

Assistance was rendered during the month to the interpretation of analytical results from the one-gram thoria irradiations (mentioned last month) and the three thorium (metal) slugs which were discharged, sectioned, and analyzed this month. Development of a one-slug dissolver and other equipment for installation in B-Cell and of flowsheets for use in same continued with equipment fabrication, testing and installation expected to be complete early in November.

Results from the one-gram samples of thorium oxide which were irradiated in DR-Reactor showed an apparent fission spectrum (fast) neutron flux a factor of five higher than expected on theoretical grounds, if all of the U-232 found was formed by the Th-230 (n,γ) and Th-232 $(n,2n)$ reactions previously considered. A third mechanism, a (γ,n) reaction on Th-232 was accordingly suggested. This reaction would have the same threshold (6.34 Mev) as the $(n,2n)$ reaction. Principal source of gammas of the required energy is from the (n,γ) reaction on aluminum, which yields a 7.72 Mev gamma in about 24 percent of the captures. Neither the gamma flux in the Hanford reactors nor the Th-232 (γ,n) cross section are sufficiently well known to permit precise calculation of the relative contribution of the two routes to U-232 production; however, it appears in the case of the one-gram samples, which were surrounded by one-half inch of aluminum, that the (γ,n) reaction is of at least the same order of magnitude as the $(n,2n)$ reaction. In the full-size thorium slugs, which were clad with far less aluminum, the (γ,n) reaction was apparently much less important. Measurement of the fission spectrum flux by determination of Na-24 (formed by Al-27(n,α) reaction) in the aluminum clad of the thorium slugs gave a value in reasonable agreement with the flux calculated from the U-232 analyses. The fission spectrum flux measurement from Na-24 production could, however, be high if some of the Na-24 came from thermal neutron activation of Na-23 impurity.

Calculations are underway to investigate build-up of potentially troublesome thorium isotopes in recycled thorium. The (α,n) reaction in the U-233 product and the possibility of producing undesirable uranium isotopes are also under study. Initial findings are as follows:

1. Neutron yield (α,n) from U-233 containing a U-232 concentration of 1 ppm is less than that from Pu-239.

2. The isotope U-230 (20.8 day half life) could affect short-cooled handling of U-233 if one or more of the potential production routes has a significant cross section.

3. The radioisotopes Th-227 (18.17 day), Th-228 (1.91 year), Th-229 (7300 year), and Th-234 (24.1 day) will all be present in irradiated natural thorium.

4. Th-228 and Th-229 concentration in recycled thorium will increase with each irradiation cycle. (Th-228 will eventually saturate.)

5. Initial calculations indicate that Th-234 may be a radiation problem unless thorium is cooled for 200 to 300 days.

6. There are several possible production routes for Th-230. The effect of these routes in Th-230 concentration after several cycles will be determined.

7. Because of the potentially large number of thorium isotopes present in recycled thorium, the thorium capsule irradiation program should incorporate a program to measure the radioactivity of fully decontaminated thorium.

Salt Effects in the Extraction of Am-241 by DBBP

The effect of the hydrogen ion concentration at constant total nitrate ion concentration has been investigated for 10 percent DBBP in xylene. Sodium nitrate was used to maintain the desired nitrate ion concentration. At total nitrate concentrations of 4 - 6 molar, the distribution of americium decreases as the nitric acid concentration increases up to about 0.7 M (limit of measurements). The distribution can be expressed in analytical form in several ways; however, the interpretation must wait until the nitric acid distribution data are determined.

Use of Uranium(IV) in the Purex Process

Studies were made on the stability of uranium(IV) when extracted into 30 percent TBP-Soltrol to aid in determining the optimum amount of uranium(IV) in solvent to add at the 1BX column feed point. Uranium(IV) aqueous solutions were prepared from uranyl nitrate by (a) reduction with Discolite (sodium sulfoxalate formaldehyde), (b) formic acid reduction catalyzed by platinum, and (c) aluminum reduction with hydrazine as holding reductant. When uranium(IV) was extracted from solutions (a) and (b), oxidation in the organic was slower than when the extraction was from solution (c). This may indicate some extraction of formaldehyde from (a) or formic acid from (b) with a consequent slowing of the oxidation rate.

Variables studies other than the source of uranium(IV) included sparging the organic with air or nitrogen; agitation in open or closed vessels, presence or absence of equilibrium aqueous phase, plant solvent versus laboratory solvent, additional washing of plant solvent with sodium carbonate or hydrazine and the presence of additional hydrazine in the aqueous phase. The data can be summarized by the statement that one may expect from 40 to 60 percent of the uranium(IV) to be oxidized during five hours in the solvent. Hence, a long standing-time between extraction of uranium(IV) into the solvent and feeding the solvent to the 1BX column is not desirable.

Disposal to Ground

No significant changes were noted in the status of ground-water contamination in the 200 Areas and other outlying wells during the current month. Gross beta concentrations in wells monitoring the Purex 216-A-10 process condensate crib remained essentially unchanged at 4-5 x 10^{-3} μc β/cc. Strontium-90 analyses of samples obtained from these wells during the current month are in progress.

Soil column tests conducted with acidic (pH 2.3) Purex process condensate confirmed the results of previous studies which showed the early breakthrough and relatively poor soil adsorption of Sr-90. Detectable Sr-90 breakthrough occurred at 1.5 column volumes (cv), and 50 percent breakthrough was noted at about 5 cv. The three test columns were run until greater than 90 percent breakthrough occurred (∿ 12 cv). At that time the acid influent was made basic, pH 9, by adding NaOH, and the soil column study was continued. The strontium concentration in the column effluent decreased to less than 20 percent of that in the influent within 2 cv, and a continual decrease in strontium concentration in the effluent occurred over the next 40 cv to an average of about 5 percent of the concentration in the influent. These test results show that neutralization of this particular waste stream with NaOH will have beneficial effects on the soil removal of Sr-90 even after a considerable amount of acid waste has been discharged to the disposal facility.

To date, about 9 cv of acid process condensate waste have been discharged to the 216-A-10 crib. The absence of Sr-90 in monitoring well samples at concentrations above the detection and ground-water limits is probably due to the spreading of waste in the subsoil and to the very low concentration (ca. 10^{-7} μc/cc) of this radionuclide in the waste stream.

I-131 in Plant Processes

Performance studies of the small charcoal test cartridge in the Redox sand filter inlet air continued. The measured efficiency continued at about the same levels - 30 to 50 percent of the radioiodine being removed from the air stream. The efficiency of the test charcoal was measured again with elemental iodine in process air after 64 days of service and found to be 58 percent versus 86 percent at 37 days of service. While the efficiency for molecular iodine is higher than the efficiency for plant iodine, the difference is now less. This indicates increasing saturation or poisoning for adsorption of molecular iodine.

The iodine released during this month has varied widely, from a high of 500 mc/day to a low of about 5 mc/day. The charcoal backup of the caustic scrubber gave an additional iodine removal of 10 - 60 percent, but the amount did not show any relation to the rate of evolution.

Studies are currently underway to determine the constituents in the air stream and to evaluate their effect on the charcoal. The stream does have the ability to oxidize iodide ion to free iodine. These oxidants could be such things as O_3 or NO_2. NO_2^- and NO_3^- analysis of scrubber solutions are being made. NO_2 in concentrations of 0.1 to 0.2 percent in room air has been passed through charcoal with spike iodine and plant iodine. In both cases iodine was eluted from the charcoal. For "laboratory" iodine about 50 percent was removed in two hours. As the air was moist, some HNO_3 was formed, so that the action cannot be uniquely ascribed to NO_2 or HNO_3. In previous tests in 1962 with NO_2 generated with copper and HNO_3, effects of this magnitude were not observed. The concentration which was used in the present test is much higher than the expected NO_2 or NO_3^- contents of the sand filter inlet air.

WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

CSREX Engineering Development Studies

Flow sheet studies are underway to develop a CSREX process for purifying the strontium in the dissolved lead sulfate cake (used to isolate strontium from acidified alkaline sludge). Strontium losses as low as 0.6 percent and 2.8 percent, respectively, have been obtained in the 1A and 1B columns. The latter utilized a 1 M formic acid scrub for the first time as a partitioning agent. (The use of formic acid as a replacement for citric acid simplifies subsequent processing.) Flow ratios used in the successful runs were 1AF/1AS/1AX/1BX = 1/0.9/3/1.8. The respective volume velocities in the 1A, 1S and 1B columns were 510, 400 and 470 gph/ft^2. The 1AF contained emulsifying solids which limited the maximum stable frequency to about 37 cycles/min; however, Mistron(R) addition to the 1AF permitted the use of higher, more efficient frequencies.

(R) Trademark of Sierra Talc and Clay Co.

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Extraction of BAMBP by NaOH and Na_2CO_3 Solutions

Measurement of the extraction of BAMBP from solvent solutions by sodium hydroxide and sodium carbonate were completed. Extractants consisting of 0.5 M BAMBP - Soltrol and 0.5 M BAMBP - 0.3 M D2EHPA - Soltrol were contacted with ten to 200 volumes of 0.1 to 2 M sodium carbonate solutions at room temperature (22-24 C) and at 60 C. The aqueous phase was contacted with carbon tetrachloride to remove BAMBP and the BAMBP was determined spectrophotometrically. Concentration of BAMBP in the sodium carbonate solutions was dependent on the final aqueous phase pH and appeared to be independent of the volume ratio and temperature in the range studied. When D2EHPA was absent in the organic phase, BAMBP concentrations in the sodium carbonate solutions ranged from 2.7×10^{-4} M at pH 10.5 to 5×10^{-5} M at pH 11.5. Even less BAMBP was extracted when the organic contained D2EHPA.

Multiple batch contacts confirmed data reported last month (HW-79046 C) for the extraction of BAMBP from 0.5 M BAMBP-Soltrol solutions by 0.1 to 3.0 M sodium hydroxide solutions. However, these experiments indicate that the data previously obtained for extraction of BAMBP from 0.5 M BAMBP - 0.3 M TBP - Soltrol solutions were high. Revised data for the concentration of BAMBP in sodium hydroxide solutions in equilibrium with this extractant range from 0.0063 M at 0.19 M NaOH to 0.0005 M at 2.67 M NaOH. These data indicate that either sodium carbonate or sodium hydroxide can be used as a wash for the CSREX solvent without serious loss of BAMBP to the wash.

Extraction of Aluminum from Redox Acid Waste with D2EHPA

Batch contact studies of the stripping of aluminum from D2EHPA extractants by nitric acid solutions were completed. Organic solutions used in these experiments were prepared by contacting simulated Redox acid waste with either four volumes of (a) 1.1 M D2EHPA - 0.5 M TBP - Soltrol, or five volumes of (b) 0.55 M D2EHPA - 0.25 M TBP - Soltrol for five minutes at 60 C. Contact of (a) or (b) for 30 minutes at 60 C with an equal volume of 5 to 10 M HNO_3 stripped 98 to 99 percent of the aluminum. Similar contact with 3 M HNO_3 stripped 96 percent of the aluminum from (b) to produce an aqueous phase 2.1 M HNO_3 - 0.24 M $Al(NO_3)_3$ and 83 percent of the aluminum from (a) to produce an aqueous 1.5 M HNO_3 - 0.38 M $Al(NO_3)_3$.

Nitric acid strips aluminum from D2EHPA relatively slowly, even at 60 C. When solution (a) was contacted at 60 C with 3 M HNO_3, 23, 36, 54 and 83 percent of the aluminum was removed at 3, 5, 10 and 30 minutes' contact time. At 25 C, only 3, 8 and 13 percent of the aluminum was removed at 5, 10 and 30 minutes' contact time.

In-Tank Solidification

Previous calculations and experimental studies demonstrated that a finned heat transfer surface is needed in the annular circulator to be

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used for concentration of intermediate level plant wastes, subsequently, a model finned circulator was fabricated and tested. The limiting, air-side heat transfer coefficient with the finned surface was measured as approximately 40 Btu/(hr)(sq.ft)(°F) compared to a value of 10 - 12 with a straight annulus. The higher coefficient provides sufficient heat transfer to meet the design criteria of an air temperature at the bottom of the circulator of 700 F or below. Fins have now been incorporated in the design of the circulator to be installed in a 200-E Area waste tank. In addition, a boildown test was conducted using "cold" coating waste. After a volume reduction of about 4 to 1, the waste solidified on cooling. The equipment is now being dismantled for determination of possible scale deposits on the circulator walls.

An alternate approach to the in-tank solidification problem is being explored - that of direct immersion of electric heaters in the waste. This method has the potential for substantial cost reductions in that equipment for heating large quantities (5000 cfm) of air would not be needed. Tests have been performed to characterize the scaling tendencies on high flux heat transfer surfaces. With heaters immersed in the air circulator, no scaling has occurred using heaters with a 100 watts-per-square-inch rating. With the heater immersed in the main body of fluid, scale was deposited only on the non-wetted surface of the sheath, i.e., that part of the heater above the liquid level. Additional investigations of this approach are in progress.

Technetium Recovery

A second, and this time fully successful, plant test of the anion-exchange technetium recovery process was carried out this month by personnel of the Chemical Processing Department. In the plant demonstration, supernate from the 103-C waste storage tank was passed through an ORNL Shielded Transfer Tank (STT) filled with 400 gallons of IRA-401 resin, during regular filling of cesium shipping casks. Although geometry and hydraulics of the STT were far from optimum, approximately 30,000 gallons of waste were passed through the cask for an average recovery of about 60 percent and recovery of over one kilogram of Tc-99. Use of a better designed column (with less channeling) should substantially decrease the losses and further improve capacity. To complete the demonstration, it is planned to elute the absorbed technetium (with 8 $\underline{M}$ HNO_3), concentrate the resultant 1200-2000 gallons of product to a manageable volume by evaporation at Semiworks, and ship to the 325-A Building for final purification, isolation and conversion to metal.

Cesium Extractants

Attempts to synthesize phenols with BAMBP-like structure are under way. The following compounds are in the process of being prepared: (a) 4-methyl-2(α-methylbenzyl) phenol, and (b) 4-acetoxy-2(α-methylbenzyl) phenol. Other phenols containing electron withdrawing groups will be prepared.

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Successful gas chromatograms have been made from solutions containing BAMBP. Previously this identification method had not been successful. Now, using short columns and high helium flow rates, it is possible to separate mixtures of TBP, Soltrol, BAMBP, and even detect the presence of D2EHPA. The last determination is possible presumably because of the thermal decomposition of the D2EHPA in the injector system of the gas chromatographic apparatus. The decomposition product has a longer retention time than that of 2-ethylhexanol.

Cesium Removal from Alkaline Waste

Tests were made to evaluate the use of raw water for washes during operation of a B-Plant ion-exchange column for cesium removal from supernatant wastes. Purex raw water was brought to pH 12 with caustic and pumped through a column of Duolite C-3 resin while being constantly stirred to suspend small amounts of precipitated calcium. Pressure drop began increasing almost immediately, increased 25 percent by 25 column volumes and another 15 percent by 40 column volumes. After 200 column volumes the pressure drop had more than doubled, and at about 300 column volumes the column plugged. A white gelatinous calcium precipitate deposited in a layer on top of the bed and was also observable in the top inch or two of voids. Although backwashing removed the material from the bed, it appears desirable to use demineralized water in column wash cycles in B-Plant.

A column of Duolite C-3 resin (4.4 grams) was loaded with approximately 5 curies of Cs-137 to determine the effect of radiation on the resin. Distilled water was pumped through the loaded column for 14 days. This exposure is estimated to be equivalent to that of 1 - 2 years of exposure of resin loaded with Cs-137 from Redox waste from the SX Farm. The water wash was made to remove soluble degradation products. About 99 percent of the activity was eluted with ammonium carbonate after 14 days. No visual change of the resin was observed. The resin was reloaded with Cs-137 and retained about 85 percent of its original capacity, an indication of good resin radiation resistance.

Fission Product Packaging

The disparity between predicted strontium loadings on Linde 4A zeolite and experimental loadings from a simulated CSREX solution was found to be due to poor kinetics. The predicted strontium loading computed from binary exchange data was 1.76 meq Sr/g compared to an experimental loading of 1.30 meq Sr/g estimated from the 50 percent breakthrough point. Recent equilibrium experiments show that 1.64 meq Sr/g are loaded when sufficient time is allowed for contact of feed solution with the zeolite.

EQUIPMENT AND MATERIALS

Scaling of Titanium Heat Exchanger Tubes

Tests with titanium heat exchanger bayonets boiling synthetic Purex 1WW [redacted]s with and without fluoride present and with varying Cr(III)/Cr(VI)

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ratios were continued. Temperatures indicated by thermocouples placed between the internal heating elements (aluminum rods) and the titanium tubes were variable but, in contrast to the initial experiment of this type, no overall increase in temperature with time was noted. An improved experimental apparatus based on a design suggested by Reactor Engineering Development personnel is being fabricated and will be used for further studies of scaling or loss of thermal transfer.

Laboratory and plant experience indicated the gradual loss of heat transfer in the Purex H-4 titanium tube bundle was due to build-up of silica on the heat transfer surfaces. In a test currently in progress, silica has not deposited, during ten days' exposure, on a titanium surface boiling simulated H-4 bottoms containing eight molar nitric acid instead of ten molar. At ten molar acid a scale about 30 mils thick was developed in ten days.

Corrosion of Mild Steel Weldments in $NaNO_3$ Solutions

Nineteen mild steel weldments fabricated by Consolidated Western Steel using a variety of manual and automatic welding techniques have been received and are being exposed to 40 percent $NaNO_3$ - 10 percent $NaNO_2$ solution at about 80 C. Primary purpose of the test is to determine if any of the weld structures are subject to attack in alkaline nitrate solutions. Previous tests revealed one weld metal structure (produced in a submerged arc weldment with Lincoln L61 wire and 760 flux) which was subject to severe attack in alkaline nitrate solution when made anodic. Attempts to produce the susceptible structure in other submerged arc weldments by heat treatment at 600, 700, 800, 900 and 1000 C followed by air cooling were largely not successful. Attack of the susceptible structure under anodic conditions was increased somewhat by heat treatment at 700 C (30 minutes in an argon atmosphere) and decreased by treatment at 900 and 1000 C.

Corrosion of Ti-Pd Alloy in $HCl-HNO_3-FeCl_3$

A material was needed in 234-5 Building operations for a heat exchanger to operate in hydrochloric acid solutions containing nitric acid ranging from dilute to zero concentration. The material is exposed to the liquid phase only. Titanium is not satisfactory for this service where the nitric acid concentration approaches zero. A Ti-2 w/o Pd alloy was investigated for the service. Corrosion rate of the alloy was 75 mils/mo in boiling 6 M HCl or 6 M HCl - 0.01 M HNO_3. However, in boiling 6 M HCl - 0.01 M $FeCl_3$ the corrosion rate was 12 mils/mo compared to a rate of > 100 mils/mo for A-55 Ti. Ferric chloride at 0.01 M is considered tolerable in the proposed application.

Non-Metallic Materials

Chemical compatibility tests of Hetron, a fire retardant polyester produced by Hooker Electrochemical Company, indicate this product has remarkably good chemical resistance. It was not affected by either 37 percent HCl or the vapor above it at room temperature, by 50 percent NaOH, Recuplex CAX, Purex HAX, carbon tetrachloride, 25 percent DBBP - 75 percent CCl_4 or D2EHPA at room temperature. It swelled 10 percent in 48 percent HF, nine percent in the vapor above 48 percent HF, and 10 percent in 60 percent HNO_3 at room temperature. It shrank 19 percent in 10 percent HNO_3 at the boiling point. Hetron is available in the form of fiberglass-reinforced sheets which have some promise as hood construction material.

PROCESS CONTROL

Plutonium Neutron Monitor

A standard HAPO BF_3 tube was tested in a high gamma flux to determine a satisfactory operating location for the plutonium monitor in a Purex recycled plutonium vessel and for other installations where high gamma backgrounds are present. Results from the test show that a slight decrease in neutron pulse height starts to occur in a gamma field of approximately 0.8×10^3 R/hr. At 2.5×10^3 R/hr considerable attenuation of neutron pulse height occurs with an increase in noise level. The detector could operate in this radiation field with proper amplifier discriminator setting; the signal to noise ratio should be approximately 5 to 1, as compared to a 60 to 1 setting out of the radiation field. At 4.0×10^3 R/hr the noise level equals the signal level, making the detector unsuitable for use in radiation fields of this intensity.

The BF_3 tube to monitor the Purex plutonium recycle tank will be located outside the vessel in a 7-inch diameter polyethylene moderator where the radiation level is expected to be less than 1.0×10^3 R/hr. The radiation level in the vessel is approximately 10^5 R/hr, accounting for the failure of a BF_3 tube previously installed in the vessel itself.

