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**PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT
NOVEMBER, 1965**

**COMPILED BY
COMPONENT MANAGERS**

DECEMBER, 1965

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RICHLAND, WASHINGTON

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

Operated By

BATTELLE MEMORIAL INSTITUTE

For The

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PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT
NOVEMBER 1965

Compiled by
Department Managers

December, 1965

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By Authority of CG PR 2 (PR24),

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By Jane M. [illegible] 2-16-94.

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Section A

SUMMARY OF ACTIVITIES

REACTOR AND MATERIALS TECHNOLOGY DEPARTMENT

Production Reactor Support

Examination of N-Reactor fuel element failures continued. Detailed examination of No. 7 has not yielded the cause of failure, and preliminary examination of No. 9 did not reveal the reason for failure of this element. Examination of failure No. 9 showed the failure to be the result of fretting corrosion under an outer fuel support. Fretting was caused by a small wire-shaped chip of Zircaloy-2.

Ninety aluminum-canned lithium aluminate target elements were completed and delivered to N-Reactor Department for final assembly in Zircaloy-2 cans. Autoclaving of the Zircaloy-2 canned assemblies resulted in varying degrees of collapse of the elements with severe collapse of the assemblies containing vibrationally compacted particles. Plans are being made to replace the severely damaged elements with assemblies canned in heavier free-standing Zircaloy-2 cladding.

Weld procedures were developed for high quality closure welds on aluminum clad target elements and 150 elements were processed without weld rejects. The final seal of the pressure relief vent hole is accomplished by either TIG welding of an upset pin or magnetic force resistance welding of a pin. A satisfactory joint design for both processes was developed.

Specimens of uranium alloy fuels irradiated to burnups from 2200 to 3400 Mwd/ton are being examined in Radiometallurgy to establish the effect of alloy content and heat treatment on fuel performance. Diameter measurements and visual examination of 60 specimens have been completed.

Studies are in progress to develop a buggy spring support for use with fuel-target assemblies. This work is necessitated by the decreased annulus to be utilized in a 20-tube experimental conproduct test.

Examination of aluminum clad fuel processed by the hot die size process at various stages in the process and with variations in process parameters is proceeding in efforts to reduce or eliminate regions of critical grain growth in the cladding that are considered undesirable. The recrystallization and grain growth behavior of an alternate cladding alloy, HDA-4, was determined.

Tooling was fabricated for investigating the "cold closure" technique for aluminum clad ThO_2 elements in attempts to develop an improved closure for this type fuel.

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Section A

Severe internal pitting has been found on the downstream end of D-Reactor process tubes. Insert removal or redesign has been recommended for immediate remedial action. Galvanic potential measurements are being made to help define pitting parameters.

Studies to further define the pitting attack on N-Reactor nozzles show that erosion-corrosion reactions from high crud concentrations in the coolant to be a significant variable in the loss of metal from carbon steel in high temperature, high velocity water. The threshold for this reaction appears to be around 29 fps at 540°F (280°C). Velocities greater than 29 fps cause erosion-corrosion; those less than 29 fps have little effect.

Examination of hydride levels in the nonirradiated downstream portions of process tubes 3181 KE and 4372 KE revealed extensive hydriding.

Laboratory tests show that Zircaloy-2 coupons can be charged with hydrogen by application of a 3-volt potential.

Three processes for nickel plating 6063 aluminum followed by precipitation heat treatment showed that none gave completely satisfactory adherence of the nickel to the aluminum.

Concentrations of 3% beryllium and 4% uranium in Zircaloy-2 markedly increased the corrosion of Zircaloy-2 exposed to 300 and 360°C, pH-10 (NH₄OH) water.

Hydrogen analyses of two Zircaloy-2 graphite cooling tubes removed from N-Reactor showed no significant increase in hydrogen over hydrogen concentration as charged.

Boiling burnout experiments were resumed using an electrically heated test section representing a proposed coproducer fuel element. The experiments are designed to allow extrapolation of data obtained earlier in experiments for the regular N-Reactor fuel.

Laboratory experiments were completed to investigate the capabilities of I&E fuel elements to operate at higher power densities in K-Reactor. It was found that the use of larger fittings at the outlet of the process tube reduced the susceptibility of the flow to become unstable at higher power levels but less of a flow reduction was required to cause boiling burnout.