Scintillating Glass Alpha Detector

Scintillating glasses are promising candidates for in-line concentration measurement of such alpha emitters as Pu-239 and Am-241. Among their advantages are high counting yields, on the order of 40 percent, and relative ease of decontamination. Previous tests using low silica glasses showed that dilute nitric acid attacks the glass rapidly, making them unfit for many plant applications. Several new scintillators made of cerium-activated, high silica Vycor (Corning Glass Works) glass are now being evaluated. Using a 40,000 d/m dry plutonium alpha source, a net integral counting rate of about 16,000 cpm was obtained. Pulses from a radium beta source and a cesium gamma source were not detected, indicating the scintillator can discriminate against low-level beta and gamma emitters. Experiments are being conducted with varying thicknesses of cerium to ascertain the optimum thickness. Also, several of the scintillators are being exposed to nitric acid solutions, with no attack apparent after one week's exposure.

Electrodeless Conductivity Monitor

A candidate instrument for measuring electrical conductivity of process streams has been tested under non-radioactive conditions. The system consists of a glass flow tube around which two toroidal coils or transformers are situated. One coil is connected to a transmitting unit which supplies a source of stable voltage in the high audio-frequency range. The second coil or receiver coil, which functions as a current transformer, is connected to a receiving unit which is a sensitive A.C. vacuum tube voltmeter for measuring the output voltage from this winding. With constant input voltage, the output voltage of the system is proportional to the conductivity of the solution. Conductivity measurements have been made on nitric acid solutions ranging from 0.2 M to 4 M at temperatures between 30 and 50 C. On the basis of the laboratory studies, the instrument appears capable of indicating conductivity with sufficient accuracy for plant use. Re-design of the seals in the flow cell assembly proved necessary to eliminate solution leakage. Several additional problems are foreseen with a jumper installation for measuring HAW acidity and no specific plans for installation have been developed.

ANALYTICAL AND INSTRUMENTAL CHEMISTRY

Controlled Potential Coulometry

Work on the design and testing of an improved platinum-working-electrode, controlled potential coulometer cell has been completed, and construction prints distributed to the laboratories at Hanford which use coulometers. The principal changes from the cell previously in use at Hanford (HW-58491) are as follows:

1. The reference electrode has been placed between the isolated electrode frit and the working electrode.

2. The bottom (sparge) frit is specified as medium, rather than coarse porosity, to minimize the possibility of loss of sample into the frit.

3. The isolated electrode frit is specified as ultrafine porosity.

4. The glass-plug stopcock controlling the sparge and slurp lines to the cell has been replaced with a teflon plug stopcock to prevent sticking.

Hexone Impurities

Gas chromatography has been used to identify and determine acetone and mesityl oxide in hexone. Mesityl oxide can be determined well below the specification limit of 0.5 percent. Alcohol was not investigated as the samples had been previously washed with dichromate.

TBP-DBBP Identification

The structural similarity of TBP and DBBP makes positive differentiation difficult and a simple test which would also be suitable for mixtures is desirable. Both IR and NMR methods were considered. With the former it should be possible to observe the P-O stretching frequencies which are different for the two compounds, ~ 1280 cm^{-1} for TBP and ~ 1240 cm^{-1} for DBBP. A 10 percent solution in CCl_4 in a 0.02 - 0.05 mm cell was found to give a suitable spectrum in this region. Since these bands are close a convenient technique is to use a solution of TBP as a blank. Nearly complete cancellation occurs except in the P-O region where sharp bands will appear if the solutions are not identical. If desired, the solutions can be saturated with uranyl nitrate. Formation of a uranyl nitrate complex shifts the P-O bands about 80 - 90 cm^{-1} to lower frequencies and again the same effect is observed. The spectra are, however, somewhat more complex. The covalent nitrate stretching frequencies, ν_1 and ν_4, may differ slightly in the two complexes. Also DBBP will extract more uranium from an aqueous phase than does TBP. Hence additional difference peaks may be seen in the 1300 - 1600 cm^{-1} region. The diluent used is not critical. For the uranyl nitrate saturation method, a diluent giving higher distribution coefficients than found with CCl_4 would be preferable to eliminate the concentration effects. It should also be realized that band positions are slightly dependent on the diluent used.

Ten percent solutions in CCl_4 are also quite suitable for NMR investigation. The fine structure resulting from the β and γ methylene protons is quite different for the two compounds. Also it appears that the α methylene (P-C) protons in DBBP have a resonance in this region in contrast to the α methylene (P-O-C) protons which have resonances at a much lower field strength. The formation of the uranyl nitrate complex greatly shifts all α methylene protons and in the case of DBBP sharp peaks emerge.

Either technique is suitable for the differentiation of the two compounds. The IR method is recommended because of the availability of instrumentation and ease of data interpretation.

Attempts to separate the two compounds by gas chromatography have not been successful to date.

Simplified Sr-90 Determination in 300A Basin Samples

The time-consuming determination of critical Sr-90, required when 300A waste retention basin samples exceed the total beta release limit, has caused serious delay in releasing such basins. This month the delay was relieved with the advent of a simplified Sr-90 determination. It depends upon only a single carbonate precipitation which serves quantitatively to separate Sr-90. The precipitate allows a first estimate of Sr-90 content.. The method demonstrated that all basin samples having relatively high beta activity, thus far, consisted of minor amounts of Sr-90 thereby allowing immediate release of the basins. The higher beta activity comes from the normal spectrum of isotopes in the Columbia.

Technetium-99 Determination

A significant breakthrough was made in the determination of Tc-99. The newly developed method is simple, accurate and highly reliable. The single undesirable feature is the eight hours required per determination. That is offset somewhat by concurrently analyzing several samples. Also, in light of the inconsistent reliability of previous methods, the time now required is well spent.

The solution for analysis is loaded onto a 1 ml, 4 mm ID column of Dowex-1, X-4 200-400 mesh converted to the nitrate form with 4 M HNO_3. The Tc-99 is eluted with 3 M HNO_3 and 1 ml portions of the eluate are collected on stainless steel dishes, dried under a heat lamp and are beta counted with and without a greater than 80 mg/cm^2 aluminum absorber which is just adequate to stop the 0.29 Mev Tc-99 beta. The Tc-99 elution begins at fraction 9 and ends at fraction 18. The only observed interference is Ru-103-106 which "leaks" off the column during the elution. By using the no absorber/absorber ratio just prior to the Tc-99 peak a correction for Ru is made in the peak. When determining a minor technetium quantity among relatively large quantities of fission products, a single column often was not adequate for the necessary Ru decontamination. By collecting the portion of the eluate containing technetium, taking it to dryness at 90 C and dissolving the technetium in 0.1 M HNO_3 for loading onto a second column, an additional ruthenium DF of 20 was obtained. As many as three columns were used with less than two percent loss of technetium.

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Sequential Coulometric Titration of Uranium and Chloride

Reduction of 30 percent in analysis time was possible when determining both uranium and chloride in a sample. After the reduction of the uranium to the + 4 state in 1 M H_2SO_4 at -0.30 volt vs. saturated calomel electrode, the chloride was titrated at +0.30 volt. The mechanism involved in the chloride titration is the anodic dissolution of the mercury electrode and precipitation as mercurous chloride until the depletion of the chloride from the solution. The simultaneous determination can be made when uranium concentration is 5 to 50 g/l, and chloride is 2 to 20 g/l.

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Salt Cycle Process

Electrochemistry in Molten Chloride Salt Solutions - Previous measurements of the EMF of the reaction

$$UO_2 + Cl_2 = UO_2Cl_2,$$

as occurring in molten chloride salt solutions, have indicated strong dependence of EMF upon the composition of the melt. In an effort to provide a basis for the interpretation of the results for the UO_2-Cl_2 reaction, experimental work has started on similar measurements for simpler and better understood reactions. In this program, the EMF-melt composition curve for the reaction $Ni + Cl_2 = NiCl_2$ in the NaCl-KCl salt system has now been completed. As with the UO_2-Cl_2 reaction, the EMF was found to increase linearly with increase in the KCl/NaCl ratio. However, the slope of the EMF-composition line for the Ni-Cl_2 reaction was such as to suggest less dependence of EMF upon chloride "activity" than in the case of the UO_2-Cl_2 reaction. This may mean that the chloro association constant is smaller for Ni(II) than for UO_2(VI). The work will be extended to the LiCl-NaCl and LiCl-KCl melts.

Chloride Contamination in Electrolytic UO_2 and UO_2-PuO_2 - Further studies have been made of the effect of melt impurities upon the chloride content of electrodeposited UO_2 and UO_2-PuO_2 solid solutions. It now appears probable that the high chloride content of pilot plant UO_2 resulted from the presence in the melt of corrosion products from the Hastelloy C electrolysis pot lids. The evidence leading to this conclusion came from a laboratory experiment in which a sheet of Hastelloy C was suspended above a new melt for sixteen hours while UO_2 was being dissolved in the melt with a chlorine sparge. Severe corrosion of the metal occurred, with at least part of the corrosion products dropping into the melt. Subsequent electrodeposition of UO_2 from this melt gave a product which, after grinding to -200 mesh particle size and thorough washing, contained 220 ppm chloride. A sample which had not been washed so thoroughly contained 1500 ppm chloride.

Thoroughly-washed samples of UO_2 containing 0.53 and 3.3 w/o PuO_2 in solid solution were found to contain 110 ppm and 80 ppm chloride, respectively. It is gratifying to find that incorporation of PuO_2 in the UO_2 does not increase the chloride impurity level of the product.

Engineering Development - In-cell testing of the equipment installed in C-cell of the High Level Radiochemical Facility for the Salt Cycle Demonstration has been completed. Unirradiated, cold-swaged, Zircaloy-clad UO_2 fuel rods were bisected and introduced into the cell; approximately 25 pounds of UO_2 were removed from the cladding by a vibratory decladder and oxidized at 400 C to U_3O_8 with air at 2 liters per minute for 19 hours. The cladding was cut into 4-inch lengths with a hydraulic press and removed from the cell.

The U_3O_8 was dissolved in a 70-pound molten LiCl-KCl salt bath at 600 C by sparging with 4 liters per minute of 50 v/o chlorine - 50 v/o hydrogen chloride for 72 hours. The off-gas was treated with caustic, an absolute filter and a charcoal bed in series. Hypochlorite formed in the caustic scrubber was destroyed by periodic addition of hydrogen peroxide to protect the stainless steel in the spent caustic storage tanks.

A 24-pound batch of UO_2 was deposited on a pyrolytic graphite-coated cathode by electrolysis at 1.4 volts for 22 hours using a gas sparge of 50 v/o oxygen - 50 v/o chlorine. The deposit was easily removed from the cathode by splitting with a wedge mounted on the hydraulic press. The UO_2 was then cut into 1/2-inch cubes, washed with water to remove chloride, dried, crushed and screened to the proper size for reloading into fuel rods by vibratory compaction.

Plutonium Recycle Fuel Processing Economic Study - The MELEAGER computer runs required to obtain fuel burnup data for a new reactor case have been completed. There are two important differences between this reactor case and the previous one, both of which should shift the economics in the direction of increased incentive for close-coupled processing; (1) the enrichment level is higher for a given exposure, which should reduce the optimum exposure and thus increase reprocessing throughput, and (2) there is a significantly higher fraction of fissile plutonium in the spent fuel and this, combined with a higher U-235 enrichment in the spent fuel, should result in better utilization of the spent fuel uranium in close-coupled recycling. Before proceeding with the economics computations, the two reprocessing economics programs are being revised to simplify input and output formats and to provide additional output information.

Trilaurylamine Extraction of Plutonium - A single batch contact with an equal volume of 0.3 M TLA-Soltrol extracted from 95 to 99 percent of the plutonium from solutions simulating dissolver solutions that might result from HNO_3-$Hg(NO_3)_2$ catalyzed dissolution of PRTR spike fuel elements (Al-Ni-Pu alloy). This approach has been suggested as a means of making a preliminary separation of plutonium to facilitate processing two special PRTR fuel assemblies in the Redox plant.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENT

Calcination Studies

A shipment of Redox waste, sufficient for two spray calciner runs, was obtained from the Redox Plant and analyzed. A series of runs were then made in the 8-inch ("cold") spray calciner with a feed of this composition to define conditions for a forthcoming hot-cell run. Due to the fact that aluminum nitrate (a major ingredient in Redox waste) spray calcines to a very voluminous, refractory, alumina powder, use of additives (such as borate) is necessary to produce a melt. Several runs were required to define a recipe (using both borate and silica) for a "pumpable" and "atomizable" feed which will calcine to a powder melting smoothly at $\leq$ 900 C. The solidified melt appears to be a true glass of very low solubility and preparations were underway at month's end for a run with actual waste.

Cold Semiworks Spray Calciner

Measurements were made with an aspirated and moveable thermocouple to determine recirculation rates and heat duties with water feed to the spray calciner. Recirculation rates for the short (4.75 ft.) 14-inch diameter draft tube ranged from 2100 to 5000 lb/hr. Some increase in recirculation rate with increasing total mass flow (steam plus feed) was observed. There was no significant change in recirculation rate when the calciner wall temperature was changed from 700 C to 800 C. The calculated heat duty or net enthalpy change of the feed plus steam through the draft tube showed large experimental error but appeared to increase with feed rate and wall temperature.

Similar measurements with the long (8.5 ft.) 14-inch diameter draft tube have been completed but have not yet been evaluated. A new 16-inch diameter, finned draft tube has been installed to study the effect of the narrower annulus and extended surface on heat transfer to the spray. A moveable, wick-covered thermocouple has been constructed to map the wet spray zone.

Calcine Powder Melter

During the month a melt run with a powder feed rate of 10 pounds per hour was carried out. Calcined, synthetic Purex waste with added phosphate, lithium and calcium was manually fed to the melter for one and one-half hours. A layer of powder, 1/4 - 1/2-inch thick, was maintained on the surface of the melt for the duration of the run. No excessive off-gassing, foaming or pressurization of the vessel was observed. Liquid temperatures were maintained at 900 - 950 C while powder layer temperatures ranged from 400 to 800 C.

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Manual feed of the powder was necessary since plugging of the screw feeder by condensing steam still remains a problem. The powder, which is stored exposed to the atmosphere, absorbs moisture from the air. When the powder melts, the moisture is released and recondenses in the powder at the discharge of the screw feeder. New batches of powder are being stored in sealed plastic bags to prevent moisture absorption.

The bellows valve in the melter became plugged during the last run of the month. The top flange of the melter was disconnected revealing a black slag-like melt in the pot. Bits of fused metal were dispersed throughout the melt. The exact cause of this behavior has not yet been determined. Examination of the inside walls of the melter and of the bellows valve revealed no visible corrosion.

Denitrator-Evaporator

A Nionel steam heater bayonet was substituted for the previously tested 304-L stainless steel bayonet in the denitrator-evaporator. After 25-hours of boiling in phosphate adjusted simulated Purex waste using a ΔT of 40 F the overall heat transfer coefficient gradually rose from 190 to 210 Btu/(hr)(sq.ft)(°F). Inspection of the heater after boiling showed corrosion of the welds, but little pitting of the Nionel. Three Inconel-sheathed Watlow Firerod(R) electrical heaters were substituted for the Nionel bayonet. Under conditions similar to previous tests a maximum heat flux of 50,000 Btu/(hr)(sq.ft) was achieved without scaling; above this flux value a hard scale started to build up. It is estimated that the above heat flux is the critical value for the phosphate adjusted Purex waste boiling at 135 C. Flux values of less than 10,000 Btu/(hr)(sq.ft) were obtained in the steam heated bayonets due to overall heat transfer and available steam pressure limitations.

Continuous Glass Making

A laboratory study is underway on the effects of composition on the properties of two types of glass: (1) phosphate glasses derived from Purex-type waste containing fission products equivalent to 10,000 MWD burnup, and (2) ORNL-type lithium phosphate "glass". The high MWD Purex-type waste is considered representative of waste from processing of power reactor fuels and is being used as the basis for a factorial study of composition effects. In experiments to date, the presence of (synthetic) fission products raised the drip temperature to over 1050 C while increasing the phosphate to one-third over stoichiometric had no

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effect. Presence of fission products did result in a more homogenous system. Reason for interest in the ORNL lithium phosphate formulation was observation of a yellowish green second phase in material produced in the 321 Building. This second phase was shown by X-ray fluorescence analysis to be much enriched in sulfate. An additive to eliminate this second phase and improve retention of sulfate in the melt is being sought. Calcium was scouted but was ineffective.

Preparations for the "hot cell glass experiment" continued. Some items of equipment for the "cold" pilot set-up have been received and the remainder are expected soon. The 20 kw induction heater, which was to be used to heat the glass furnace, proved defective during testing. Since repair will require a minimum of several months, resistance heaters are being procured for use in initial "cold" runs.

Thermodynamic Properties of Zeolites

Rational thermodynamic equilibrium constants and standard Gibbs free-energies were determined for several synthetic zeolites and clinoptilolite with the reaction

$$2Na_{zeolite} + Ca^{+2}_{solution} \rightleftarrows Ca_{zeolite} + 2Na_{solution}.$$

The equilibrium constants of the above reaction for Linde 4AXW, 13X, AW-400, AW-500, Norton Zeolon and clinoptilolite were 10.68, 3.96, 1.08, 0.660, 0.550 and 0.572, respectively. The standard Gibbs free-energies, expressed to the nearest 100 cal/mole, were -1400, -800, 0, +200, +400 and +300 for 4AXW, 13X, AW-400, AW-500, Zeolon and clinoptilolite, respectively.

Thermodynamic constants derived from the related exchange reactions reported in the last three months can be used to determine the accuracy of the equilibrium data. For example, the Gibbs free-energy for the exchange reaction

$$(2Na_{zeolite} + Sr^{+2}_{solution} \rightleftarrows Sr_{zeolite} + 2Na^{+}_{solution})$$

minus the Gibbs free-energy for the reaction

$$(Ca_{zeolite} + Sr^{+2}_{solution} \rightleftarrows Sr_{zeolite} + Ca^{+2}_{solution})$$

should equal the Gibbs free-energy for the reaction

$$(2Na_{zeolite} + Ca^{+2}_{solution} \rightleftarrows Ca_{zeolite} + 2Na^{+}_{solution}).$$

An average error of $\pm$ 100 cal/mole from the above energy relationships indicated that the experimental equilibria of the above three systems were valid for all the zeolites studied.

Low and Intermediate Level Waste Treatment

The pilot plant for processing radioactive condensate waste was started during the month, with Purex Tank Farm condensate (PTFC) used as feed. The process consists of removing ammonia, TBP and diluent by steam stripping and removal of radioactive isotopes by a three-bed demineralizer.

The demineralizer used in the first run consisted of a thin (1/2") bed of 60-100 mesh clinoptilolite followed by a strong base anion resin and a strong acid cation resin. The effluent from the anion column was acidified to pH 3.5 - 4.0 before entering the cation column.

The first run was terminated after about 400 gallons of PTFC had been processed when an unidentified solid started plugging the thin bed and cation bed. The solid appears in PTFC after 2-5 days and is not removed by stripping. No solids are apparent in PTFC which is stripped within two days and allowed to stand for two weeks.

Before the second run is attempted, the piping will be modified to permit stripping of PTFC within two days. The stripped condensate will then be stored as feed to the demineralizer. Laboratory investigations are underway to identify the solid.

BIOLOGY AND MEDICINE - 06 PROGRAM

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

The last link needed for careful evaluation of possible numerical errors in the permeability calculation method (electrical analog of ground water flow) was obtained. Also, a closed-form solution of the stream function for the preliminary analog was obtained with the assistance of personnel in Applied Mathematics. This solution will permit direct comparison of the streamlines generated numerically with the exact flow paths which are described analytically.

All materials (power supply, resistors, cables, taper pin blocks and accessory items) for the preliminary, 662-node, analog network arrived on site and fabrication of the network was started. This experimental model is expected to be operable by the end of November.

Laboratory work for obtaining the soil parameters for partially-saturated flow is underway. The tensiometers, inflow barriers, and outflow barriers, for the soil columns were fabricated from "Lucite" and ceramic material. A 20-inch long column of fine sand from the Project was packed and saturated, and a run of permeability vs. capillary pressure during drainage is in progress. The data appear to be consistent with previous imbibition data on the same material. The static pressure head from the building ventilation system seems to present no problem with this method of obtaining data. The constant values of capillary pressure throughout the column indicate good uniformity of packing by the new soil column packer.

The rate of drilling of wells by rotary methods, as used by Geophysical Services, Inc., on Contract SA-269, was compared to that by cable-tool drilling methods used at Hanford. GSI records indicate that the Mayhew-1000, truck-mounted rotary drill they used drilled a total of 5860 feet of 4-3/4-inch hole in 182 hours of actual drilling, or an average of 32.2 feet per hour. This compares to the all-time Hanford average for cable-tool drills of 16 to 17 feet per day for Speed-Star 71 cable-tool drills or equivalent. Basalt was drilled by rotary methods at a rate of 4.2 feet per hour compared to 5-10 feet per day for comparable basalt by cable-tool methods. Should the rotary methods prove as advantageous from the cost standpoint and the technical standpoint as from the performance standpoint, then definite consideration should be given to using this method for drilling future wells at Hanford.

RADIOLOGICAL AND HEALTH CHEMISTRY

Multidimensional Gamma Spectrometry

Measurements of some members of natural radionuclide chains with multidimensional analyzer gamma spectrometry indicate that high sensitivity and selectivity can be obtained. A measurement of Bi-214 is representative of the U-238 chain from Ra-226 and would provide a sensitivity of 4 μg U-238 if the whole chain is in equilibrium. Measurement of Tl-208 is representative of the Th-232 chain from Th-228 and would provide a sensitivity of 20 μg for Th-232 if the whole chain is in equilibrium. These figures are based on a 1000 minute count and would permit measurements at levels of 0.02 ppm U-238 and 0.1 ppm Th-232 on 200 gram samples.

Measurement of U-232 in the presence of U-233 and other fission products can be made by multidimensional gamma spectrometry. Following a uranium separation from the thorium element and a short decay period a measurement of the Tl-208 is made from which the U-232 content may be determined. The Tl-208 present in 10^{-4} μg U-232 can be accurately measured with a 10 minute count 24 hours after separation of the uranium. This would provide the ability to measure 1 ppm U-232 in 100 μg of U-233.

Radiation Chemistry

The protection indices of sodium nitrite, ethanol and thiourea were determined by the dye bleaching competition method in water, isotonic sodium sulfate, and isotonic sodium chloride at various temperatures from 25 - 65 C. Except for minor differences, the protection indices in isotonic sulfate are identical with those in water while those for isotonic sodium chloride are only about 65 percent as great. Thus, it appears that the active species causing bleaching is the same in the cases of water and isotonic sodium sulfate but different in the case of isotonic sodium chloride. These studies are important in extending our work to living systems.

The reaction of hydrogen peroxide with the dye, Tropeolin O, at a pH of 11.8 was investigated over the temperature range of 56 - 87 C. The rates observed were not sufficiently large to account for the bleaching observed on irradiation of Tropeolin O, indicating that simple hydrogen peroxide attack on the dye molecules is not responsible for the bleaching. These findings are in agreement with studies made on erioglaucine.

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ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Particle Sampling Studies

Studies continued of errors resulting from sampling air streams at rates much lower than isokinetic. Results were obtained for wind speeds of 3.7 and 5.0 mph with sub-isokinetic sampling flows of 0.29 cfm. This corresponds to isokinetic to sub-isokinetic flow rates of 16.0 and 21.6, respectively. For the 5.0 mph tests, the errors associated with 4- to 25-micron-diameter particles were determined to range from 2.0 to 5.9-fold (3.2 at 18 microns). For the 3.7 mph tests, the errors associated with 4- to 18-micron-diameter particles were determined to range from 1.3 to 2.0-fold. These results are consistent with the data reported last month for 3.2 and 6.9 mph.

The difficulty in interpreting the results is that the error associated with the smallest particles, 3- to 4-micron, is significantly greater than the error for 8- to 10-micron-diameter particles for the wind speeds of 5.0 and 6.9 mph. In contrast, the error is expected to decrease continuously with decreasing particle size. Although some of the smaller particles may have been buried in the filter surface for the isokinetic sample, a satisfactory explanation for the higher error with the smaller particles has not been found.

ISOTOPES DEVELOPMENT - 08 PROGRAM

Promethium Radiochemical Studies

Work has continued aimed at defining the mode of formation of Pm-146 and at resolving measured discrepancies in the Pm-146 content of Pm-147 prepared at various sites. (Since Pm-146 is shielding limiting in well-aged promethium, this knowledge is potentially important to some proposed uses of promethium.) Two experimental approaches are being followed: (1) preparation of pure Pm-146 (by cyclotron bombardment) for decay scheme and neutron capture cross section studies, and (2) short term irradiation of U-235 in a Hanford reactor for independent measurement of Pm-146 fission yield. A sample of Nd-146 was bombarded with 8 to 10 Mev protons in the University of Washington cyclotron to produce Pm-146. Yield was lower than anticipated but appears adequate for the studies contemplated. A small sample of U-235 was irradiated for a short time in DR-Reactor and then processed for promethium purification. Sample dissolution and initial separation was performed in hot cells by carrier precipitation of rare earths on uranium followed by a rare earth fluoride precipitation to remove uranium. The fluoride was removed by NaOH metathesis and separation of individual rare earths accomplished by ion exchange chromatography using α-hydroxyisobutyrate as eluant.

Three cycles of ion exchange were required to obtain the ultra high purity needed for Pm-146 detection. The yield of Pm-148 was much lower than expected from the predicted fission charge distribution curves. More detailed investigation of this apparent discrepancy and measurement of Pm-146 will have to await decay of 53-hour Pm-149.

Ion Exchange Purification of Promethium

Experimental comparison of EDTA and DTPA as eluants for the large-scale chromatographic ion-exchange purification of rare earths was reported last month. Two runs with HEDTA, at pH 5.5 and 6.8, were made this month. Although analytical work is not yet complete, it was observed qualitatively that (1) separation was quite sharp, (2) the acid restraining bed worked quite well, (3) there was no precipitation problem, and (4) lead elutes well ahead of everything else. These runs were made with 0.015 M HEDTA to facilitate direct comparison with the EDTA and DTPA runs; however, it may be possible with HEDTA to go to considerably higher concentrations (this will be investigated in a subsequent run). As with DTPA (but not EDTA), use of HEDTA assumes a yttrium-free feed.

Cesium Purification with BAMBP

Pilot plant studies using BAMBP [4 sec butyl-2-(α-methylbenzyl) phenol] to extract and purify cesium were successfully completed. In the final series of runs, cesium losses were consistently less than 0.6 percent in the 12-foot extraction column at combined volume velocities up to 1090 gph/ft^2. Cesium losses were under one percent in the 9-foot tall stripping column at flow rates up to 550 gph/ft^2 but increased to 7 percent at 700 gph/ft^2, perhaps because of insufficient pulsation. Overall sodium and potassium decontamination factors were as high as 13,800 and 1000, respectively; however, at the highest rates tested, 700 gph/ft^2 in the scrub column, the DF's were decreased to 3800 and 490, respectively.

O F Hill
for Manager
Chemical Laboratory

WH Reas:cf

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BIOLOGY LABORATORY

A. ORGANIZATION AND PERSONNEL

Dr. Jack J. C. Hsieh joined the Radioecology Operation on October 1, 1963, as a Biological Scientist.

B. TECHNICAL ACTIVITIES

FISSIONABLE MATERIALS - 02 PROGRAM

Vertan 690

Bioassay of the "spent" Vertan 690 with young trout and cichlids during the discharge of roughly 120,000 gallons of Vertan 690 on October 2, 1963 at NPR showed that lethal concentrations to fish existed in the effluent (4700 ppm) before river dilution and at least 150 yards downstream from the shore outfall. (Past laboratory work provides an estimate of LD_{10} at 1500 ppm for trout.) Trout tested in the undiluted effluent died within two hours and the hardier cichlids died in about six hours. Five out of 15 trout held in three different live boxes placed at 50-yard intervals from the outfall died in about 18 hours. From a consideration of the protection of fish life, a higher concentration than desired was discharged.

Columnaris

The summary of the total mortality, columnaris incidence and growth for 18 weeks of observations on young trout reared in four different temperature treatments is presented below:

Water temperature	Total Mortality (%)	Columnaris Incidence (%)	Avg Wt (g) SD
Normal river water	14.4	56.3	20.5 ± 1.1
4F above normal	59.2	41.9	12.2 ± 1.4
4F below normal	13.4	46.3	31.6 ± 4.5
UV-treated river water	25.9	72.5	20.1 ± 1.0

Although UV-treated river water alleviated the columnaris incidence during the early period, this past month this group showed a striking sharp increase. This increase was also apparent to a lesser degree in other groups with the exception of the group in refrigerated water. The peak of columnaris was observed in late July in 1962, compared to the 1963 peak in October. The reason or reasons for this shift in the peak in 1963 is not clear, but possibly it may be related to the much warmer water experienced this year. In any case, the decreased mortality in fish from columnaris seems to be related to increasing resistance to the disease, as by an antigen-antibody response.

In general warmer water tends to accelerate growth in trout; however, when some "optimum" is exceeded there is a depressing effect. The growth data clearly show cooler water promotes better growth than normal river temperatures.

BIOLOGY AND MEDICINE - 06 PROGRAM

METABOLISM, TOXICITY, AND TRANSFER OF RADIOACTIVE MATERIALS

Zinc

Earlier short-term experiments indicated relatively high concentration of Zn^{65} in the gastrointestinal tract and gill filaments of trout. To follow this further over a longer period, about 100 trout were given 230 μc and killed serially. The average μc of Zn^{65} per gram of tissue at 118 days post-administration follows:

High		Intermediate		Low		Very Low	
Mid-gut	.86	Bone	.23	Gills	.025	Muscle	.0037
Hind-gut	.43	Eyes	.12	Spleen	.018		
Stomach	.38			Liver	.015		
Pyloric caeca	.38			Kidney	.015		
				Plasma	.012		
				Blood	.009		

The body burden at killing was on the average of 22 μc or about 10% of administered dose. The reason for the high concentration in the gastrointestinal trout may be due to the pancreas which is not a discreet organ in trout but is diffused, particularly over the surface of the mid-gut.

Strontium

With installation of the new radiographic unit (500 MA, 150 KVP, 1/120 - 10 sec.) nearing completion, radiographic procedures are being developed so that the entire skeleton may be visualized with at least two radiographic views of each portion of the skeleton. This complete radiographic examination, which requires 12 to 15 films per animal, is necessary to provide the greatest opportunity for detection of developing bone damage.

Examination of radiographs of two animals which were started on Sr^{90} feeding as young adults and have ingested 125 μc Sr^{90}/day for three years revealed no outstanding osseous changes. However, only a few control animals have been radiographed for comparative purposes.

Studies on the bacterial endotoxin-induced leukocyte response of miniature swine as a means of evaluating the animal's neutrophil reserve were extended by observing the response to three intravenous injections of endotoxin at 12-hour intervals (as compared to the single injection used previously). Following the initial injection of endotoxin, a transitory absolute lymphopenia, followed by an absolute neutrophilia, was observed. With each additional injection of endotoxin the neutrophil response was diminished--a phenomenon that may be associated with depletion of the neutrophil reserve or adaption to the endotoxin stimulus. The neutrophil response was evaluated on the basis of actual or percentage increase in neutrophils at their maximum values as compared to pre-injection values and by integration of total neutrophil numbers over the 48 hours following the initial injection of endotoxin. On these bases, the response of control animals was greater than animals ingesting 125 μc Sr^{90} daily. This suggests that at this level of Sr^{90} ingestion, even though the animals remain in good health under our experimental conditions, if exposed to an acute bacterial stress they would respond less favorably than "normal" swine. This extension of our previous work is part of our continuing program to develop sensitive indicators of radiation damage. To our knowledge, this is the first application of this technique in evaluating the effects of internal emitters following prolonged daily administration.

Iodine

Three Holstein cows maintained on a daily oral intake of 50 μc I^{131} per day were sacrificed. Thyroid concentration in two 5-month fetuses was about half that of their dams. Mean (I^{131}) concentrations in tissue and exudates (excluding the thyroid) ranged from 6.5×10^{-5} percent of the daily dose per gram of material in the high producing cow to 13×10^{-5} percent in the lowest producing cow. Highest tissue concentrations other than in thyroids were seen in the mammary gland, placenta and uterus. Other tissues were 1/2 to 1/3 of plasma concentrations. Thyroid concentration in the cows was ~100% of the daily dose, and milk concentrations were 0.3-0.5% of the daily dose per kilogram. It was of practical interest to note that using a regular portable field monitor (HGM-107) on a routine survey of the cows prior to autopsy, the readings obtained over the thyroid region were about 20 times background, while the readings taken over the udder and outside a container holding three gallons of milk showed only a slight deflection of the meter. Readings that were twice background were obtained by submerging the HGM tube in the milk.

This concludes our laboratory studies with dairy cattle. Only limited field studies or small short-term studies with dairy cattle can be accomplished until suitable facilities are available. There are a number of laboratory studies that we feel should be done.

Neptunium

Studies in Np^{237}-burdened female rats, employing C^{14}-labeled palmitate acid and C^{14}-labeled acetate, indicated that both decreased oxidation and increased synthesis of liver lipids was involved in the neptunium toxicity syndrome.

A technique for obtaining liver biopsy samples from sheep has been developed during preliminary work in preparation for Np^{237} toxicity studies. (This work is an extension of the earlier work done here with Np^{237} in sheep to test the effects of sex.)

Inhalation Studies

In earlier experiments DTPA was shown to be effective in removing inhaled $Ce^{144}O_2$ from dogs and rats. In these studies the $Ce^{144}O_2$ was prepared by oxidation of cerium nitrate with sodium peroxide. Preliminary results of recent tests show that DTPA is as ineffective in removing inhaled $Ce^{144}O_2$ prepared by calcination at 370 - 400 C as it is in removing $Pu^{239}O_2$ prepared by the same method.

Plutonium-239 analyses of tissues from two dogs that died 3 years, 8 months following a single exposure to $Pu^{239}O_2$ show that the lung contained 46 and 50% of the body burden, the bronchial lymph nodes 49 and 37%, and liver 2 and 6%, and bone 2% in both dogs. The total body burden at death was 2 and 2.7 μc, respectively.

Three dogs that died as a result of lung damage occurring after inhalation of $Pu^{239}O_2$ showed bronchiol-alveolar tumors, malignant type. The dogs died 154 days, 3 and one-sixth years, and 3 and one-half years after exposure, and had body burdens at death of 16, 1.4, and 1.8 μc, respectively.

Two weeks after exposure to $Ce^{144}O_2$ dogs show possible increases of levels of 17-ketoseroids in blood plasma. This was not observed in earlier experiments as is being examined further.

Although the dog colony presently exceeds adequate housing facilities, it was decided to begin the breeding program again to provide dogs for experiments planned to begin next fall. It will be necessary to modify certain existing temporary type runs in order to house the additional animals. Present crowded conditions in the colony are responsible for numerous dog fights. However, the value of the experiments now in progress justifies our endeavor to maintain a colony of about 150 dogs, one-half of which are on long-term experiments. Completion of the 60 new runs approved for construction is scheduled for September 1964.

Neutrons

Thirty-one male rats which had survived neutron doses of 150-275 rads were mated with control females, three months subsequent to irradiation. No offspring resulted, indicating complete sterility of these males extending for at least three months post-irradiation. Eighteen similarly irradiated female rats were mated with control males, one month subsequent to irradiation. Fifty percent of these irradiated females produced offspring. Average litter size was reduced some 40% below normal levels. There was no clear correlation of infertility with neutron dose.

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* Present estimates are that these runs will cost $175,000 instead of $150,000, the earlier estimate. The local AEC is preparing a revised project proposal to obtain approval for expenditure of the additional required funds.

Radiation Effects Therapy

Several generalizations can be drawn from the work done in the last several months on xenogeneic chimeras. (1) It is usually the case that host-antigraft titers appear within the first three weeks of treatment. The animals then show a diminished titer and die in the next few weeks. (2) Animals demonstrating graft-antihost titers usually demonstrate this response about a month after treatment. Their mortality rate after antibody appearance in the serum is not as decisive as in (1). (3) The appearance of antibody titer against both host and graft is usually accompanied with a significant fall in hematocrit values. (4) Because the majority of the animals succumb to the "delayed death syndrome" without showing any demonstrable antibody titers, the humoral hemagglutinin formation is not a typical response of these chimeras. Tentatively, it is felt that the emergence of these titers represents a late and drastic immunologic response between host and graft cells. It may be that another mechanism is more closely allied to the immunologic interplay between graft and host cells; i.e., cell-bound antibodies, or indeed, the cells themselves acting as macrophages and destroying their reciprocal tissues.

Microbiology

The qualitative nature of compounds released from irradiated yeast cells has been evaluated using thin layer chromotography. At least three ultraviolet absorbing components have been separated, as well as eight amino acids. Two of the ultraviolet type compounds are normally present in the supernatant from the non-irradiated cells as well as six of the amino acids which were detected. The identification of the two amino acids and one ultraviolet absorbing compound has not yet been completed.

The changes in protein-nucleic acid relationships which have been previously described as accompanying a transfer from H_2O to D_2O appear to be the consequence of a very slow physiological adjustment of bio-synthetic activities when cells are placed in D_2O. With time, the ratios of concentrations of RNA, protein and DNA appear to be re-established approximately as they exist in cells grown in H_2O.

Plant Nutrition

There has been a strong suggestion that zinc would be taken up more rapidly by plants if foliar portions were exposed to the zinc in the irrigation water. In a comparison of sprinkler irrigated with rill-irrigated alfalfa and grass, no such difference was apparent. In these tests the irrigation water was applied as two - three each irrigations, or a total of six inches of water, in the one case as sprinkler, in the other case, by rill irrigation. A more frequent and smaller application might show a slightly different result.

Plant Ecology

By following the ion content of leaves of sagebrush and hopsage from early spring until August a consistent increase in uptake of potassium in the leaves of Hopsage, but not in the leaves of sagebrush, was observed. In both, potassium was the most abundant cation measured, with calcium, magnesium, and sodium following in order. Potassium increased from 40 mg/g dry hopsage leaves in April to over 100 mg in the latter part of July when leaves were beginning to fall. During this same period potassium ranged from 15 to 30 mg/g dry sage leaf with no apparent rise during the season. Since sagebrush retains leaves throughout the year it may be that a lack of mineral buildup in sage is a factor which permits leaves to be retained over a longer period.

Columbia River Limnology

Average radionuclide content of net plankton pumped from the river below Hanford was 18 nc P^{32}, 17 nc Zn^{65}, and 57 nc Cr^{51}- all per gram wet weight. A rough approximation of preliminary data shows that in September the river was carrying approximately 100 tons of net plankton past a given point per day. The amount of plankton is dropping rapidly as cooler weather sets in.

Likewise the concentration of periphyton decreased about 50% during the past month. Zinc-65 concentration on periphyton averaged about 20 nc/g wet weight, the same as observed in net plankton.

Techniques

Attempts to implant bovine fetal thyroid tissue into the anterior chamber of sheep eyes have been only moderately successful with only two of six implants viable. This technique is being attempted as a possible tool in study of thyroid tumor development under varying conditions of radiation exposure and levels of hypothyroidism.

A previously reported ataractic agent ("Tranimal")* has been used as a pre-anesthetic in miniature swine in an attempt to produce a more predictable stage and duration of surgical anesthesia. A 100% calculated dose (C.D.) of Tranimal (5.5 mg/kg) and 50% C.C. of Pentobarbital Sodium (75 mg/10 kg) produced surgical anesthesia of 1 - 2 hours duration. Reducing the Tranimal to 75% C.D. and still giving a 50% C.D. of Pentobarbital Sodium produced ~ the same plane and duration of anesthesia as the higher dose, but reduced the recovery time ~ 50%. This combination of drugs appears to have promise in overcoming some of the difficulties encountered with general anesthetics in swine. (The dosage ranges will be extended to determine the optimum dose.)

Various techniques and sites for obtaining bone marrow from miniature swine have been attempted. At present, the iliac crest appears to be the site of choice for obtaining bone marrow biopsies in older swine for detecting changes due to chronic administration of bone-seeking radionuclides.

* Chemical structure unavailable at this time.

HG Conley
Manager
BIOLOGY LABORATORY

HA Kornberg:es

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TECHNICAL INTERCHANGE DATA
BIOLOGY LABORATORY

I. Speeches Presented

a. Papers Presented at Society Meetings and Symposiums

Annual Meeting of the Northwest Section of the Society for Experimental Biology and Medicine, Washington State University, Pullman, Washington, October 5, 1963:

Ballou, J. E. Distribution and retention of plutonium and neptunium in the rat adrenal.

McClellan, R. O. Erythropoietin in miniature swine.