Graphite burnout monitors discharged recently from N-Reactor indicate that the oxidation rate is considerably less than the maximum allowable rate.

A total of fifty target elements was delivered to N Fuels Engineering during November. As part of this work, lithium aluminate powder was densified and completely clad in aluminum by the pneumatic impaction process. Measurements of the thermal diffusivity of impacted lithium aluminate between 300 and 1200°K show effects of possible radiative heat transfer above 800°K.

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Section A

PHYSICS AND INSTRUMENTS DEPARTMENT

Reactor Plant Support

An exponential determination of supercell material buckling in separate target supercells is under way in the N-Reactor exponential assembly. A loading has been measured which has a target-to-driver ratio of 1 to 3.

Data from the KVE experiments in the PCTR are being analyzed to determine the infinite medium multiplication factor and thermal utilization of wet and dry KVE fuel in a 7-1/2" graphite lattice. k_{∞} measurements are being made in the PCTR for a graphite lattice containing Pu-Al and Th in alternate cells with an 8" pitch. The dry case is completed, and the wet case is in progress now.

Reactor area field dosimetry measurements are being improved through development of a compact, solid state, personnel dose rate meter.

Conversion of the N-fuel cap-to-clad ultrasonic bond test from the pulse echo to through transmission technique is complete. Engineering specifications and quality control standards have been revised to be compatible with the new test. Completion of this conversion completes the production line tester improvement program. Program results have resulted in a \$50,000 per year reduction in tester operating cost.

Separations Plant Support

The ability of a neutron beam to distinguish the interface between typical aqueous and organic solutions within a stainless steel housing has been experimentally confirmed. A neutron source and detector were positioned on opposite sides of a large, separations process container, and changes in attenuation of the beam were correlated with position of the solution interface.

CHEMISTRY DEPARTMENT

Two N-Reactor irradiated outer fuel tubes were tested for cladding integrity in a steam atmosphere below the melting point of uranium. Oxidation and internal gas pressure appeared to have caused loss of cladding and rupture.

Interpretation of a theoretical study of the behavior of dissolved noble gases released from a water basin indicates that methods could be developed for delaying and controlling the release.

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Section A

An analysis was made of impurity contributions to radiation levels in recycled bismuth. Silver and cobalt appeared to be the most sensitive elements.

The presence of sodium in thorium is believed responsible for the ready sintering at 1000 C of cold-pressed thorium to 87% theoretical density. Stability to hot and cold water is excellent.

A sample of thorium nitrate produced by pilot plant Thorex acid flowsheet was successfully converted to a stable sol.

The relative fission rates in inner and outer H-Reactor fuel elements have been determined.

The study of the effects of complexants on fission product recovery flowsheets has been expanded to include gluconic and citric acids.

Recent work has failed to confirm earlier results indicative of the extraction of polonium from sodium hydroxide solutions.

Both mixer-settler and pulse column experiments were performed on the solvent extraction of polonium using dibutyl carbitol.

Testing of the multi-stage ion exchanger for cerium was continued and produced average decontamination factors of 5×10^2 .

APPLIED MATHEMATICS DEPARTMENT

A response surface design procedure was formulated to obtain experimental data which will form the basis for predicting the absorption of H_2O and CO_2 by each of three different types of thorium, as a function of surface area and relative humidity.

A decision theoretic reliability model was formulated for each of two operating policies. The models incorporate probabilities and costs of "reactor dumps" reflecting the availability of electrical power from three on-line alternate sources.

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Section A

RADIATION PROTECTION DEPARTMENT

Samples of ground beef from a calf grown in the Riverview district of Pasco, and intended for a Zn-65 ingestion experiment were found to have an average Zn-65 concentration of 5.5 pCi/gram. The concentration was remarkable uniform, with a standard deviation of less than 13 percent for the samples analyzed.

A multiple regression calculation of whole body counting and diet questionnaire data on 2600 individuals gave estimates of the Zn-65 body burden contributed by various types of food. A planned modification of the program may alter the final regression coefficients, but the present results indicate that on the average 60 percent of the Zn-65 is obtained from drinking water, and 27 percent is obtained from seafood. Other contributors in order of importance are game birds, Columbia River fish, and fresh meat. Milk consumption continues to give a negative correlation with Zn-65 body burden.

A study of the flow regime in McNary reservoir was begun with temperature traverses above and below the Snake River and above and below McNary Dam. The reasons for extending previous studies are (1) to determine the mixing pattern of the main stream (containing reactor effluents) in McNary reservoir, (2) to determine time of travel through the reservoir, and (3) to establish a reliable continuous monitoring point at the dam.

Fading of Co⁶⁰ irradiated Bausch and Lomb "Low-Z" glass rod dosimeters during a three-month period was negligible. Most variations in glass rod readings are within the accuracy limits imposed by instrument stability. The basic geometry and shielding requirements of a personnel dosimeter badge utilizing "Low-Z" glass blocks as radiation dosimeters were developed.

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Section A

ACTIVITIES

REACTOR AND MATERIALS TECHNOLOGY DEPARTMENT

METALLIC FUEL DEVELOPMENT

N-Reactor Fuel Element Failure

Examination and analysis of the 7th N-Reactor fuel element failure (Tube 2864, 9/3/65) have continued. Metallographic examinations of a suspect support weld, the end cap closure, and an unusual appearing fractured surface of the clad near the cap have not established the site of initial water entry. Several shallow indentations on the fuel side of the inner clad have been noted. These indentations coincide with major zones fuel cracking and outer clad tearing that occurred during rupture. Detailed metallographic examination through one of the indented areas reveals a layer of apparent foreign material that has reacted metallurgically with the clad. Attempts to identify the constituents of this material have not yet been successful.

A cursory examination of the 9th N-Reactor fuel element failure (Tube 2557, 9/26/65) has been made in the Radiometallurgy facility. The rupture occurred at the end of an outer component with considerable damage resulting from fuel fragmentation. There are three major pieces of the component: one piece contains the end cap and was separated from the component by a fracture plane on a 45° angle to the tube axis; the second piece is wedge-shaped, fitting the 45° fracture on the end cap piece and a transverse fracture on the balance of the element; and, the third piece is the balance of the component including the outer supports. The site of initial water entry was not apparent, and the examination is continuing.

N-Reactor fuel failure No. 11 occurred in Tube 1864 on October 25, 1965, in a fuel element that had been irradiated to an exposure of less than 100 Mwd/ton. Basin examination confirmed the failure and revealed a single blister under a support on the outer surface of the outer component. A wire-shaped object appeared to be protruding from under the support. The presence of the wire-shaped object was confirmed in Radiometallurgy, and examination showed that this object had penetrated the clad directly under the outer support by fretting corrosion, thus providing a site for water entry and causing the failure. The wire-shaped object, about 0.2" long, 0.03" wide, and 0.003" thick; was recovered and was identified by gamma spectral analysis as zirconium (probably Zr-2).

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Section A

Target Element Development

Ninety aluminum canned target elements containing lithium aluminate ceramic cores were completed by the Metallurgy Development Section and delivered to N-Reactor Fuels Section on November 19. One-half of the targets contained sintered pellets. The other half contained vibrationally compacted particles obtained by bulk sintering of the ceramic powder, crushing, and then sizing to obtain the desired particle size distribution.

Thirty-four of the sintered pellet target elements, thirty-four of the bulk sintered and vibrationally compacted target elements, and thirty-four lithium aluminate elements fabricated by Ceramic Research and Development Section were assembled in pairs in Zr-2 cans, 24.4 inches long, by the N-Reactor Fuels Section, and these assemblies were autoclaved at 300°C in 1500 psi water for 72 hours. During autoclaving all of the assemblies sustained some degree of collapse of the Zr-2 cladding. The Zr-2 assemblies containing sintered pellets showed ovality from 3 to 5 mils. The assemblies containing vibrationally compacted particles developed sharp ridges in the Zr-2 with evidence of densification of the ceramic during autoclaving; these assemblies are considered unacceptable for irradiation. Plans are being made by the N-Reactor R&E Section to proceed with the irradiation of one column of 17 Zircaloy-2 clad assemblies (34 Al canned targets) of the sintered pellets.

Plans are being made for a replacement column of sintered pellet targets using a heavier Zr-2 cladding which will be free standing under the autoclave and reactor coolant pressures. A sufficient number of sintered pellets are now available for this replacement change.