Ragan, H. A., V. G. Horstman, and L. K. Bustad. Application of miniature goats in radioiodine studies.

Uyeki, E. M. Graft-host response in murine radiation chimeras.

Wood, D. H., L. K. Bustad, E. E. Elefson, E. C. Watson, I. C. Nelson, and R. O. McClellan. Radioiodine studies in dairy cattle.

American Society for Microbiology Meeting, Corvallis, Oregon, October 4-5, 1963:

Matchett, W. H. Physiological channelling of tryptophan in Neurospora.

O'Brien, R. T. Effect of deuterium oxide on attainable growth of yeast.

Hanson, W. C. Radioactivity in Northern Alaskan Eskimos and their food. Savannah River Chapter of Health Physics Society, Aiken, South Carolina. October 15, 1963.

b. Seminars (Off-Site and Local)

Hanson, W. C. Hanford radioecological studies. Aiken Naturalists Club, Aiken, South Carolina. October 16, 1963.

c. Seminars (Biology)

Palotay, J. Tissue culture and tissue implantation techniques. October 2.

O'Brien, R. T. Microbial adaptation to environment. October 9.

Smith, V. H. Neutrons and radiation protection. October 16.

Case, A. C. Techniques for the analysis of C^{14} using liquid scintillation counting. October 23.

Nakatani, R. E. Swimming performance of salmonids. October 30.

Longhurst, W. M. University of California, Hopland, Calif. African wildlife studies. October 31.

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c. Seminars (Biology) (continued)

Paulsen, C. A., University of Washington School of Medicine and Chief, Division of Endocrinology and Metabolism, USPHS Hospital Seattle. Studies on Re-cuplex Accident. October 22, 1963.

d. Miscellaneous

Nakatani, R. E. Applied statistics in biology research. Kennewick High School Biology Class, Kennewick, Washington. October 29, 1963.

II. Articles Published

a. HW Documents

None

b. Open Literature

Bustad, L. K. 1963. The biology of radioiodine. Nature 199: 1142-1144.

Casey, H. W., R. O. McClellan, W. J. Clarke, and L. K. Bustad. 1963. Acute toxicity of neptunium-237 and its relationship to liver function in sheep. Health Physics 9: 827-834.

Sullivan, M. F., S. Marks, R. C. Thompson. 1963. Beta irradiation of the rat intestine. Am. J. Path. XLIII: 527-538.

Bustad, L. K. 1963. Hanford Symposium on the Biology of Radioiodine. J. Am. Vet. Med. Assoc. 143: 770-772.

Bustad, L. K. 1963. Radioiodine: its nature and effects. Science 142: 510-516.

III. Visits and Visitors

a. Visits to Hanford

10/5/63 - Dr. R. Daubenmire of Washington State University, Pullman, checked research plots with Dr. W. H. Rickard.

10/9/63 - Drs. W. W. Cone and Grant Devoe of the WSU Experiment Station, Prosser, discussed radioecology research with Dr. Rickard and inspected insects on Rattlesnake Ridge.

10/11/63 - J. Terril and Dr. Dawes of the U. S. Public Health Service, Washington, D.C., discussed Alaskan Eskimo research data with Drs. Thompson and Hungate.

10/11/63 - Brian J. Earp, J. R. Clark Company, Seattle, consulted with P. A. Olson and R. E. Nakatani on food storage, nutrient needs of trout and disease control.

10/28/63 - Dr. William McGrane, McGrane Animal Hospital, Mishawaka, Ind., examined eyes of experimental animals.

10/28/63 - Dr. Charles Hoppen, Consolidated Edison, New York, toured.

10/28/63 - -10/31/63 - Dr. William Longhurst and wife toured facilities. Dr. Longhurst presented a seminar and especially studied procedures for counting samples with Radioecology personnel.

b. Visits Off-Site

10/5/63 - D. D. Mahlum, H. A. Kornberg, L. K. Bustad, B. J. McClanahan, J. L. Murray, R. C. Thompson, and F. P. Hungate attended the Annual Meeting of the Northwest Section of the Society for Experimental Biology and Medicine held in Pullman, Washington, at Washington State University. Papers were presented by R. O. McClellan, J. E. Ballou, D. H. Wood, H. A. Ragan, and E. M. Uyeki.

10/1-3/63 - F. P. Hungate attended the Annual Health Physics Meeting in Oak Ridge, Tennessee (ORNL).

10/4-5/63 - W. Matchett and R. T. O'Brien presented papers at the Northwest Branch of the American Society of Microbiology in Corvallis, Oregon.

10/9-17/63 - W. C. Hanson discussed Alaskan Eskimo research with AEC in Washington; toured Patuxent Research Refuge at Laurel, Maryland; presented a paper at the Health Physics Society Meeting at Savannah River Operations in Aiken, South Carolina, and also presented a paper to the Aiken Naturalists Club.

10/10/63 - C. E. Cushing, L. L. Eberhardt, F. P. Hungate, W. H. Rickard collected vegetation samples at the Wooten Game Range, Dayton, Washington.

10/11-15/63 - H. A. Kornberg discussed research with Dr. Ordal at the University of Washington, Seattle, and attended the Bio-Medical Program Directors Meeting in Boston, Mass.

10/20-26 - H. A. Kornberg toured Department of Defense laboratories on the east coast with other Hanford personnel: Picatinny, New Jersey; Edgewood Arsenal, Md., and USA Biology Laboratory at Ft. Dietrick, Md. Also discussed the budget with Division of Biology and Medicine personnel, AEC, Washington, D.C.

10/2-5/63 - W. J. Bair and B. O. Stuart attended the AEC Inhalation Toxicity Meeting held at Lovelace Foundation, Albuquerque, N.M.

10/9-12 - A. C. Case attended the 9th Annual Bioassay and Analytical Chemistry Conference in San Diego, California.

10/10-11 - E. G. Tombropoulos attended the Lipid Transport Meeting at Vanderbilt University, Nashville, Tennessee and discussed isolation of co-factors and determinations with Dr. Burch at Washington University, St. Louis, Mo.

10/29/63 - H. E. Erdman toured facilities at University of California - Donner Laboratory, Berkeley, Calif.

10/31/63 - W. H. Rickard and L. Haverfield visited the WSU Experiment Station at Prosser, Washington, to discuss ecological problems.

IV. Achievements - None

V. Honors and Recognitions - None

VI. Professional Group or Organization Assignments - None

APPLIED MATHEMATICS OPERATION

MONTHLY REPORT - OCTOBER, 1963

ORGANIZATION AND PERSONNEL

There have been no changes in organization and personnel in the Applied Mathematics Operation during the month of October.

OPERATIONS RESEARCH

Work has begun on designing an information system for maintenance activities connected with B and C Reactors. This system will encompass recording, transmission, and filing of data on maintenance work as it is done; and will yield maintenance schedules and reports. One basic element in the system is the identification of each piece of equipment in the B and C Areas. The development of such a unique list is now in process.

Work continues on developing an outage scheduling and analysis system for KW and H Reactors.

STATISTICAL AND MATHEMATICAL ACTIVITIES FOR OTHER HAPO COMPONENTS

N-Reactor Department

A format for summarizing preirradiation fuel data obtained from the PIM gauge was modified to reflect the long range goals of the Quality Measurements Group. The general structure and content for a course composed of a series of formal training sessions in statistics for all Quality Measurements personnel is being outlined.

A test procedure was proposed employing an "incomplete block design" to obtain data relative to temperature variations both within and between pre-extrusion ovens.

Assistance was given in developing a proposal for allocating total reactor production to the various process channels.

A test procedure was designed and the resulting data analyzed relative to the precision inherent in a prospective method of measuring the amount of deformation in metals under stress.

A formula was derived which gives a method of apportioning exposure in an N-Reactor tube to the various types of charges in a tube when the power curve along the tube is of a truncated cosine nature.

Work is continuing on the problem of fitting enthalpy and specific density relationships to temperature. Empirical formulae of a simple nature suitable for routine programmed data handling techniques are desired.

Work is continuing on the problem of obtaining a relatively simple formula for estimating the Pu produced in different charges in an N-Reactor tube.

Irradiation Processing Department

A statistical analysis was made of the preirradiation dimensional and weight measurements of fuel elements canned under the AlSi and hot-die-sizing processes. The results consisted of the averages, the standard deviations, 95 percent confidence limits on the mean, 50 percent and 95 percent tolerance statements for 95 percent of the population as well as comparisons of the process variances. The characteristics used were the weight and length, the maximum and minimum outside diameters measured at three points, the warp and the inside diameters measured at three points.

Work is continuing on fitting an empirical relationship between the melting point of an alloy composed wholly of Al, Zn, and Sn and the percentages of these elements.

A statistical analysis was completed on experiments to test whether such variables as the air pressure in bellows, the probolog speed, and the water flow in the tube affect the probolog readings. Eight experiments were run for different wall thickness and annulus values. The results in general indicated that the probolog values depend on some of these variables. The results included prediction relationships as well as the ordinary analysis of variance.

Work is continuing on the problem of programming a method of putting confidence limits on a predicted x, when y is known as the dependent variable in the original least squares fit. This is a problem which arises frequently in calibration work and in this instance arose in connection with the calibration of probolog equipment.

The effect of the furnace temperature and the preheat time in the hot-die-sizing canning process on five different fuel element quality characteristics are being studied with the objective of optimizing these aspects of the canning operation.

Analyses were made on the effects of increased amounts of H_2SO_4 in the normal F cleaning process on different fuel element quality characteristics such as the external and internal total count, the external and internal bond count and the total bad discs.

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An analysis of R data from nine tubes has been completed. The results, which indicate the nature of the temperature imbalance in a tube as the tubes become corroded with time are to be used as aids in designing tubes with maximum life expectancies. Regression relationships were given for each tube.

A discussion was held concerning the design of an experiment which will compare the AlSi and hot-die-sizing fuel elements ability to withstand the effects of both increased bond layer temperatures and cyclical exposures to these thermal stress conditions.

Confidence limits of 95 percent were given on the average amount of uranium per unit volume lost in the AlSi scrap, effluents, stripper caustic, and rinse water sources. The limits were based on fairly extensive sampling over a year's period.

A discussion was held on the feasibility of using a systems rather than a component reliability approach in evaluating the Panellit pressure monitoring system. Methods for acquiring pertinent information and for presenting valid indicators of reliability were discussed.

Work is being done on describing the temperature distribution in the hot-die-size fuel element during the preheat, sizing, and cooling phases of the process.

Some assistance was given in determining a method of sampling the length of times taken during morning and afternoon breaks by maintenance personnel.

Work is continuing on the problem of fitting percent burnup curves for PRTR fuel elements when MWD data from rods for the inner, middle, and outer rings are available.

Assistance is being given in the problem of estimating the amount of Pu in a crib based upon gms-Pu/liter of earth measurements.

Chemical Processing Department

Discussions were held on the possibility of constructing a mathematical model to aid in studying certain facets of group workforce efficiency.

A Brief review of plutonium buttons was made using available tabulated data (IBM program reports on button quality) regarding feasibility of meeting purity specifications recommended by Rocky Flats Plant.

A brief outline delineating scope and purpose of a proposed Z-Plant Accounability Model was informally presented.

[redacted] DECLASSIFIED

A review of the validity of the AEC survey comment on the inadequacy of the statistical control of CPD's sampling and volume measurement errors is being made.

STATISTICAL AND MATHEMATICAL ACTIVITIES WITHIN HL

2000 Program

Careful checks were continued on several FORTRAN programs involving matrix multiplication and inversion. Double precision debugging is still being used in order to locate the numerical instability connected with the solution of the nonlinear system describing mass transfer in the experimental pulse column.

Closed form solutions were obtained to a partial differential equation model for ground water flow in a medium of variable diffusivity. These solutions now provide the first opportunity of checking the accuracy and reliability of an EDPM numerical approximation method which has been proposed to handle more difficult problems.

Discussions were held on possible approaches to mathematical models to aid in studying migration and nucleation phenomena in metals.

3000 Program

Work continued to improve and simplify the EDPM program which prepares the magnetic tape input for the prototype Sheffield rotary contour gauge.

4000 Program

Analysis continued on the mathematical model for the transmission of visco-elastic waves in a Voigt solid.

Mathematical aid was given on a problem of heat conduction, and on an optimization problem in many variables. Both problems arose in the study of nondestructive testing methods.

Solutions were obtained to several heat conduction problems arising from Chemical Development's study of waste disposal methods.

Nearing completion is the HW formal report describing quantitative metallographic techniques used in evaluation of electron microscopy studies.

5000 Program

Work continued on the FORTRAN routine for calculating the channel group probabilities used in GEM calculations for the IRA file, in order that the program be compatible with the system and thus allow routine adjustment of probabilities.

The final draft of an HW formal report, "Fixed Time Estimation of Counting Rates with Background Corrections" is being printed for distribution.

Work continued on the actinide research program.

Mathematical assistance was provided in obtaining solutions for equilibrium constants which characterize the formation of polymers in an organic solution.

6000 Program

Work progressed on the Monte Carlo and triangular diffusion studies. Several modified versions of the EDPM programs for each have been successfully debugged, the data machine plotted and the results are now being studied and compared.

The statistical analysis of data from a study to investigate the discrimination against Sr^{90} relative to Ca^{45} as related to the age of miniature swine at the time of radionuclide administration was completed.

A computer program was written to record data and calculate certain factors from data of radioactive particle inhalation studies. The program also fits the percent body burden as a power function of time by method of least squares and calculates 95 percent confidence limits.

The analysis of data from an experiment to investigate cell proliferation in salmon using C^{14} thymidine was started. The origin and migration of cells in several tissues of yearling silver salmon for different water conditions will be investigated.

A statistical analysis was initiated of data from Pm^{144} inhalation studies performed on dogs. The purpose of the analysis is to express the whole-body retention of Pm^{144} as a function of time and to compare the statistical models for different dogs.

Carl A Bennett

Manager
Applied Mathematics

CA Bennett:dgl

[redacted]

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Assessment of French Cycle Analysis Efforts

A review of papers presented by the French at the Baden-Baden meeting on Fuel Cycles for Power Reactors, September 9-14, 1963, was begun. One paper by Messieurs Gaussens and Thiriet presents a model for a completely managed nuclear power economy using fast breeders and thermal burners. They point out a necessity of planning for the distant future. Because it takes five years or so to construct a reactor which will then operate for thirty years, decisions made today affect events beyond the year 2000. Their model is based on linear programming techniques. Optimization of the economy are performed for each year within the framework of a "dynamic program" for the entire length of the period studied. This "dynamic program" does not seem to be the same technique as that used by Bellman, Aris, and others for multistage optimization.

Another paper by Monsieur Gibrat on the long term energy economy stresses the importance of fast breeder reactors to conserve the world-wide resources of fissile material. He foresees an exponential increase in power requirements, which nuclear power will fulfill to an ever-increasing extent. He chides the Americans for their slowness in starting towards the nuclear future. Several pages of the report are devoted to implication of plutonium "price." The Americans -- and others -- are chided for their confusion of price and value. The work done on this subject at Hanford is highly praised and used as a basis for much of his discussion.

Plutonium-Seed-Uranium-238-Blanket Studies

Revisions have been made to the ALTHAEA code output and to the QUICK code so that QUICK can calculate and tabulate fuel costs in a multiregion reactor weighted in proportion to the volume and power generation in the region.

Due to the very slow debug turn around time and heavy load of computing this month incomplete results were obtained. Figure 1 shows the variation of the power generated in the blanket as the seed-to-blanket volume ratio is varied. It was assumed that the reactor was composed of many control-seed-blanket modules, each module being cylindrically symetrical with the control rod in the center surrounded by seed with the blanket surrounding the seed. Three curves are shown for control rods of various radii. The small control rod seems to give the best power sharing (that is, the largest contribution from the blanket) but the extreme macroscopic absorption required in this rod causes marked radial variations of flux (and therefore, power) in the seed. An effort will be made to find a better design.

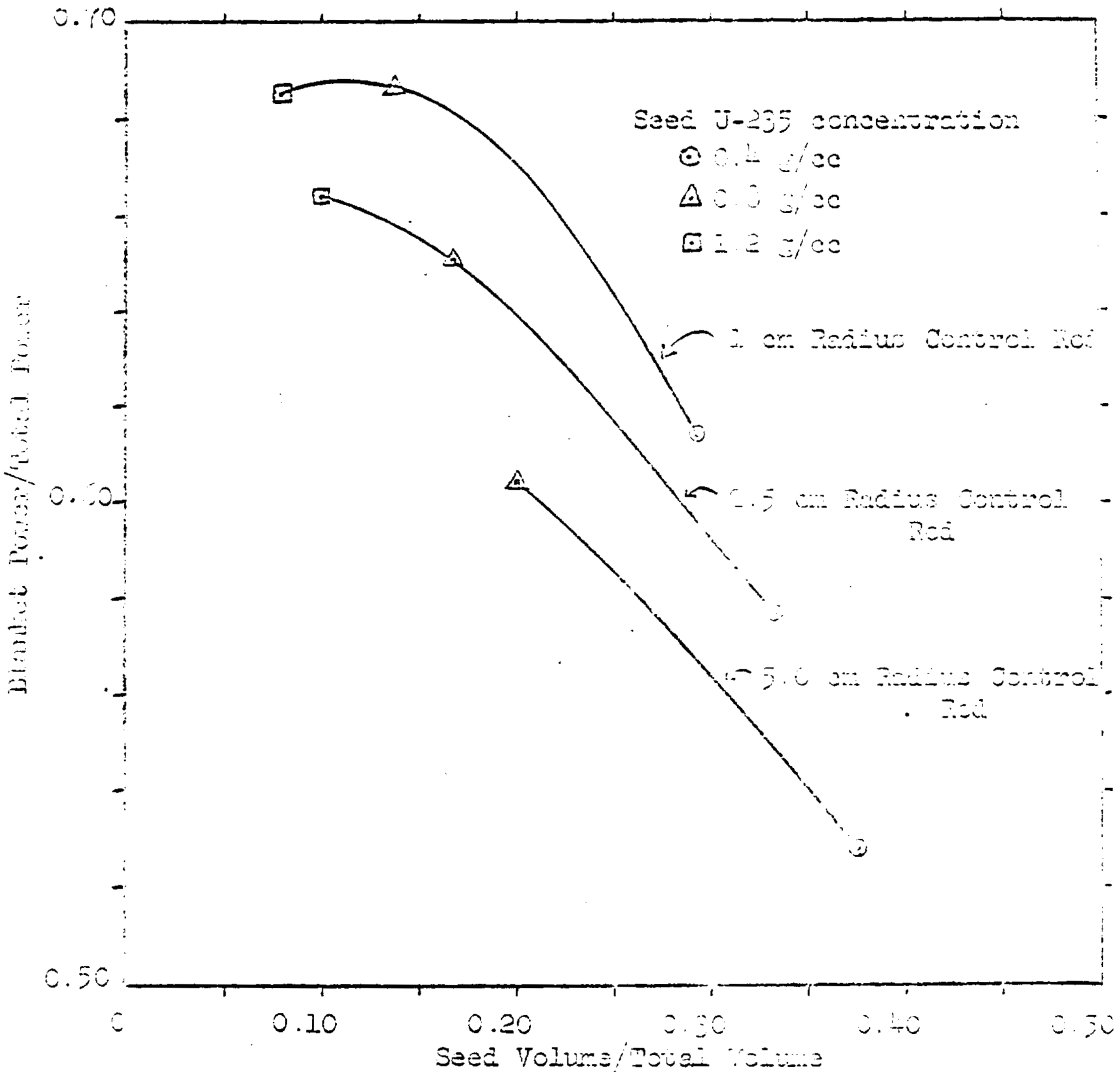

FIGURE 1

BLANKET POWER VERSUS SEED VOLUME

UNCLASSIFIED

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Indifference Pricing of Isotopes Leading to the Formation of Curium-244 and Plutonium-238

Curium-244 and plutonium-238 are both valuable isotope heat sources. Setting a high price on curium-244 would surely distort the plutonium prices systems now in use. Calculations were made to determine the effect on the price of individual precursor isotopes as a function of curium-244 or plutonium-238 price. The work was done by irradiating widely varying plutonium and uranium-236-neptunium-237 compositions in a simulated water reactor and adjusting the isotope prices so that the fuel cost is the same for each composition. At this point, the prices are set so that the reactor operator would be indifferent as to whether he produced curium-244, plutonium-238, or not; so, if the market price is greater than the indifference price, he would sell the plutonium or neptunium; and, if below, he would make higher isotopes. Figures 2 and 3 show the indifference value of each isotope as a function of curium-244 and plutonium-238 price. Note, in the curve with plutonium, if the value of curium-244 is set to zero the plutonium isotopes have thermal reactor values. The difference in the increase in plutonium values above this point represents the value of these isotopes as precursors of curium-244. It is obvious from these curves that setting a high price on these heat source isotopes will distort the plutonium value considerably and also reduce the fuel cost of reactors using plutonium recycle.