In addition to the above target elements for irradiation, twenty aluminum canned elements of sintered pellets were fabricated and shipped to the Savannah River Project for test runs through the extraction process.

Target Element-Closure Weld Development

The use of the alternating current tungsten inert arc welding process for making the circumferential closure welds on the aluminum clad target material has proven to be very successful. Approximately 150 pieces were fabricated with no closure weld rejects.

In order to produce the required quality of circumferential welds, it was necessary to provide a pressure relief hole in the second end cap. A very successful method was developed for sealing the hole which produced approximately 150 pieces without one weld reject. The process involves placing an aluminum pin in the vent hole in a boss and upsetting the edge

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Section A

of the hole against the pin. The top of the pin is slightly above the bottom of a "V" countersink in the boss on the cap. After the pin is sealed into the hole, the boss is fused into a hemispherical shape by filling the "V" by melting the top portion of the boss and continuing to melt the remainder of the boss with the alternating current tungsten inert gas shielded arc process. It takes approximately one and one-half seconds to fuse the boss. The process is very sensitive to operator response. A much less sensitive method of making the closure is under development and has reached the point that it appears successful. The boss on the cap has a small straight vent hole. A straight pin the same diameter as the boss and having the end that comes in contact with the boss rounded with a radius equal to one half the diameter of the pin is used. By using a magnetic force resistance welder a fusion zone is produced that is at least equal in length to the wall thickness of the boss. Helium leak checking of 25 consecutive pieces has shown no leaks. Metallographic examination of the sections of 35 seal welds has disclosed that the thickness of the fusion zone is approximately 0.010 inch, and no evidence of unbonded or void areas has been found. The pins on 25 caps were broken off, and all the fractures occurred at the base of the boss on the cap.

Alternate Target Element Closure Systems

Closure of the target element vent hole by resistance welding has been investigated and found to be very satisfactory. The vent hole on previous tests has been closed by resistance welding of a small disc over a hole in a flat cap. This time the caps were modified to have a projection and a bullet nosed stud was welded to it.

Initial development was done on a Sciaky 3-phase 100 KVA welder using a forge impulse at the end of the weld. Although good closures could be produced with caps supported on a base plate, the forge force proved to be too heavy for actual elements and dished the ends or otherwise deformed them. A spring mounted low inertia electrode was used with much lighter pressures and the process was successful. A small number of pieces were welded to demonstrate the process before it was adapted to single phase magnetic force resistance welding machines.

Alternate Uranium Compositions

A NaK capsule irradiation test of 120 specimens of 17 uranium base alloys containing small additions of Zr, Mo, Nb, Al, Si, Fe, P, and C is being conducted to evaluate the irradiation performance of uranium alloys in terms of composition, heat treatment, fabrication history, corrosion behavior, and irradiation history. Opening of the NaK capsules and examination of approximately 60% of the fuel samples irradiated to exposures ranging from 2200 Mwd/ton to 3400 Mwd/ton has been completed. Of the 60 Zr-2 clad samples

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Section A

examined thus far, 17 have narrow longitudinal splits in the cladding. The splits vary in length from about 1/4 inch to the full length of the 3-inch long samples. Diameter measurements have been completed on the unfailed samples and on the less severely split samples. Irradiation temperatures are now being calculated to determine the relative irradiation performance of the fuel alloys. Volume measurements on all undamaged samples and burnup determinations on a selected few are planned.

Recrystallization temperatures and hot hardness measurements are being obtained on unirradiated samples of certain of these alloys. The object of this study is to determine whether or not the irradiation performance can be correlated with the properties of the unirradiated alloys.

N-Fuel Support Development

The design of supports for N-Reactor target elements was undertaken with the goal of providing a tested support design for use in a 20-tube test. In view of the tight schedule it was judged advisable to adapt the buggy-spring concept for this test rather than attempt to develop a new design. Requirements for the supports are nominally the same as for the N-inner supports, except they are to be used in a 0.190 inch annulus. This reduced annulus places a severe forming requirement on the Zircaloy-2 strip to be used for the support.

The initial information needed on this work concerned the fabrication and ductility of 0.025-inch thick Zircaloy strip as compared to the 0.036 inch thick strip now in use. Experimental strip was formed by rolling a small lot of material using a slight modification of the production rolling schedule. Adequate bond ductility was maintained in the finished 0.025 inch sheet. An investigation of tooling and forming schedules was begun, and test fixtures for status and fatigue tests were completed.