Uranium-233-Thorium Breeder Study

A study was made of the variation of breeding ratio as a function of specific power and enrichment in a heavy water moderated reactor that used uranium-233 in thorium-232. The results obtained demonstrated that breeding ratio is inversely proportional to specific power and enrichment.

Figure 4 shows the variation of breeding ratio with exposure (a function of enrichment) for various specific powers. When specific power was increased from 10 megawatts per ton to 75 megawatts per ton, the exposure at which the maximum breeding ratio occurred decreased from 26,000 megawatts per ton to 7000 megawatts per ton while the maximum breeding ratio decreased from 1.138 to 1.035. In general, breeding ratio increased with increasing exposure (increasing enrichment) to a maximum value. Then, as exposure (enrichment) further was increased, breeding ratio decreased. The decrease was due to the build up of fission products (which eventually override the formation of uranium-233) as the time required to attain a given exposure increased. The decrease of the maximum value of breeding ratio with increasing specific power is due to the higher flux level which increases the probability of absorption in protactinium-233 and the resulting loss of uranium-233 atom. The higher flux also means the equilibrium xenon and samarium concentration will be higher. It also results from the short exposure which limits the amount of uranium-233 formed by the decay of protactinium-233.

ISOTOPE PRICES VERSUS CURIUM-244 CREDIT FOR A WATER REACTOR

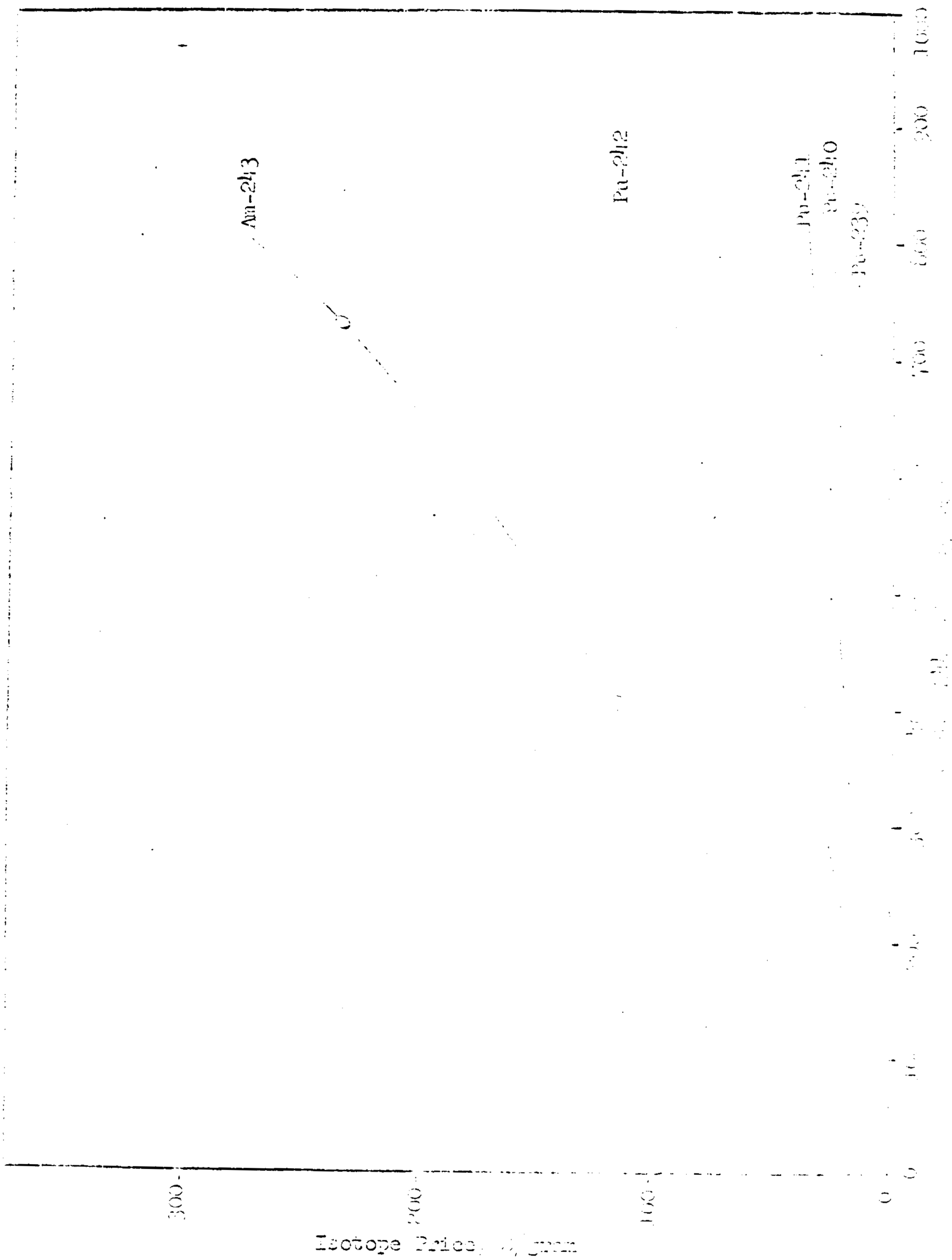

FIGURE 2

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ISOTOPE PRICES VERSUS PLUTONIUM CREDIT FOR A WATER REACTOR

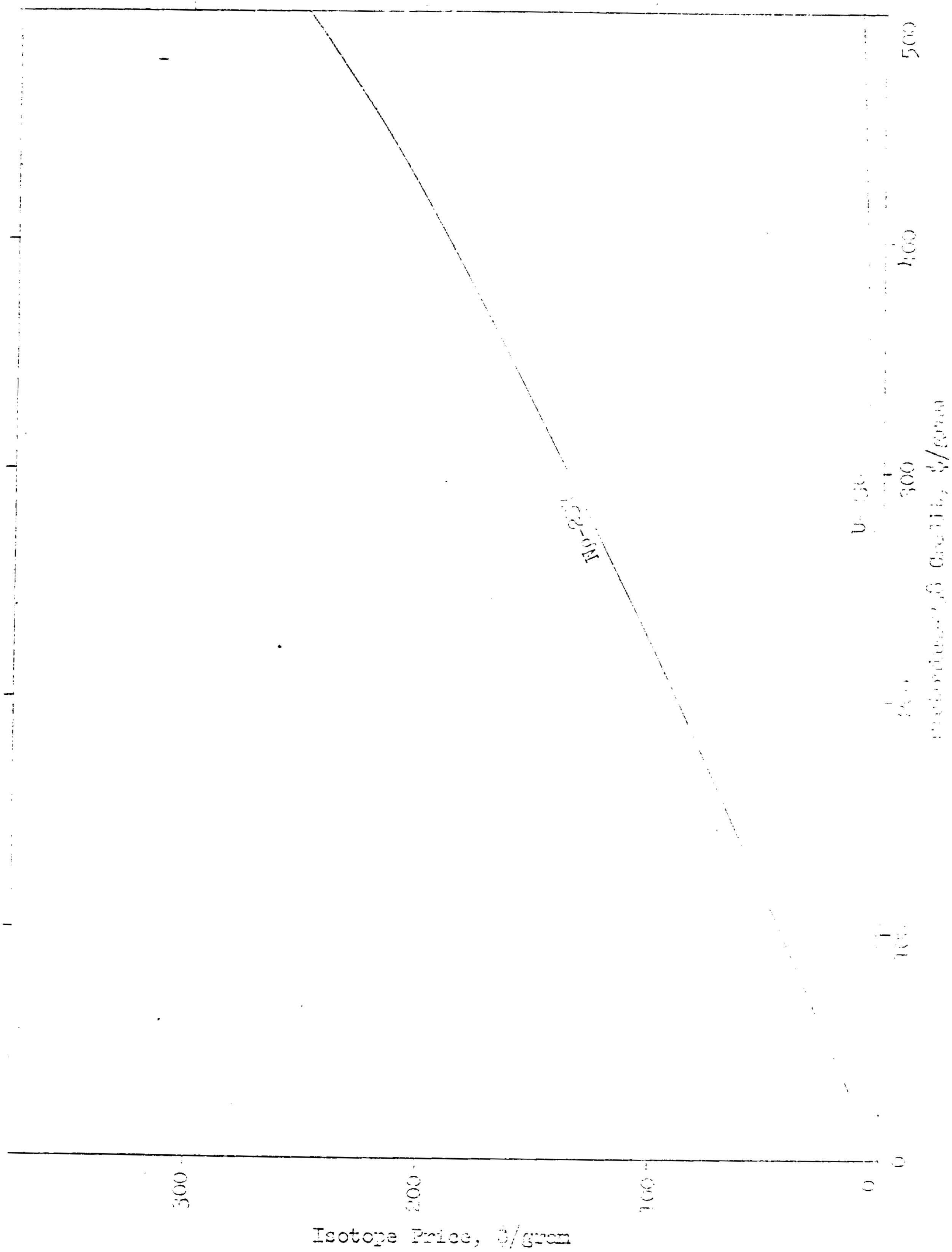

FIGURE 8

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BREEDING RATIO VERSUS EXPOSURE FOR VARIOUS SPECIFIC POWERS

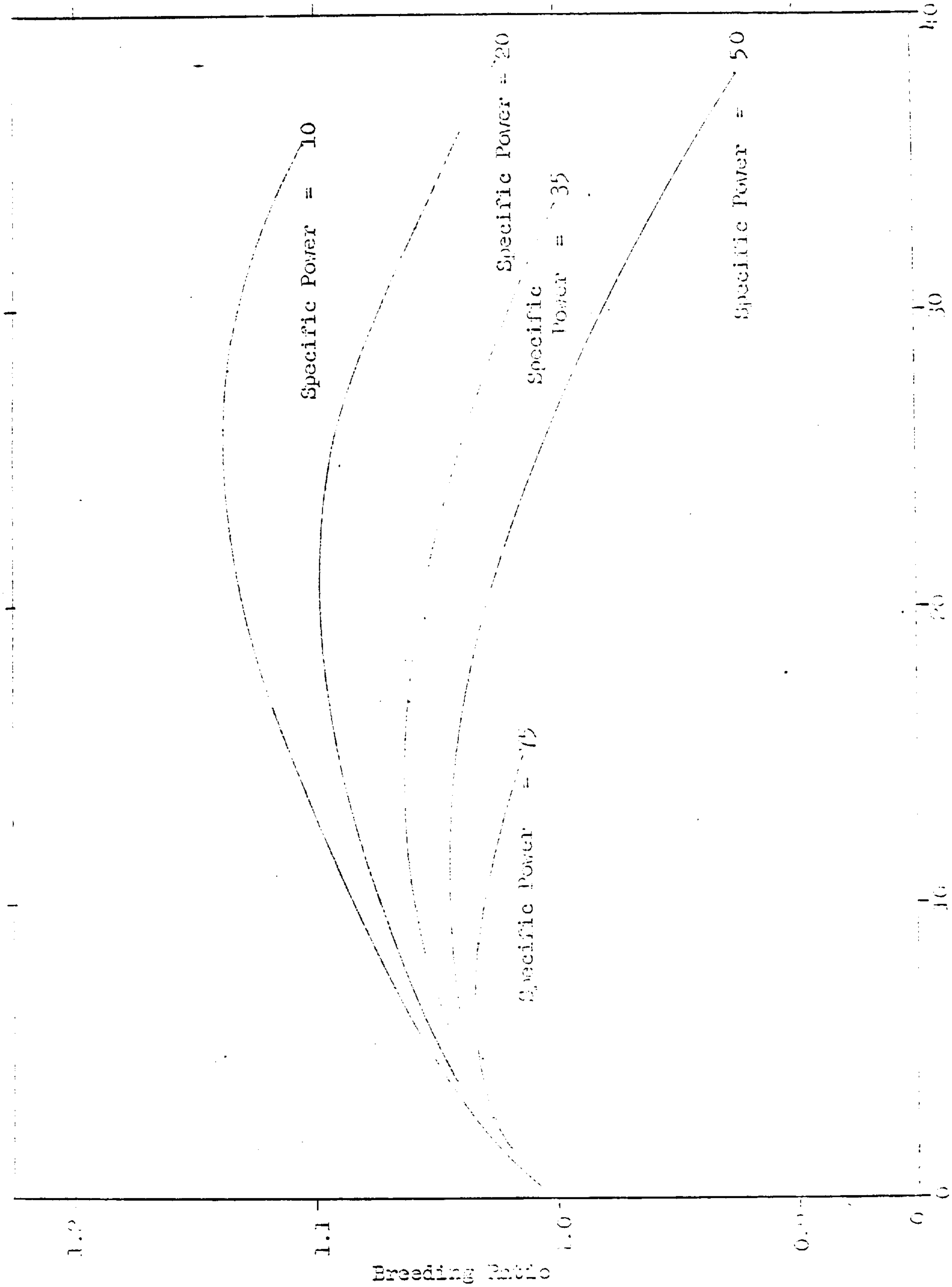

FIGURE 4

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The variation of breeding ratio with enrichment at different specific powers is shown in Figure 5. At low specific powers, breeding ratio increases rapidly with enrichment to a maximum value, after which further enrichment is of no benefit. The rapid increase in breeding ratio at low specific power occurs at that enrichment which causes the reactor to run to a large exposure. During this time, the protactinium-233 present can decay to uranium-233 with small losses due to neutron absorption.

Reduced Density Thorium Fuels

Previous studies of reduced density fuels have centered primarily on plutonium enriched uranium fuels. Another fueling scheme of interest is the plutonium enriched thorium fuel cycle. Results obtained to date indicate that when plutonium batches containing significant amounts of plutonium-240 are used to enrich thorium, it is often desirable to reduce the effective fuel density.* As with uranium fuels, this becomes desirable when the fertility of the system is greater than can be economically supported.

The cases reported simulated a water moderated and cooled reactor with zirconium cladding and a moderator-to-fuel-volume ratio of 1.7/1. The operating specific power was constant at 165 watts per cubic centimeter. (This is equivalent to 15 megawatts per ton of UO_2 at 10.5 grams per cubic centimeter density, a specific power level in a range commonly considered for central station power plants.) Figure 6 shows curves for comparable uranium and thorium cases. In the plutonium enriched uranium cases, the concentration of fertile material became excessive above 5 grams per cubic centimeter effective fuel density (the uranium concentration was varied while the grams per cubic centimeter of plutonium were held constant) and the reactivity became less than necessary for criticality.

In the plutonium enriched thorium cases, it appeared that fuel operation would be possible at any effective fuel density between 1 gram per cubic centimeter and 10 grams per cubic centimeter; however, the most economical effective fuel density appeared to be in the range of 4 to 5 grams per cubic centimeter. Note that for the batch irradiation method

* The "effective fuel" is defined to be the isotopes in the fuel with an atomic number of 90 (thorium) or greater. In these cases, it includes thorium, uranium, and plutonium in the initial fuel. The effective fuel consists of fissionable or fertile material and intermediate decay and capture products.

The "fuel" is defined as including all material in the fuel region and includes both diluents and effective fuel. A clear distinction between "fuel" and "effective fuel" is essential.

Diluent is defined as any material in the fuel other than the "effective fuel."

BREEDING RATIO VERSUS URANIUM-233 ENRICHMENT AT VARIOUS SPECIFIC POWER

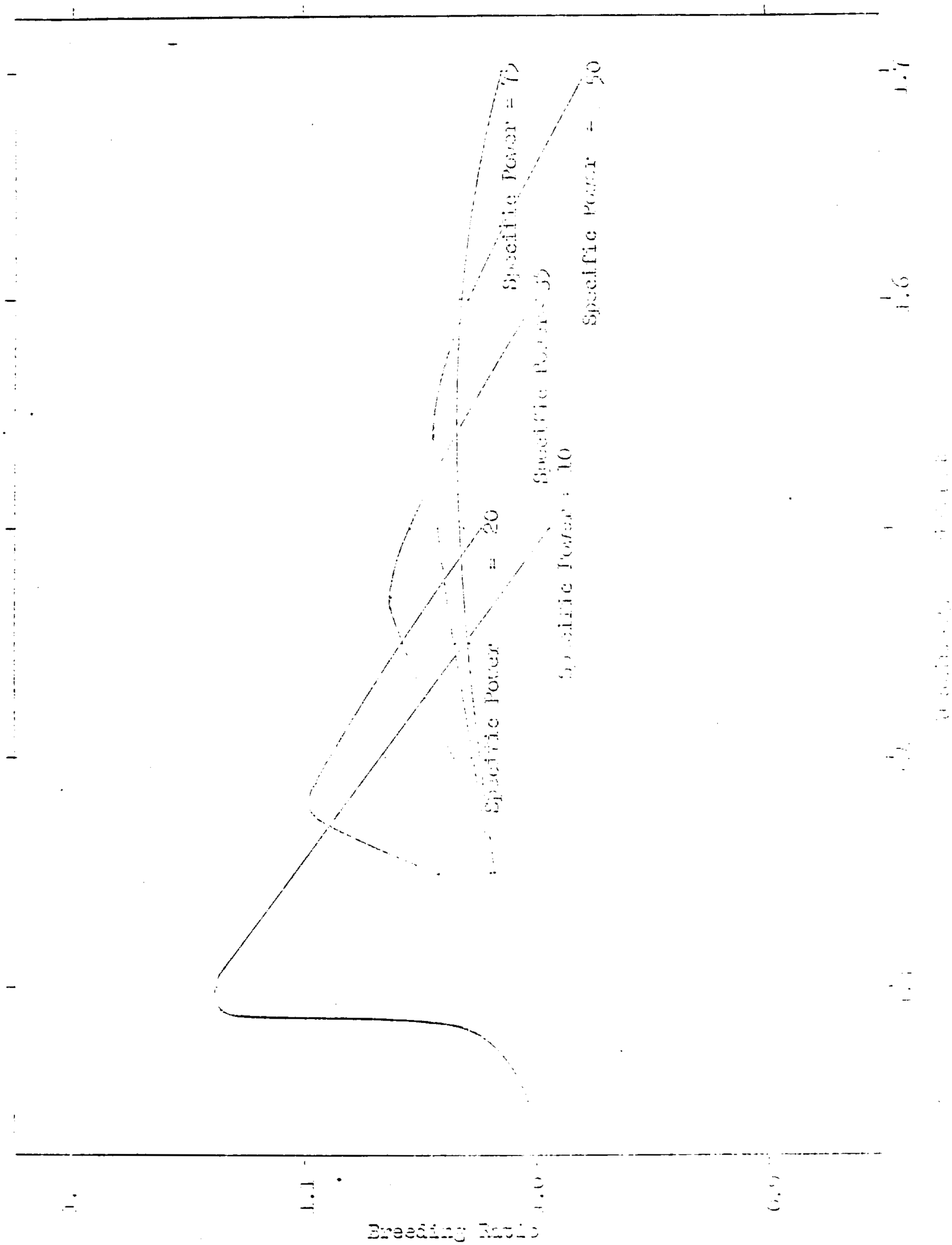
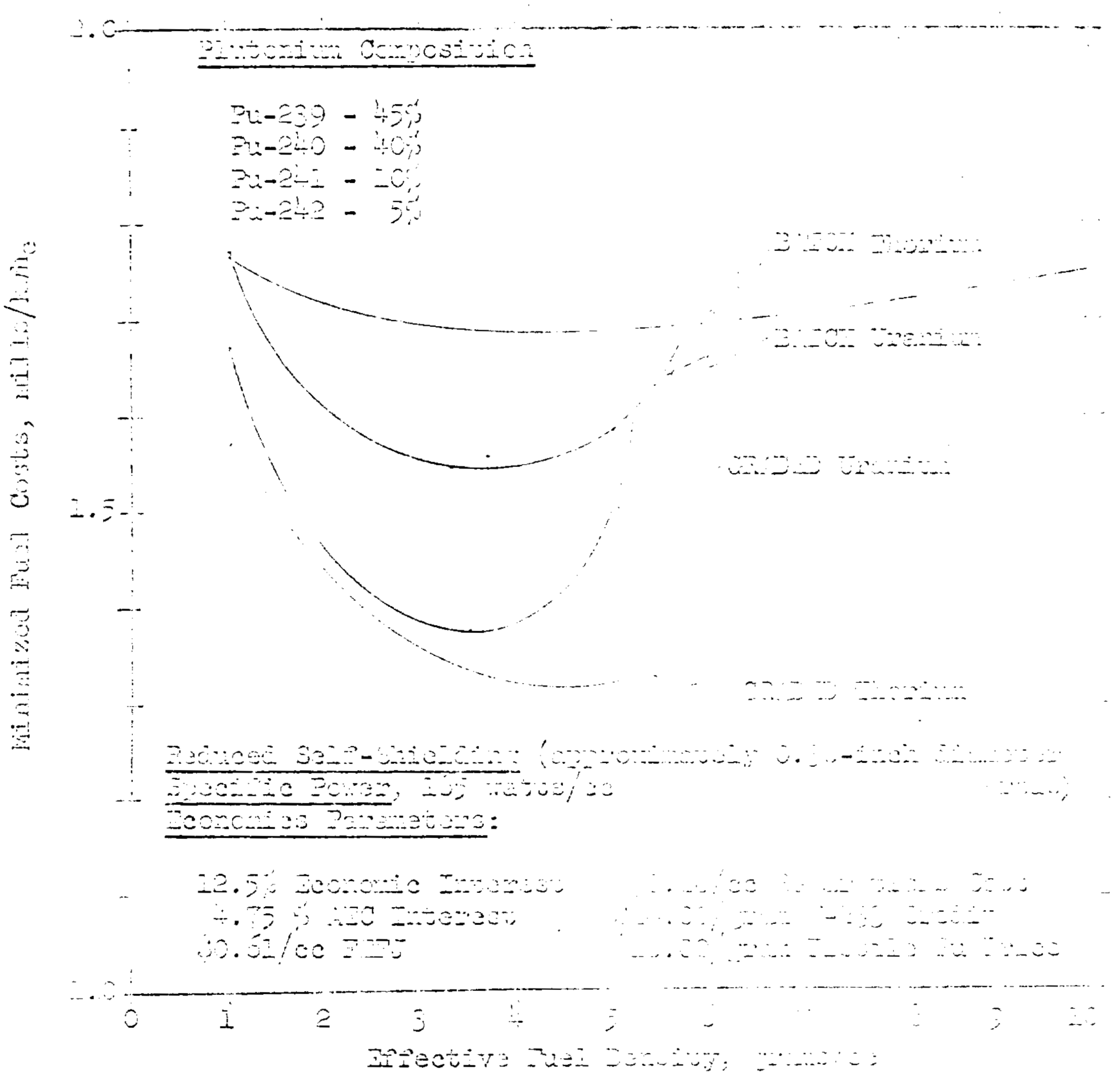


FIGURE 5


2.0
Plutonium Composition
Pu-239 - 45%
Pu-240 - 40%
Pu-241 - 10%
Pu-242 - 5%
Minimized Fuel Costs, mills/kwh
BATCH Uranium
1.5
Reduced Self-shielding
Specific Power,
Economics Parameters:
12.5% Economic Interest
4.75 % AEC Interest
1.0
0
1
2
3
4
5
9
10
Effective Fuel Density,

1228943

the fuel costs are nearly constant from 1 to 10 grams per cubic centimeter effective fuel density.

Other data have shown that when thorium was enriched with uranium-235 or with plutonium of composition 95% Pu-239 and 5% Pu-240, fuel costs increased with a decrease in effective fuel density.

When thorium was enriched with plutonium of composition 65% Pu-239, 25% Pu-240, 8% Pu-241, and 2% Pu-242, fuel costs were essentially flat over the range 5 to 10 grams per cubic centimeter effective fuel density. For cases with normal self-shielding (1/2-inch rods), these results indicate that plutonium-enriched-thorium fuels should offer similar advantages to plutonium-enriched-uranium fuels. In particular, the possibility of a plutonium-enriched-thorium-zirconium metallic fuel appears feasible from the standpoint of fuel costs. Thorium-zirconium fuels can be fabricated over a wide range of zirconium/thorium content ratio and have good high temperature characteristics.

CODE DEVELOPMENT

JASON Fast Effect Test

A test was made to determine the variation in fast effect as a function of rod diameter for the computer code, JASON. JASON calculates the four factors (ϵ, η, p, f)[a] for a unit lattice cell. The cases were 1 percent enriched uranium-235 in uranium-238 UO_2 and uranium metal with oxide density 10.5 grams per cubic centimeter, metal density 18.9 grams per cubic centimeter. Six experiments were made using metal or oxide with D_2O, H_2O, or graphite as moderating material. Volume weighted slowing down powers[b] of 1 and 2 were used. The fast effect, as calculated, was independent of the moderator used and the moderator-to-fuel ratio, so that each curve in Figure 7 represents results for D_2O, graphite, and H_2O moderators at two w/u ratios.

An optional fast effect calculation is available in JASON. Tests are being made to determine how the optional calculation compares with the standard method. The standard method calculates lower epsilon values than are determined experimentally. Results from a single test of the optional fast calculation show it to be much nearer to the published data for the fast effect of a conventional water moderated power reactor.

(a) ϵ = fast fission effect in U-238
η = neutron yield per neutron absorbed in fuel
p = resonance escape probability
f = thermal utilization factor

(b) SDPV = volume weighted slowing down power

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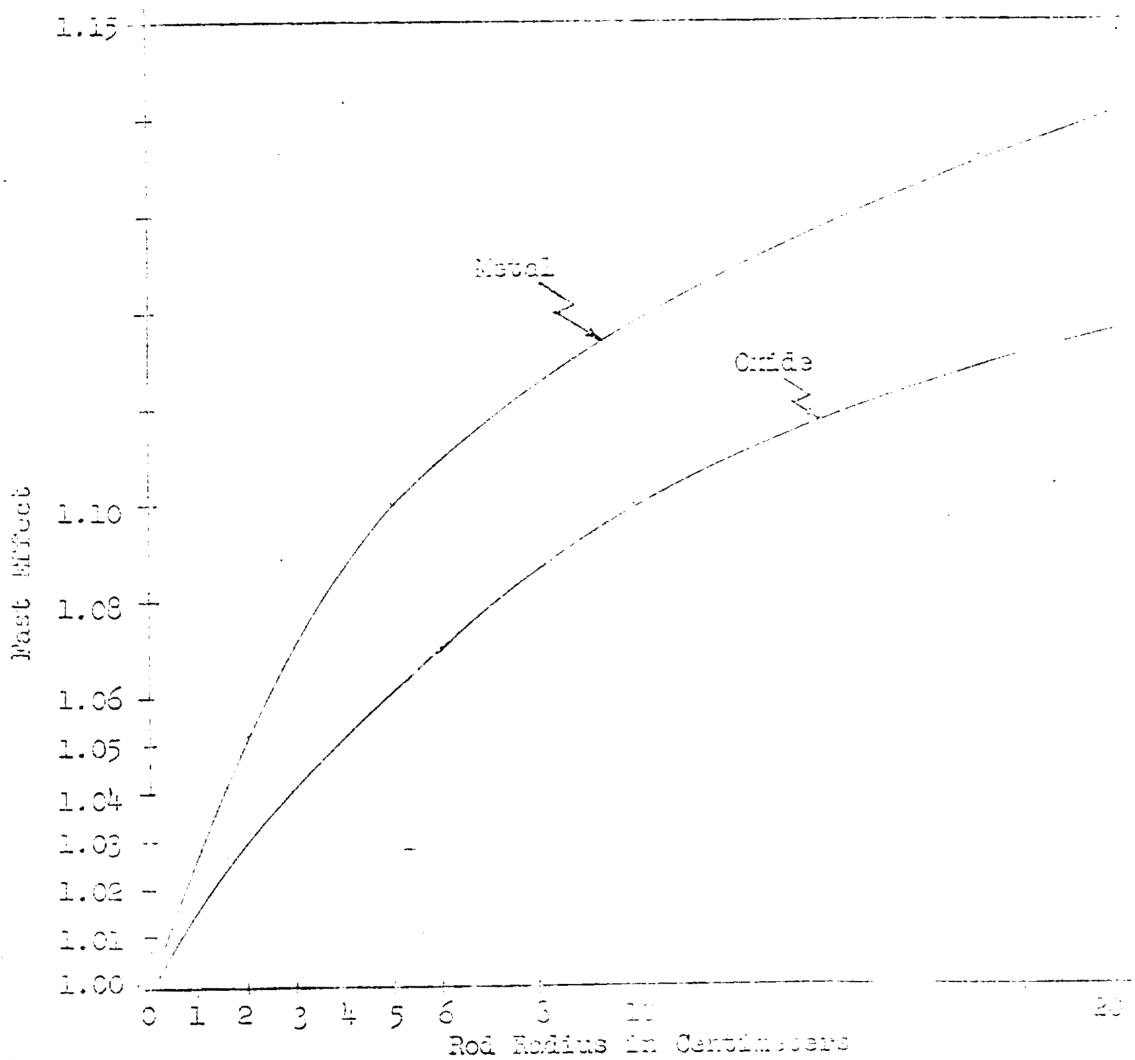

FIGURE 7

FAST EFFECT AS A FUNCTION OF FUEL ROD DIAMETER

FOR THE COMPUTER CODE [illegible]

1228945

Data Preparation Code

A computer code named PERCENT was written that would prepare input data for the JASON CASE GENERATOR code. This code requires minimal data preparation and uses less than 10 seconds computer time to prepare input cases that formerly required two days of hand computer calculations. The code is debugged and completely operable.

Neutron Balance Calculations

Calculated neutron balances were sent to the AEC Planning and Forecasting Branch. These data were generated for four reactor types, each with twelve different fueling modes (uranium-235, plutonium, and uranium-233) for use by the AEC to evaluate the neutron economy of various reactor types. These data are pessimistic to plutonium as the self-shielding of the plutonium-239 resonances is insufficient in the code and indicates the alpha values of plutonium-239 and plutonium-241 are greater than expected in full reactor loads. In addition to neutron balances, summaries of fuel costs and uranium-235 concentration data for these cases were prepared and forwarded. The data are so voluminous that no further details are included in this monthly report.

Nuclear Safety Activities

Approval was granted by the AEC on September 30, 1963 to increase the power level limit for PRCF from 100 watts to 10 kilowatts. The increase is needed to achieve the required precision on period measurements with irradiated fuel without interference from the shutdown neutron flux.

Based on information provided, the Atomic Energy Commission approved operation of the PRTR Fuel Element Rupture Testing Facility in accordance with the proposed temperature and pressure limits and the pressure tube surveillance program as proposed for operation with a standard PRTR pressure tube installed in the test section.

Additional information was provided ACRS and AEC staff members in response to questions on material furnished for the forthcoming review of N-reactor. Preparations were made for the meeting with ACRS scheduled November 7.

[signature] for M Lewis

Acting Manager
Programming

M. Lewis:rm

RADIATION PROTECTION OPERATION
REPORT FOR THE MONTH OF OCTOBER 1963

A. ORGANIZATION AND PERSONNEL

L. C. Rouse terminated on the last day of September to accept a position with the AEC in Denver, Colorado. J. V. Panesko concluded his rotational assignment with Environmental Studies and Evaluation. A. R. Maki transferred from Radiation Monitoring to ES&EO. T. C. Mehas, Engineer, transferred from RMO to Test Reactor and Auxiliaries. D. E. DeFord joined RMO as a Radiation Monitoring Trainee.

B. ACTIVITIES

Occupational Exposure Experience

One new plutonium deposition case was confirmed by routine bioassay analysis during the month. The body burden for this employee, an operator in Fission Products Processing Operation, was estimated to be 1% of the maximum permissible body burden. The MPBB for plutonium (bone as reference) is 0.04 μc. The total number of individuals who have internal plutonium deposition at Hanford is 325 of which 235 are currently employed.

There were nine incidents involving 13 employees that required special bioassay sampling for plutonium analysis this month. The following is a brief description of the more significant incidents.

A CPD pipefitter received a plutonium nitrate contaminated injury on October 4, 1963 at the Weapons Manufacturing Building (234-5). The employee was connecting a pump in Hood 7C when he snagged his finger on a burr from a pipe thread. Examination at the Whole Body Counter showed 1.7×10^{-3} μc present in the wound. After excision, the wound count was 1.2×10^{-4} μc. The employee is not a previous deposition case. The detection limit for the plutonium wound counter is approximately 1×10^{-4} μc of plutonium.

A CPD operator received an injury at the Weapons Manufacturing Building (234-5) on October 10, 1963, while handling a tensile sample container. Examination at the Mobile Wound Counter failed to disclose any contamination.

A CPD operator received nasal contamination ranging from 1300 d/m to 2300 d/m at the Weapons Manufacturing Building (234-5) on October 11, 1963. The contamination occurred when he was sealing off glove ports in Hood 40-DB where leaky or ruptured hood gloves had been found. While performing this type of operation, operating personnel normally wear appropriate respiratory protection. However, this employee failed to wear any respiratory protection thus permitting the high level nasal contamination to occur.

A HL technician received an injury at the PFPP Building (308) on October 15, 1963, while working in a hood. A survey of the employee's surgeons glove showed 1,000 d/m contamination. A piece of stainless steel was found in the wound. Examination at the Plutonium Wound Counter failed to disclose any contamination.

A situation which occurred at the PFPP Building (308) on October 17, 1963, resulted in a spread of plutonium contamination to four minor construction employees and a radiation monitor. A 6' x 6' x 8' green house was built around a hood in Room 124 where the construction personnel were to remove two ovens. This work was being done prior to providing a new filter box and exhaust line to the hood. A panel was removed without any prior cleaning in the hood, thus permitting the spread of contamination in quantities unmeasurable by normal survey equipment. The five personnel leaving the green house were found to have contaminated protective clothing with readings up to 50,000 d/m. Upon removing this clothing, contamination up to 40,000 d/m was spread to the skin of these employees. However, no nasal contamination was detected. The contaminated personnel were sent to PRTR for decontamination since there is no decontamination shower available in the 308 Building.

A CPD operator received skin contamination ranging from 2,000 to 20,000 d/m and nasal contamination ranging from 200 to 250 d/m while preparing to seal a drum filter unit into maintenance hood 170 at the Weapons Manufacturing Building (234-5) on October 25, 1963. Equipment which was contained in double plastic bags had been surveyed and had been found free of contamination prior to this work. Therefore, no respiratory protection was required. However, apparently during the operation, the plastic was ruptured thus permitting the spread of contamination to the employee and to a large portion of the surrounding area.

The total number of plutonium contaminated injuries for the year is 20 with 14 requiring excision. In 1962, there were a total of 15 contaminated injuries with 8 requiring excision.

Also during the month there were two incidents involving five employees that required evaluation for possible intake of radioactive materials other than plutonium. These incidents are summarized below.

Two electricians and a radiation monitor received beta-gamma nasal contamination up to 850 c/m while working on the west crane in the Purex Building (202-A) canyon on October 1, 1963. The employees were wearing respiratory protection during the work on the crane. Therefore, it appeared that the contamination occurred when the employees removed protective clothing and respiratory protection prior to leaving the work area. Examination at the Whole Body Counter revealed a trace of zirconium-niobium-95 in one employee. The results were negative for the other two employees.

Two HL employees received beta-gamma nasal contamination ranging from 400 to 500 c/m and skin contamination ranging from 25 mrads/hr to 50 mrads/hr at the Radiometallurgy Building (327) on October 21, 1963. These employees had been moving irradiated cobalt samples from F cell into the room when the contamination was found. Examination at the Whole Body Counter on October 22, 1963, revealed a trace of chromium-51 in one employee and 1 μc of chromium-51 in the other.

Several other significant incidents occurred during the month.

An IPD operator received an unplanned exposure as measured by the film badge dosimeter of 0.5 rems gamma and 1.2 rems beta during spline removal operations at the 105-B reactor on October 27, 1963. The badge was worn for a period from October 4, 1963, through the date of the incident. The operator used a long pole in an attempt to dislodge a spline can containing an activated spline from the spline coiler apparatus. When this was ineffective, the operator opened the door to the spline coiler apparatus where he found the can to be held up by a broken piece of spline approximately 20 inches in length. He discharged both the spline can and the broken piece to the tank by striking them with a wrench. The dose to the employee's hand was estimated to be approximately 15 rems. This calculation was based on data from the employee's film badge dosimeter, and facts brought out at the IPD investigation. Medical personnel examined the employee's hand. The employee was scheduled for re-examination in 7 to 10 days.

Seventeen IPD exempt employees received exposures ranging from 400 to 900 mrems gamma at 105-KE reactor during October 9 and 10 while attempting to remove a fuel element which became stuck at approximately 1200 on October 9 in a scum gutter. (The scum gutter is located at the surface of the water below the chute at the rear face of the reactor.) An attempt was made initially to remove the element using normal handling tools. At approximately 1900 on October 10, the element was removed using a specially designed snare at the end of a 30-foot extension. Dose rates involved during this work ranged from 20 r/hr at the 18 foot level platform to 5 r/hr at the 38 foot level. The average dose for the 17 employees utilized in this work was approximately 650 mrems gamma.

A false critical radiation alarm occurred at the 306 Building at 1357 on October 19, 1963, as the result of a faulty detector. When steam was turned off to the building in preparation for some maintenance work, the temperature in the building dropped causing detector No. 17 to operate erratically. The alarm signalled for a very short period of time thus only a small number of personnel in the building heard the alarm. The response of those people hearing the alarm appeared to be good. Detector No. 17 was removed from the system and repaired. This is the seventh false critical radiation alarm that has been reported since the first of the year.

Environmental Experience

A thyroid burden of 73 pc of I-131 was measured October 19, 1963, in a 3-1/2 year old boy living at the West Richland farm where maximum I-131 concentrations were measured in milk samples collected after the recent Purex I-131 release. The thyroid burden of this boy's eight year old sister was <30 pc I-131. The dose calculated for the boy ($\sim$4 gram thyroid) is 0.03 rem for the period of September 3 through October 19, 1963. It is probable that this represents the maximum off-site thyroid dose attributable to the Purex I-131 release.

The B plant stack monitor alarmed twice during the morning of October 24. Apparently, both alarms were the result of shorts caused by excessive moisture in the sample head. Source checks of the instrument made later in the day did not show any malfunction of the monitoring equipment. The steam line which heats the sample intake line will be turned on and should eliminate the moisture problem.

Concentrations of fallout materials in the air of the Pacific Northwest averaged 2 pc β/m^3 of air during October, 1963. This is not significantly different from the average value of 3 pc/m^3 noted during September.

One aerial environmental survey was conducted during the month following flight pattern 2-E (Pasco, Ritzville, Ellensberg, Prosser). No anomalies in radiation levels were detected.

A total of 406 biological, produce, and food samples were obtained for radiochemical analyses. They were:

Milk	234 gallons	83 samples
Pasture grass and hay		73 samples
Beef Thyroids		25 samples
Oysters	6 pounds	3 samples
Dried Oysters		19 samples
Fish (fresh fish sold from local groceries)	3 pounds	3 samples
Duck muscle		28 samples
Duck heads		118 samples
Fish		54 samples

About 100,000 gallons of "Vertan", a proprietary chemical cleaning solution, was released to the Columbia River shoreline at N Area on October 2, 1963. This material had been used to clean and de-scale secondary system piping at the N-Reactor. Because of a cap on the end of the effluent pipe, this material could not be discharged to the main river channel as originally planned. The shoreline discharge resulted in unfavorable concentrations in the immediate vicinity of point of release. Heavy foaming occurred from turbulence at the end of the flume, but dissipated within a mile downstream. An observer was present from the Washington State Pollution Commission at the time of the discharge. The remainder of the "Vertan" solution was subsequently discharged through the uncapped outfall line.

Studies and Improvements

The Hanford Laboratories radioactive waste disposal practices were reviewed. The 300-N and "Wye" burial grounds do not appear to be appropriate locations for disposal of the more toxic long-lived radioactive materials such as the plutonium isotopes and strontium-90. The possibility of using CPD burial grounds which already contain large quantities of these materials was discussed with CPD personnel. They were quite willing to work with Hanford Laboratories in establishing methods for using their disposal sites. Subsequent to these discussions a letter was circulated throughout Hanford Laboratories recommending that plutonium and strontium-90 be disposed to 200 Area burial sites.

The mobile whole body counter was operated in 100-F Area during October. Three hundred and eleven (311) examinations were completed representing more than 90 percent of all Biology Laboratory and IPD employees in the area. Ten employees were re-examined because of detected scandium-46. Showers removed up to 50 percent of the scandium indicating external contamination. In all cases, the quantity of radionuclide measured was less than 1 percent of the applicable MPBB. None of these cases will be recorded as internal depositions. Whole body counter operation under

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field conditions was demonstrated to be feasible yielding valid results under conditions of unstable background. Scandium-46 contamination in a shop in F Area was detected and reported to responsible management.