Hot Die Size Fuel Cladding

The temperature and forming conditions used in the "hot die size" (HDS) cladding process have resulted in sharp changes in grain size and hardness in some regions of the fuel cladding. Examination has been made of claddings processed with variations in sizing reduction, end bonding temperature, and prior condition of cladding component to determine process conditions necessary to reduce or eliminate the regions of critical grain growth. The work hardening, recrystallization, and grain growth characteristics of an alternate cladding alloy, HDA-4, 10% Si, 1% Ni, 0.6% Fe were determined.

Improved Closures for ThO₂ Elements

The tools necessary for applying the "cold closure" technique to ThO₂ elements have been procured. A 400-ton press and its immediate area have

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Section A

been set up as a ThO₂ control zone in which the closure development will take place. The elements will be loaded with ThO₂ and necessary aluminum caps in production facilities. After closure in the 306 Building, the element will be shipped to 303 Area for sectioning and evaluation.

N-Reactor Meltdown Tests in the IRP

The large amount of radioactivity that was released to the IRP loop and supporting framework during a rupture test with a defected irradiated fuel element has been removed and the loop readied for further testing. The element was defected by induction heating to 1950°F (1065°C) to simulate an N-Reactor loss of coolant accident. Testing in the IRP with high temperature water at 570°F (300°C) caused the bulk of the core to be washed out after about 8 minutes. An entire month was then spent reducing the activity levels in various pieces of loop tubing and fittings, the loop filter, the cell floor and loop framework, cask and dolly, and drain lines coming from the loop. Most of the activity has been reduced to an acceptable level (up to 200 mR/hr) with the exception of the drain lines. These still have readings up to 1 R/hr through 1 ft of concrete and 2 inches of lead. Further attempts to clean the drain lines will be postponed until the entire series of meltdown tests are completed.

Additional meltdown product release tests are planned. Two elements defected by melting have been received during the past month and are scheduled for testing in the IRP.

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Section A.

CORROSION AND WATER QUALITY STUDIES

Downstream Pitting of Aluminum Process Tubes

Severe internal pitting is occurring in a region extending from 1/2 inch to 3 inches from the downstream end of many of the aluminum process tubes in D-Reactor. Sharp, deep pits are located mostly along the tube ribs; the majority of severely pitted tubes are original tubes, but many new tubes are also deeply pitted. The pitted region on these tubes coincides with the location of a Van Stone seal insert (to prevent flange leaks) which restricts flow in this downstream region. Other factors that may contribute to this pitting include: galvanic effects (1100 alloy tube in contact with 356 cast aluminum alloy insert), the water chemistry changes (silicate addition, pH adjustment) that have been tested at D-Reactor, and the frequent use of a diatomaceous earth paste mixture to plug small tube leaks.

It has been recommended that the seal inserts either be removed or re-designed to assure adequate flow in the rear of the tube. Laboratory polarization studies were also suggested, in order to define pitting corrosion parameters.

Preliminary measurements of galvanic potential indicate that the choice of aluminum alloy may influence this pitting. Measurements of relative potentials were made on unetched samples of aluminum, and indicated that 6061 and X-8001 alloy samples were about 10 to 30 mV anodic to 1100 alloy (stainless steel sample was 360 mV cathodic) in pH 6.6 filtered water, but that 6061 alloy samples were 200 mV cathodic to 1100 alloy in de-ionized water (X-8001 aluminum and stainless steel samples remained about 30 mV anodic and 360 mV cathodic, respectively, to 1100 aluminum in deionized water). Galvanic potential and polarization studies of this pitting corrosion are continuing. Samples of the 356 cast aluminum alloy are being obtained for use in these studies.

Hydride Formation in Zircaloy Tubes

Zircaloy-2 samples have been removed after 14 days of exposure in the out-of-reactor TF-20 facility in tests to help determine mechanisms and find control measures for the severe hydriding that has occurred in Zircaloy process tubes in the K-Reactors. Four different pretreatments were used for these samples:

- Etching only,
- Etching plus insertion of aluminum pin, plus scribed defecting,
- Etching plus insertion of magnesium pin, plus scribed defecting,
- Etching plus insertion of zinc pin, plus scribed defecting.

Thin oxide layers (showing interference colors) formed on all of the samples during this exposure. Slight corrosion of the aluminum and zinc pins occurred; magnesium was corroded completely away. These samples are now being processed for hydride measurements.

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Groove and Ledge Corrosion of Aluminum

Aluminum samples are being exposed in the out-of-reactor TF-20 facility to filtered Columbia River water at pH 6.6 (with no inhibitor added) to determine the influence of turbulent high-velocity water on groove corrosion of aluminum, and to determine the relative susceptibility of different alloys to this type of corrosion. Flow constricting weir pieces are used to provide turbulent flow patterns in this facility. Initial samples removed after 14 days of exposure show film patterns corresponding to expected flow patterns. Aluminum Alloy 6063 samples show a slight erosion pattern; X-8001 and 1100 alloy samples show no apparent surface effect of flow after this initial exposure.

Pitting of Carbon Steel in Filtered Water

Pitting corrosion of the N-Reactor carbon steel nozzles has been observed. A two-week test was made in TF-8 (an out-of-reactor loop) to further study the severe erosion-corrosion attack recently found on carbon steel in flowing water containing high concentrations of iron oxides (crud). The extreme erosion-corrosion reported in October was reproduced with crud levels of 0.8 to 16 ppm in 540°m (280°C) water moving past test specimens with velocities of 1, 8, 29, 43 and 58 fps. The edges of the upstream coupon in each row of coupons at velocities of 29 fps and greater were eroded to knife edges. Increased velocity increased erosion-corrosion. Velocities of 1, 8, 29, 43 and 58 fps led to upstream coupon weight losses corresponding to 0.37, 0.39, 26, 29 and 40 mils, respectively. Penetrations rapidly decreased with distance away from the upstream sample. One row of specimens, for example, showed weight losses corresponding to penetrations of 28, 15, 17, 0.5, 0.08 and 0.4 mils.

Storage Basin Studies

A laboratory study of the corrosion and precipitation behavior of borax solutions in storage basin water has been completed. This study was conducted to determine the source of flocculent material during the tests to determine the compatibility of 17.6 g/l borax with N-Reactor storage basin materials. The corrosion penetrations of 6061 aluminum, Zircaloy-2, 304, 410, 440-C and 416 stainless steels, lead Oilite bronze, commercial brass, Admiralty metal, A245 carbon steel and weathered concrete were all less than 0.02 mil after 22 days of exposure in filtered Columbia River water treated with borax. The penetration of copper was 0.11 mil. The borax significantly reduced the corrosion of carbon steel as compared to corrosion in filtered water without borax.

Additions of 20 ml concentrated ammonium hydroxide/liter to the borax solutions accelerated the corrosion of copper alloys. Small amounts (0.028 ppm) of ammonium hydroxide in the borax solution did not affect the copper alloy corrosion.

Extensive flocculent material was formed by a reaction of the borax with concrete. Green concrete forms voluminous flocculent material, weathered concrete only small amounts. Commercial borax reacts more aggressively with the concrete than chemically pure borax.

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Section A

Examination of K Reactor Process Tubes

Examination of hydride levels in the nonirradiated downstream portions of process tubes 3181 KE and 4372 KE has been completed. Process tube 3181 KE was in service for 2 years, process tube 4372 KE for only 5-1/2 months. Process tube 4372 KE was autoclaved prior to reactor service. The downstream five foot portion of both process tubes was sampled for hydrogen at approximately 1 ft intervals. Half inch wide rings were used for hydrogen analyses on tube 4372 KE. On tube 3181 KE four punchings around half the circumference of the tube were cut and analyzed for hydrogen at each 1 ft interval.

Samples from the examined sections of process tube 3181 KE were heavily hydrided. The total hydrogen content diminished as the distance from the rear flange increased. The total hydrogen concentration at 6 in. was 1160 ppm as compared to 260 ppm 60 in. from the rear flange. The hydride concentration appeared to be uniform around the circumference of the tube.

Hydrogen analyses of the tube sections after etching away the hydride case show that the base metal was also hydrided to some extent. A hydrogen concentration of 91 ppm was found at 6 in. This concentration diminished to 45 ppm at 60 in.