On-plant bioassay sampling was initiated at 234-5 Building, with 59 employees scheduled for sampling during October. A meeting was held with Radiation Monitoring supervision to familiarize them with the area on-plant sampling program and plans for its expansion. Preliminary steps toward establishing on-plant sampling in 231-Z Building were completed. The current status of the on-plant sampling program was documented for internal use with instructions for sample pickup, keypunching of sample results, and scheduling instructions.

A manual containing radiological design criteria was issued for general use at Hanford. Eight manual sections were completed and included in the manual. The eight sections included are: Building and Construction Layout, Shielding Requirements, Glove Boxes, Building Ventilation, Duct Air Samplers, Room Air Samplers, Stack Monitoring, and High Efficiency Filters.

Two IPD log picoammeter criticality alarm systems were tested for response with the Flash X-ray machine at doses of about 10^9 R/hr and irradiation duration of 0.12 microseconds. The first system alarmed readily at dose rate sensitivity settings up to its maximum range of 10^3 R/hr. A similar system using another company's log picoammeter would not alarm when the dose rate sensitivity was set above 50 R/hr. A total of 16 X-ray pulses were used in this test.

A new lot of 4" x 8" H-70 filter paper was received by Central Stores. Field radiation monitoring groups reported that the filters appeared to vary in flow characteristics. Examination of a small sampling of the filter paper indicated that about 10% of the filters required about a 20% greater vacuum in inches of water to obtain the same flow rate. Discussions with the purchasing agent resulted in sending samples of the paper to the manufacturer. Their comments were not yet received by the buyer.

A solid state detector, vacuum chamber system was assembled and tested with various prepared sources of alpha and beta emitting radionuclides for possible applications in air filter counting. No difficulty was encountered in differentiating between Pu-239, Am-241, and Np-237 in a mixed source. The detector used does not have sufficient resolution to differentiate between Pm-147, Bi-207, and Tc-99. The use of this device for examination of air filters with mixtures of alpha emitting radionuclides will be investigated.

Purchase requisition G-590836 for 100 CP meters was prepared. Requisition G-558675 for 25 Technical Associates, Inc., Juno Meters was returned for further "no-substitution" justification.

All 30 Scintillation Portable Poppies were placed into service and the 10 obsolete plant-made portable poppies were removed from service.

The annual radium inventory was resolved with Plant Accountability on September 30, 1963.

Research Studies

Effect of Reactor Effluent on the Quality of Columbia River Water (02)

A study of the effects of reactor effluent on river water quality with emphasis on temperature continued. Dye studies continued, with releases at H and F areas for measurement of time of travel, rate of dispersion, and temperature change in the tagged water masses. The dye tests to date have shown a significant time lag (one to three hours) between mid-stream and shoreline arrival times at the PRTR intake. Equipment for the portable meteorological station was assembled and put on test operation at the 183-KW water plant. The purpose of this test is a comparison of calculated and measured evaporation at a controlled, but relatively humid location.

Mechanisms of Environmental Exposure

Work began on correlation and plotting of I-131 measurements obtained on samples of air, vegetation, and milk collected from the Hanford environs following the Purex I-131 release of September.

C. RELATIONS

Four suggestions were evaluated. Three suggestion awards were made.

Safety meetings were held throughout the Section during the month. There were no security violations.

A two-hour practice session was held for members of the Radiological Emergency Staff. The session included a tour of the new 703 Building Emergency Plotting facilities, discussion on the new telephone system and radio monitor, followed by a practice problem regarding dispersement of airborne material.

D. SIGNIFICANT REPORTS

HW-79326 - "Estimation of Radioactive Contamination Activity by Radiation Survey Instruments" by F. H. Sanders.

HW-76525-9 - "Radiological Status of the Hanford Environs for September, 1963" by R. F. Foster

HW-79485 - "Monthly Report - October 1963, Radiation Monitoring Operation" by A. J. Stevens

HW-79206 - "Radioactive Liquid Waste Disposal for August", by R. H. Wilson.

HW-79073 - "Environmental Effects of a Fuel Element Failure", by R. B. Hall. (Distribution is being withheld pending clarification of classification.)

PERSONNEL DOSIMETRY AND RADIOLOGICAL RECORDS

	October	1963
External Exposure Above Permissible Limits		
Whole Body Penetrating	0	1
Whole Body Skin	0	0
Extremity	1	1
Hanford Pocket Dosimeters		
Dosimeters Processed	6,313	63,839
Hanford Beta-Gamma Film Badge Dosimeters		
Film Processed	9,229	96,494
Results - 100-300 mrads	180	1,706
Results - 300-500 mrads	17	156
Results - Over 500 mrads	6	37
Lost Results	10	235
Average Dose Per Film Packet - mrad (ow)	9.5	7.7
- mr (s)	37.5	30.9
Hanford Neutron Film Badge Dosimeters		
Slow Neutron		
Film Processed	1,624	16,965
Results - 50-100 mrem	3	18
Results - 100-300 mrem	0	2
Results - Over 300 mrem	0	0
Lost Results	5	102
Fast Neutron		
Film Read	472	4,597
Results - 50-100 mrem	8	268
Results - 100-300 mrem	49	487
Results - Over 300 mrem	0	6
Lost Results	5	81
Hand Checks		
Checks Taken - Alpha	37,356	378,759
- Beta-Gamma	56,252	576,944
Skin Contamination		
Plutonium	39	248
Fission Products	24	382
Uranium	0	5
Tritium	0	0
Thorium	0	1

Whole Body Counter

Subject	Number of Examinations: 747-A WBC	1963	Mobile WBC	1963
GE Employees				
Regular	61	600	310	322
Incident Cases	12	144	1	1
Terminations	2	124		1
New Hires	26	491		1
Special Studies	44	476		
Non-Employees				
Children	2	21		
Visitors	2	44		
Environmental Studies	0	19		
	149	1,919	311	325

Bioassay

	Current Reporting Limit	Results Above Reporting Limit: Oct.	1963	Samples Assayed: Oct.	1963
Plutonium	2.2×10^{-8} μc/sample	45	856	544	5,895
Fission Prod.	3.1×10^{-5} μc/sample	20	98	497	5,322
Strontium	3.1×10^{-5} μc/sample	-	45	-	45
Tritium	5.0 c/l	148	1,315	325	2,275
Uranium	0.14 ugm/l	-	-	207	1,581
Special Studies		-	-	30	358

Calibrations

	Number of Units Calibrated: October	1963
Portable Instruments		
CP Meter	1,090	10,642
Juno	240	2,584
GM	590	5,738
Other	188	1,906
Audits	114	1,071
	2,222	21,941
Personnel Meters		
Badge Film	816	8,012
Pencils	75	1,065
Other	312	3,038
	1,203	12,115

1228956

	Number of Units Calibrated	
	October	1963
Miscellaneous Special Services	194	14,249
Total Number of Calibrations	3,619	48,305

AR Keene
Manager
RADIATION PROTECTION

AR Keene:HVL:ald

TECHNICAL INTERCHANGE DATA
RADIATION PROTECTION OPERATION

I. Speeches Given

None

II. Articles Published

An article entitled, "Environmental Radioactive Contamination as a Factor in Nuclear Plant Siting Criteria", by E. C. Watson was published in the IAEA Proceedings of the Symposium on "Siting of Reactors and Nuclear Research Centres".

III. Visits and Visitors

See attached Visits and Visitors form.

IV. Achievements

None

V. Honors and Recognitions

None

VI. Professional Group or Organization Assignments

None

FINANCE AND ADMINISTRATION

ACCOUNTING

Cost Accounting

Estimates requested by RLOO-AEC show the cost to initiate the Containment Test Program in FY 1964 to be $420,000 for construction and $80,000 for research and development. Total estimated construction cost of the facility to be located in the "T" Plant is $870,000.

During October, Hanford Laboratories was allocated an additional $350,000 for 02 Program equipment obligations, $18,000 for 03 Program equipment obligations and $76,000 for capital work order costs. Total FY 1964 funds authorized for these purposes are now:

Equipment Obligations	Authorization
02 Program	$ 600 000
03 Program	60 000
04 Program	278 600
05 Program	100 000
06 Program	125 000
08 Program	25 000
Total	$1 188 600
Capital Work Orders	$ 130 000

A revised allocation of Hanford Laboratories' FY 1964 depreciation accrual to end programs prepared during the month follows:

Research and Development Program		(In thousands)
02		$ 649
03		165
04		2 359
05		148
06		327
08		10
Subtotal		3 658
Special Off-Site Requests		148
Assessments to:	NRD	35
	IPD	138
	CPD	86
	C&AO	249
Total		$4 314

A special accounting code was established for the activity described below:

.3W Irradiation Services to Lewis Research Center - NASA. An authorization of $10,000 was received for the irradiation and analysis of two tungsten-UO_2 cermet fuel plates.

The following program code changes became effective during October:

Codes Established

.WA CPD - Foundry - Proposed R & D (03 Program)
.WB CPD - Foundry - Process Development (03 Program)
.WC CPD - Mechanical Working - Proposed R & D (03 Program)
.WE CPD - Mechanical Working - Process Development (03 Program)
.WF CPD - Metal Cutting - Proposed R & D (03 Program)
.WH CPD - Metal Cutting - Process Development (03 Program)
.WK CPD - Inspection - Process Development (03 Program)
.WL CPD - Component Processing - Process Development (03 Program)
.WM CPD - Auxiliaries & Support - Process Development (03 Program)
.WN CPD - New Processes - Proposed R & D (03 Program)
.WP CPD - Plutonium Technology - R & D (03 Program)
.WR CPD - Plutonium Purification - R & D (03 Program)
.52 IPD - U-233 R & D (02 Program)
.53 IPD - U-233 Fuel Compaction (02 Program)

Codes Canceled

.62 CPD - Weapons R & D (03 Program)

General Accounting

Following is the status of letters requesting AEC concurrence in proposed actions:

AT-241 Addendum No. 2	Hanford Visitors Center	Approved 10-22-63*
AT-313	Professional Research and Teaching Leave	In hands of AEC
AT-316	AEC Monograph on Iodine 131 - L. K. Bustad and J. K. Soldat	In process

*Approved with certain conditions.

The following new and revised OPGs were issued during the month:

1228960

OPG No.	New	Revised	Title
22.1.6		x	Applied Mathematics Operation
22.1.3		x	Reactor and Fuels Laboratory
22.1.1		x	Hanford Laboratories
55.10		x	Employee Location Card
1.15 (pp. 1 & 2)		x	Compliance with Antitrust Laws
7.1		x	Security Requirements - Protection of Property and Information
7.8 (pp. 1-12)		x	Control of Documents Classified Secret and Confidential
66.7		x	Compliance with Antitrust Laws (Cancellation Notice)
22.1.12		x	Test Reactor and Auxiliaries Operation
22.3.1 (pp. 1, 2a, 2b & 2c)		x	Approval Authorizations
22.1.11		x	Programming Operation
2.2		x	Organization Announcement
2.3.10	x		Technical and Business Planning Operation Manager Position Guide
99.4.3		x	Motor Vehicle Operator's Permit
99.4		x	Use and Control of Motor Vehicles
8.8		x	Standards for Maintenance of Plant and Equipment
8.11		x	Control of Materials, Tools, Jigs and Fixtures on Purchase Orders or Contracts
8.15		x	Property of Others at HAPO

Classification activity for the month included the review of 1,142 purchase requisitions, 884 work orders, and 17 appropriation requests for capital or expense determination, compounding, work review, and reimbursability.

Hanford Laboratories' net material investment at October 1, 1963 totaled $23.9 million as detailed below:

		(In thousands)
SS Material		$22 421
Reactor and Other Special Material		1 269
Spare Parts		342
Yttrium		47
Subtotal		24 079
Reserve: Spare Parts	$81	
Yttrium	47	(128)
Net Inventory Investment		$23 951

1228961

The cumulative value of nuclear material consumed in research by Hanford Laboratories during FY 1964 (at October 1, 1963) is $91,520, comprised as follows:

02 Program	$ 327
03 Program	12 698
04 Program	78 495
	$91 520 -1)

(1- Returns from PRTR resulted in a net reduction of $17,442 in nuclear materials consumed in research during the month of September.

Total plant and equipment investment and net value at September 30, 1963, follows:

Area	Asset	Reserve-1)	Net Book
100-B	$ 103 550	$ 16 061	$ 87 489
100-D	2 057 181	605 829	1 451 352
100-F	3 757 988	1 385 666	2 372 322
100-H	18 268	5 722	12 546
100-K	857 998	259 074	598 924
100-N	605	198	407
200-E	1 403 733	279 568	1 124 165
200-W	4 906 863	2 232 114	2 674 749
300	55 529 049	14 974 698	40 554 351
700	415 116	131 275	283 841
1100	36 424	11 720	24 704
Off-site	1 317 567	318 232	999 335
White Bluffs	627 084	203 825	423 259
4R-2)	4 819 701	1 861 929	2 957 772
	$75 851 127	$22 285 911	$53 565 216

(1- Reserve is computed through 12-31-63.

(2- Includes plant and equipment inside the perimeter barricade, excluding the limited areas, uncatalogued equipment and unclassified equipment.

Laboratory Storage Pool activity is summarized as follows:

	Current Month		FY to Date	
	Quantity	Value	Quantity	Value
Beginning Balance	1 712	$791 063	1 480	$ 811 520
Items Received	310	57 474	891	241 416
Items Reclaimed by Custodians	(77)	(4 751)	(122)	(41 341)
Equipment Transfers	(40)	(10 539)	(143)	(49 220)
Items Disposed by PDR	(54)	(6 360)	(140)	(11 290)
Items Disposed by Excess	(59)	(13 581)	(174)	(137 779)
	1 792	$813 306	1 792	$ 813 306-1)

(1- Includes 152 items valued at $114,181 on loan at 10-31-63.

During the month, 102 items valued at $34,279 were loaned and/or transferred in lieu of purchases. A total of 292 items valued at $110,152 has been redirected to useful purposes this fiscal year. Operating costs for September were $1,339 and for FY 1964 were $4,002.

Total value of equipment and material in custody of the Laboratory Storage Pool at October 31, 1963 was $1.7 million, including Reactor and Other Special Materials of $290,984, SS Materials of $163,800 and other materials valued at $423,574.

Action during the month on projects follows:

New Money to Hanford Laboratories

CAH-100	High Temperature Lattice Test Reactor	$ 6 000
CAH-916	Fuels Recycle Pilot Plant-1)	(67 000)

(1- Transfer of Title III to the CPFF-AE.

Physical Completion Notices Issued

*CAH-958 Plutonium Fuels Testing and Evaluation Laboratories, 308 Bldg.
CGH-992 Additional Fuel Loading Equipment, 308 Bldg.
*CAH-995 Air Conditioning Modifications, 309 Bldg.

*AEM Services only.

Construction Completion and Cost Closing Statements Issued

CAH-938 Coolant Systems Development Laboratory, 1706-KE
CAH-963 Geological and Hydrological Wells - FY 1962

The following contracts were processed during the month:

L- 43	City of Richland
CA-411	Dr. David C. England
SA-310	Washington State University
MRO- 67	Fisher Scientific Company
MRO- 68	X-ray Products Corporation
MRO- 70	RCA Service Company

Personnel Accounting

Salary administration statistics as of October 1, 1963 were completed during the month. Statistics obtained from report indicated that 86.9% of Hanford Laboratories' exempt employees are college graduates and that 47 nonexempt employees have college degrees. The following schedule compares the number of college graduates reported for the past seven years.

Graduates	1963	1962	1961	1960	1959	1958	1957
Exempt	682	602	617	590	559	490	415
Nonexempt	47	41	39	35	30	32	37
	729	643	656	625	589	522	452
Highest Degree							
PhD	127	122	109	105	103	104	95
MS	168	137	126	114	107	105	103
BS	434	384	421	406	379	313	254
	729	643	656	625	589	522	452

HAMTC Council required a list of all nonunit employees who have seniority in a bargaining unit together with their seniority date, seniority group, and classification at the time employees transferred from the unit. The list contained 170 employees of Hanford Laboratories.

H. C. Burkepile, Electrician Trainee, died October 27, 1963.

Patent Awards were made to the following employees:

R. H. Moore	HWIR-1393	A Method for Preparing Reactive Absorbents with High Flow Characteristics from Finely Divided Materials.
W. B. Silker	HWIR-1436	The Effectiveness of High Coagulant Feed in Reducing Hanford Reactor Effluent Radioisotope Concentrations.

R. D. Weed HWIR-1585 Decontamination by the Oxidation and Dissolution of Uranium Metal and Its Oxides for Removal from a Reactor System.

Personnel statistics follow:

Employee Changes	Total	Exempt	Nonexempt
Employees at beginning of month	1 761	776	985
Additions and transfers in	39	14	25
Removals and transfers out	14	2	12
Employees on payroll at end of month	1 786	788	998

Overtime Payments During Month	October	September
Exempt	$ 6 239	$ 5 888
Nonexempt	26 627	33 323
Total	$ 32 866	$ 39 211

Gross Payroll Paid During Month		
Exempt	$ 771 974	$ 769 370
Nonexempt	558 299	670 530
Total	$1 330 273	$1 439 900

Participation in Employee Benefit Plans at Month End	October		September	
	Number	Percent	Number	Percent
Pension	1 565	99.4	1 550	99.4
Insurance Plan - Personal	418		413	
- Dependent	1 357	99.9	1 334	99.9
U.S. Savings Bonds				
Stock Bonus Plan	155	39.3	156	39.9
Savings Plan	67	3.7	66	3.8
Savings and Security Plan	1 237	88.7	1 213	88.7
Good Neighbor Fund	1 283	71.8	1 260	71.7

Insurance Claims				
Employee Benefits	Number	Amount	Number	Amount
Life Insurance	-0-	$ -0-	2	$64 160
Weekly Sickness and Accident	9	924	8	641
Comprehensive Medical	57	3 521	45	3 814
Dependent Benefits				
Comprehensive Medical	101	10 057	98	9 832
Total	167	$14 502	153	$78 447

TECHNICAL ADMINISTRATION

Employee Relations

Fourteen nonexempt vacancies were filled; twenty requisitions are unfilled.

Suggestion plan activity included 68 submissions, 28 adoptions and 34 rejections.

AEC approval has been received to refund 100 per cent of tuition costs on graduate school courses effective September 1, 1963.

Information and Presentations

Visitors Center activity:

October attendance	1 488
Average attendance per day open	54
Cumulative attendance since 6-13-62	58 113
Conducted groups	10 (totaling 277 people)

Plant tour activity:

	Number	Total People
General Public Relations Tours	4	145
Special Tours	5	52
Cumulative attendance since 1-1-63 (all tours)	-	2 573

Documented information flow during the month was comprised of 1,714 titles (14,913 copies) received at Hanford and 125 titles (9,778 copies) sent off-site.

The NPR reactor model was installed in the Visitors Center. Approval was received from RLOO-AEC to continue operation of the Center indefinitely (1) provided City of Richland provides free space and janitor service, (2) provided RLOO-AEC approves addition or major improvement of displays, and (3) subject to quarterly reviews of visitor attendance.

Professional Placement

Advanced Degree - Three PhD applicants visited HAPO for employment interviews. One offer was extended; four rejections were received. One offer is currently open.

BS/MS (Direct Placement) - Two offers were extended. One acceptance and no rejections were received. Two offers are currently open.

BS/MS (Program) - Four offers were extended. No acceptances and no rejections were received. There are four open offers.

Technical Graduate Program - One Technical Graduate was placed on permanent assignment. Two new members were added to the roll. Current program members total 76.

FACILITIES ENGINEERING

At month end, Facilities Engineering Operation was responsible for seven active projects, having total authorized funds in the amount of $6,424,500. The total estimated cost of these projects is $10,269,000. Expenditures on them through September 30, 1963 were $1,261,000.

The following summarizes project activity in October:

Number of authorized projects at month end -------------- 7

Number of new projects authorized ---------------------- 0

Number of projects completed ---------------------------- 2
CAH-958, Pu Fuels Testing and Evaluation Laboratories, 308 Building
CGH-992, Additional Fuel Loading Equipment, 308 Bldg.

New projects submitted to the AEC ---------------------- 2
CAH-116, PRTR Decontamination and D_2O Cleanup
CAH-119, PRTR Storage Basin and Experimental Facilities Modifications

New projects awaiting AEC approval --------------------- 2
CAH-116, PRTR Decontamination and D_2O Cleanup
CAH-119, PRTR Storage Basin and Experimental Facilities Modifications

Other project proposals being prepared ----------------- 6
CAH-114, Critical Mass Laboratory Addition
Laboratory Fire Protection System - 300 Area
Heat Transfer Apparatus for Model Studies
Waste Transport System
141-M Building Addition
Geological and Hydrological Wells - FY 1964

The following is the status of active projects:

CAH-916, Fuels Recycle Pilot Plant - Construction is 13 per cent complete compared to a scheduled 12 per cent. Erection of concrete sub-structures is still the major activity at the construction site. Approximately 1,900 cubic yards of concrete have been placed. Piping work is progressing along with the structural work. The design changes to eliminate piping interferences in the storage vault are being processed.

CAH-922, Burst Test Facility for Irradiated Zirconium Tubes - On-site construction is 68 per cent complete and on schedule. Procurement is still lagging. Installation of improper heaters in the five containment vessels will necessitate removal of the containment shells to gain access to the heaters. Following completion of the outer shells by the vendor, the vessels are to be brought to Richland where the heater changes and final welding are to be performed in the J. A. Jones shops. Vessels were due September 3, 1963 and have not as yet been shipped. The instrument panel was due October 24 and is now promised November 15.

CAH-958, Plutonium Fuels Testing and Evaluation Laboratories, 308 Building - Work was physically completed by the directive completion date at a cost of $120,000. The amount authorized by directive was $150,000. This project will no longer be reported.

CAH-962, Low Level Radiochemistry Building - The Architect-Engineer submitted the detailed design drawings for Commission and Company review on September 23, 1963. The drawings were reviewed and comments were transmitted to the Commission on October 15, 1963. The Commission forwarded the information to the Architect-Engineer, who in turn reviewed it with the Commission. The Architect-Engineer's comments are now being analyzed.

CAH-977, Facilities for Radioactive Inhalation Studies - Comments on the detailed design were forwarded to the Architect-Engineer on October 8, 1963. Detailed design is expected to be completed early in November.

CAH-982, Addition to Radionuclide Facilities, 141-C Building - On October 2, 1963, the Architect-Engineer was authorized to perform detailed design of these facilities. The work is approximately 30 per cent complete. The scheduled completion date is November 26, 1963.

CGH-992, Additional Fuel Loading Equipment, 308 Building - The project was physically completed on October 10, 1963, slightly ahead of schedule. The total project cost was $158,897 compared to an authorized cost of $165,000. This project will no longer be reported.

CAH-995, 309 Building Air Conditioning Modifications - The project was physically completed on September 30, 1963, the directive completion date.

CGH-999, Plutonium Recycle Critical Facility Conversion to Light Water - Construction is 60 per cent complete compared to a scheduled 55 per cent. Mock-up of all major mechanical components is nearing completion in the J. A. Jones Company shops. Shutdown of the Plutonium Recycle Critical Facility for installation of new components is scheduled for November 11, 1963.

CAH-100, High Temperature Lattice Test Reactor - The Commission authorized an additional $60,000 interim funds for performance of design. Vitro is preparing a design schedule containing a Title I design completion date of November 15, 1963, and a completion date for detailed design of August 31, 1964.

Selection of suitable materials for structural components, control rods, and other mechanisms has proved to be a difficult problem. Hastelloys and other materials tested under the proposed operating conditions have undergone nitriding sufficiently severe to preclude their use. Further testing is being performed to solve this problem.

CAH-116, PRTR Decontamination and D_2O Cleanup Facilities - Directive number AEC-227 dated October 28, 1963, authorized the Commission $43,500 for performance of design.

Pressure Systems

Investigative work was started on the technology of liquid metals systems in the 2500° to 3500° F range.

Assistance was provided on the waste calcination and helium loop studies in the application of appropriate pressure codes.

The bottle gas station at 3717 building was reviewed and safety improvements are being formulated.

Building 314 was surveyed for location of the pressure bonding autoclave.

Engineering Services

Engineering work performed in support of design and construction on active projects and project proposals and design criteria for new projects included: (1) field liaison, review of shop drawings, and approval of submitted materials on CAH-916, FRPP, (2) review of A-E design on CAH-962, Low Level Radiochemistry Building, (3) review of A-E preliminary design and consultation with A-E on CAH-982, Radionuclide Facilities, (4) scope and design criteria for a fast reactor critical facility, (5) preliminary study of relocation of Biology Laboratory, and (6) preliminary study of a consolidated maintenance shop.

Engineering and consulting work provided to customers included: (1) engineering assistance on experimental neutron spectrometer 105-KE building, (2) engineering of cell door operator at 747 building, (3) study of feasibility of extending future waste disposal railroad spur to 324 building, (4) engineering assistance on high temperature water test loop, 314 building, (5) engineering assistance on installation of 329 building x-ray equipment, (6) operation and engineering of NPR charging machine modifications to utilize magazine loader, (7) purchase of waste flow metering and monitoring equipment, (8) study of relocation of Biology Laboratory to 300 Area, (9) operation and modification of 108-F source handling equipment, (10) engineering and test work for installation of 20,000 psi pressure bonding autoclave, (11) ventilation studies of 251 and 151-H buildings, (12) assistance on set-up for air flow test of fuel section of 314 gas loop, and (13) instrument engineering assistance to the Chemical Processing Department.

Plant Engineering

Plant engineering service was provided on numerous maintenance and laboratory modification and improvement jobs. Major items were: (1) study of 306 building process sewer requirements and engineering for installation of an auxiliary water supply, (2) study of effectiveness of 308 building acoustical treatment and investigation of means to reduce noise transmission from dynapak machines throughout the building, (3) completion of 309 building paging and alarm system, (4) review of 327 building expansion requirements, (5) study of filter changing, 327 building, to reduce radiation exposure, and (6) study of requirements for activation of 5201 building for Laboratories use.

Facilities Operation

Costs for the month of September were $149,344, which is 99 per cent of the forecast for the month. Costs for the first three months of the fiscal year were $441,118, or 107 per cent of the predicted. During this month, improvement maintenance costs continued lower than the forecast with $3,674 spent compared to $15,000 predicted. Building and general maintenance expenditures, however, were both higher than the forecast, as they have been for each month. Consequently, maintenance costs in total exceed forecasts by $32,000 for the first quarter.

Fire drills were conducted on October 11 in Hanford Laboratories' buildings. Although the performance was satisfactory, improvements will be made in certain locations to provide better coverage of the alarm sound.

Criticality alarms were tested on October 29. "Test" signs have been provided at the buildings to inform personnel of the scheduled test. There was one spurious signal on the system in October.

The following table summarizes waste disposal operations:

Item	August	September
Concrete waste barrels		
disposed to: 300-N burial ground	20	8
300 Wye burial ground	0	10
200-W pu burial ground	0	11
Loadluggers of dry waste disposed from 325 Building		
to: 300-N burial ground	2	1
300 Wye burial ground	0	2
200-W pu burial ground	0	1
Loadluggers of dry waste disposed from other 300 Area sites		
to: 300-N burial ground	22	7
300 Wye burial ground	0	13

The "Wye" burial ground, contaminated during a milk pail dumping operation, was cleaned satisfactorily.

There was an increase in the amount of nonroutine decontamination performed by this group.

Activity in the area of building service operation included:

1. Frequent changes of air balance in 308 building necessitated by construction work in this building.

2. All heating and ventilation controls in the 306 and 306-A buildings are being overhauled.

3. New heating and ventilation unit for the mezzanine section of 325 building was completed and placed in service on October 15.

4. New booster coil was installed in the 326 building main system. No. 1, 2 and 3 vortex dampers were rebuilt. Circuitry of basement sump pumps failed on two occasions during the month. New control wiring will be installed and revisions made as soon as new parts are available.

5. All supply exhaust dampers in 327 building were inspected and adjusted on Saturday, October 19.

6. Fan sheave on No. 2 exhaust fan, 329 Building, was found to be cracked at four of the six spokes. Replacement sheave is on order. Meanwhile, the unit is being used only for standby service. New fan bearings were installed in this unit during October. New water meter was installed.

7. Air flow measurements were obtained in 325 basement for Vitro. Approximately 80 hours were required to complete.

The justification for the rail transport of crib waste project was rewritten to reflect recent operating conditions and to respond to suggestions made by AEC.

Drafting

The equivalent of 194 drawings were produced during the month for an average of 22 man-hours per drawing.

Major jobs in progress are: (1) high temperature gas loop, (2) HTLTR mock-up core, (3) powder processing line, (4) inhalation hood as-builts, (5) "C" cell salt cycle process, (6) fast super pressure power reactor concept, (7) HTLTR control rods, (8) thoria processing line, and (9) dynapak furnace hood. Also work was produced in support of engineering reported under previous sections of this report.

Construction Supervision

Activity during the month on construction work (J. A. Jones Company) being performed for Hanford Laboratories components is given below:

	Unexpended Balance	Waste Calcination
Orders outstanding beginning of month	$272 726	$34 538
Issued during the month (including suppl. and adj.)	114 751	51 500
J. A. Jones expenditures during month (includes C.O. costs)	167 943	19 839
Balance at month end	219 534	66 199
Orders closed during month	151 363	

In addition, work on two maintenance work orders issued to plant forces and having a face value of $1,622 was supervised.

SA-289 Crane Safety Inspection - Inspection of cranes and hoists was conducted as listed below:

Building	No. of Pieces of Equipment
321 - 321-A	8
314	4
309	6
328	4
1171	1 (Consulting work for transportation on 45 ton crane)

Copies of inspection reports are being sent to the building custodians for corrective action.

Major nonproject jobs in progress are: (1) precooling system rooms 201 and 202, 108-F, (2) feeder stalls and feeder troughs, 141-C, (3) building modifications and laboratory furniture, 141-H, (4) sawdust silo, 144-R, (5) concrete pad, 144-F, (6) hay storage area, 100-F, (7) construct maintenance shop addition and install tensile test glove box, 231-Z, (8) install drain lines, 271-CR, (9) construct building addition, 292-T, (10) install electrical bus and water filter, 306, (11) provide start-up service for fuel fabrication line, modify lighting, room 108, relocate small dynapak and install new dynapak, and renovate room 125, 308 building, (12) construct block addition for gas loop, install PRTR tube replacement mock-up and install HTLTR mock-up, 314 building, (13) construct roof enclosure, construct maintenance shop, modify room 520 and construct retaining wall 325 building, (14) construct exhaust system, install floor drains and install unit heater, 327 building, (15) install lighting, 3717-B, and (16) fabricate waste calcination equipment for 324 building.

Three requisitions were issued during the month totaling \$350. Total value of equipment being processed in \$66,000.

WSale
Manager
Finance and Administration

W Sale:JVM:whm

REACTOR DEVELOPMENT - 04 PROGRAM

PLUTONIUM RECYCLE PROGRAM

Plutonium Recycle Test Reactor

Operation

Reactor output for October was 1248 MWD, for an experimental time efficiency of 78% and a plant efficiency of 57.5%. There were four operating periods during the month, one of which was terminated for scheduled refueling and planned maintenance, one was terminated manually due to ventilation unbalance, one was terminated by a spurious Log N period trip, and one operating period continued through month-end. A summary of the fuel irradiation program as of October 31, 1963, follows:

	Al-Pu		UO_2		PuO_2-UO_2		Other		Program Totals	
	No.	MWD	No.	MWD	No.	MWD	No.	MWD	No.	MWD
In-Core	18	1636.7	0		67	7063.6			85	8700.3
Maximum		100.9				171.8				
Average		90.9				105.4				
In Basin	25	1895.0	32	3815.0	13	382.0	1	7.3	71	6099.3
Chem. Proces.	32	2309.3	35	1965.8	—	—	—	—	67	4275.1
Prog. Totals	75	5841.0	67	5780.8	80	7445.6	1	7.3	223	19074.7

Note: (MWD/Element) X 20 = MWD/TU for UO_2 and PuO_2-UO_2.

D_2O and indicated helium losses for October were 859 pounds and 114,011 scf., respectively.

A shipment of 19,091 pounds of scrap D_2O was made to Savannah River.

Equipment Experience

A total of 71 reactor outage hours were charged to repair work. Main items were:

D_2O and helium leaks	16 hours
Flash tank repairs	11 hours
Ion exchanger replacement	11 hours
Valves	8 hours
Instrumentation	8 hours

Preventive maintenance required 275 hours or 5.5% of the total maintenance effort.

The automatic controller was operated during the reactor outage for controlling the moderator level under simulated power conditions. Deliberate disturbances were induced, which demonstrated the ability of the controller to respond in a proper and timely manner. No serious problems were detected.

Several modifications were made to the Light Water Injection System (final reactor backup coolant system) increasing its reliability.

Improvement Work Status (significant items)

Work Completed

Filtered Water Pump #3
#2 Exide 125-V Battery Charger
Flash Tank Modification
Platform at -12'
Light Water Injection System Modifications
Helium Compressor Piping Modification for Snubber Installation
Actuator with Hydraulic Snubber for Bottom Blow-Down Line
S-60 Valve Trim Change

Work Partially Completed

In-line Gas Sampling
Process Tubes Level Indicator
Inlet Gas Seal Replacement
Backup Emergency Power to Primary Pumps
Helium Compressor Unloading Orifice Modification
125-Volt Battery Disconnect Contactor
Shim Rod Shroud to Top Cap Modification
Improved RTD Connector Sealant and Bracing
Instrument Power Transfer System
Installation of New Alarm Annunciator
Reactor Automatic Controller

Design Work Completed

Flow Monitor Tubing Snubber Installation
Pressurizer to Stack Valve Relocation
DT-1 Storage Tank Sight Gauge
Galvanometer Shunt Replacement
Indication of DC Solenoid Failure
Emergency Personnel Air Lock Door Operators
PRTR Data Handling System
Holdup Tank - High Level Alarm

Design Work Partially Completed

Additional Fuel Storage and Examination Facility
Vibration Snubbers for Earthquake Protection
Decontamination Building and D_2O Cleanup Facility
Flux Wire Scanning System
Supplemental Emergency Water Addition
Permanent Installation of Closed Circuit TV
Rupture Monitoring System Modifications
PRTR Increased Power Level
Battery Power for Galvanometer Light
Containment Valve By-Pass for Sump Pumpout Lines

Process Engineering and Reactor Physics

PRTR Test Number 34 (Shim Temperature After A Scram) was completed during the month. The maximum temperature reached after a reactor scram was between 250 and 260 F.

A computer program was written and used to generate tables of the constant K_6 as functions of coolant inlet temperature and coolant ΔT, with the coolant isotopic purity as a parameter. The tables are used in the manual calculation of PRTR power.

The deaerator vessel was drained and inspected which revealed that there was a measurable increase in the depth of pits from the previous inspection conducted in April.

Procedures

Operating Procedures Issued	2
Revised Operating Procedures Issued	7
Revised Operating Standards Issued	13
Temporary Deviations to Operating Standards Issued	2
Equipment Standards Issued	2

Drawing As-Built Status:	October	Total
Approved for As-Built	18	1 032
In Drafting	(3)	20
In Approval		14
Deleted or Voided		81
		1 147
Scheduled for Review		333
		1 480

Personnel Training:	Man-Hours
Qualification Subjects	224
Specifications, Standards, Procedures	82
Emergency Procedures	7
Maintenance Procedures	64
	377

Status of Qualified Personnel at Month-End:

Qualified Reactor Engineers	9
Qualified Lead Technicians	6
Qualified Technicians	17

Experimental Reactor Services

The status of the various test elements at the end of October 1963, is shown below. Those elements which had reached their assigned goal exposure or had been permanently discharged for other reasons prior to October 1, 1963, have been deleted from the table.

Test No.	Channel Location	F.E. Number	Description	Date Initial Charge	Date Discharged	Approx. Accumulated MWD	
14	1956	5097	Moxtyl-Swaged	4/2/62	--	105.5	repad
14	1352	5098	Moxtyl-Vipac	5/8/62	--	171.8	repad
14	1758	5099	Moxtyl-Vipac	5/8/62	--	127.0	repad
48	1253	5150	Moxtyl ($\frac{1}{2}$" x $\frac{1}{2}$" pads)	8/1/62	--	117.8	
47	1647	5121	Unautoclaved LX Pu-Al	6/13/63	--	73.5	
47	1653	5194	Unautoclaved LX Pu-Al	7/6/63	--	66.5	
47	1453	5193	Unautoclaved LX Pu-Al	7/6/63	--	64.7	
54	1542	5116	Moxtyl (clip-on pads)	5/8/62	--	122.0	
54	1554	5118	Moxtyl (clip-on pads)	5/8/62	--	169.7	
61	1247	5185	Moxtyl-Physics	5/28/63	--	83.1	
61	Basin	5186	Moxtyl-Physics	5/28/63	10/15/63	77.3	
61	1847	5187	Moxtyl-Physics	5/28/63	--	96.3	
61	1556	5192	Moxtyl-Physics	6/13/63	--	82.3	
67	1144	5119	Moxtyl (Repaired Wire)	10/20/63	--	22.3	
67	1459	5117	Moxtyl (Repaired Wire)	10/20/63	--	64.8	

Visual examination of 14 process tubes showed one tube that had a .020" high blister 8' from the top flange. This tube has been replaced and is being stored in the basin for further nondestructive tests.

Mockup testing of the Gamma Scanner Facility was completed during the month and it was disassembled from the mockup area for basin installation at month-end.

Wire wraps on two fuel elements were repaired and the elements were charged into the reactor under PRTR Test 67.

Four fuel elements were inspected in the basin and five were inspected in the Fuel Element Examination Facility (FEEF), one of which was rejected.

Completed improvement work consisted of the Gamma Scanner Installation and the Autoclave Installation for ZR-2 Fretting Corrosion Studies. Design was completed for the Thermister Probe Installation in the FEEF.

Plutonium Recycle Critical Facility

Operation was routine for the month, involving both irradiated and unirradiated PRTR fuel elements with a D_2O moderator. The facility was shutdown at month-end for scheduled conversion to H_2O moderator.

Three revised Process Specifications were accepted for use.

Fuel Element Rupture Test Facility

Operation

Three operating runs were completed. The final run went eight days without interruption and survived several externally caused disturbances.

Procedures

Operating Standards Issued	6
Process Specifications Accepted for Use	7
Personnel Training:	Man Hours
Operating Procedures	160
Maintenance Procedures	4
	164

Processing of Spent Fuels

Thirty-four UO_2 fuel elements, with irradiations of $<$2000 MWD/T, were shipped to the Redox plant in 200-W. There are 11 more UO_2 fuel elements which were irradiated to $<$2000 MWD/T that are being stored for Applied Physics and Fuel Testing and Analyses. Of the remaining 21 UO_2 fuel elements, 16 will be used to support Applied Physics programs.

TECHNICAL SHOPS OPERATION

Total productive time for the period was 23,275 hours. This includes 13,804 hours performed in Technical Shops, 6,271 hours assigned to J. A. Jones Company, 3,088 hours assigned to offsite vendors, and 112 hours to other project shops. Total shop backlog is 18,315 hours, of which 90% is required in the current month with the remainder distributed over a three-month period. Overtime worked during the month totaled 761 hours or 3.1% of the total available hours. Distribution of time was as follows:

	Man-Hours	% of Total
N Reactor Department	2 876	12.36
Irradiation Processing Department	4 280	18.39
Chemical Processing Department	480	2.06
Hanford Laboratories	15 639	67.19
Hanford Utilities and Purchasing	-	-

LABORATORY MAINTENANCE OPERATION

Total productive time was 18,500 hours of 20,000 hours potentially available. Of the total productive time, 91% was expended in support of Hanford Laboratories components, with the remaining 9% directed toward providing service for other HAPO organizations. Manpower utilization for October was as follows:

A. Shop Work		2 200 hours
B. Maintenance		7 000 hours
1. Preventive Maintenance	2 200 hours	
2. Unscheduled or Emergency Maintenance	1 400 hours	
3. Normal Scheduled Maintenance	3 400 hours	
4. Overtime (included in above figures)	760 hours	
C. R&D Assistance		9 300 hours

WD Richmond

Manager
Test Reactor and Auxiliaries

WD Richmond:bk

INVENTIONS OR DISCOVERIES

All persons engaged in work that might reasonably be expected to result in inventions or discoveries advise that, to the best of their knowledge and belief, no inventions or discoveries were made in the course of their work during the period covered by this report except as listed below. Such persons further advise that, for the period therein covered by this report, notebook records, if any, kept in the course of their work have been examined for possible inventions or discoveries.

INVENTOR	TITLE OF INVENTION OR DISCOVERY
L. A. Bray	Strontium Recovery from Acidified Sludge (HW-79286)

R S Paul

for Manager, Hanford Laboratories

[redacted]

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