Metallographic sections along the length of the process tube confirm the hydrogen analyses. The hydride case on the ID surface of the tube is approximately 3.5 mils thick at 6 in. from the flange compared to 1 mil at 60 in.

Total hydrogen concentrations for tube 4372 KE decreased from 65 ppm 6 in. from the rear flange to 26 ppm at 59 in. Hydrogen analysis at the upstream end of the tube was 22 ppm. Metallographic sections of this process tube show the hydriding to be very spotty and associated mainly with scratches through the original autoclave oxide layer. Spots of hydride that cannot be associated with apparent scratches may be associated with defects in the oxide.

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Section A

Zircaloy Tube Hydriding Studies

Continuation of the studies reported last month on charging Zircaloy-2 with hydrogen in reactor process water at 90°C by 3 volt (rms) AC potentials showed the metal did pick up hydrogen. Although analyses of the coupons at 283 hr. showed no significant amount of pickup, continuing the test to 667 hr. produced hydrogen absorption by 5 of the 7 coupons in test.

A study on the effect of various water qualities on the hydriding of Zr-2 with galvanic, i.e. coupled to 6063 aluminum, and applied potentials, 3 volts DC, has yielded preliminary results. The refreshed electrolyte was compounded from deionized steam condensate with addition of sodium bicarbonate, pH adjusted to 6.6 with either H₂SO₄ or CO₂. Another electrolyte was the same as the above except some of the bicarbonate was replaced with arsenate to produce 1 ppm As. The electrolytes had a specific resistivity at 90°C similar to process water, 2500 ohm-cm, at 90°C. The 30 mil Zircaloy coupon in each electrolyte made 3 volts cathodic to a platinum electrode for 211 hr. at 90°C, picked up 33 ppm hydrogen each. The current efficiency was about 0.3%. Coupons galvanically coupled to 6063 aluminum in the three different electrolytes showed substantial differences in galvanic current and hydrogen pickup. The solution with 1 ppm As was the highest current and the highest pickup (17 ppm). The solution neutralized with H₂SO₄ produced less current flow and less pickup (9 ppm) while the solution neutralized with CO₂ had the smallest current and no apparent pickup. The test is continuing.

Nickel Plated Aluminum

Three processes for nickel plating 6063 aluminum followed by precipitation heat treatment of the aluminum to obtain the T-6 condition were investigated. None of the methods has given completely satisfactory adherence of the nickel to the aluminum. Tensile testing of specimens prepared by the three processes showed that the tensile strength obtained was approximately that of 6063 aluminum in the T-6 condition when specimens were quenched in cold water after solution heat treatment. Somewhat lower values were obtained in specimens quenched in boiling water.

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Effect of Beryllium and Uranium on the Corrosion of Zircaloy-2

It is known that the brazed and welded end closures on Zircaloy-2 clad fuel elements may contain varying amounts of beryllium and uranium contamination. The effect of these contaminants on the corrosion characteristics and integrity of the end closures is under investigation. A test has been designed to investigate these effects under both in-reactor and out-of-reactor conditions.

Triplicate coupons were cut from Zircaloy-2 castings fabricated by Oregon Metallurgy Corp. and containing various amounts of beryllium and uranium. The Zircaloy-2 castings contained 1, 2, and 3% beryllium and 0.1, 0.5, 1, 2, and 4% uranium for a total of 15 ternary alloys. Each ternary alloy was cast in triplicate. Those coupons cut from each button were bright etched in a standard $\text{HNO}_3\text{-HF}$ bath and divided into three groups. One group was exposed to 300°C, pH-10 (NH_4OH) water; the second group to 360°C, pH-10 (NH_4OH) water. The third group is being held for in-reactor exposure.

Following two weeks in 360°C water and 4 weeks in 300°C water, an increase in the corrosion of Zircaloy-2 with increasing beryllium and uranium concentrations is apparent. Of the two contaminants, uranium has the most pronounced effect on the corrosion of Zircaloy-2. The corrosion weight gains for two weeks exposure to both 300 and 360°C, pH-10 water show a factor of at least 8 increase for the 4% uranium samples and a factor of at least 6 increase for the 2% uranium samples over the normal corrosion of Zircaloy-2. Normal corrosion weight gains for Zircaloy-2 were 10 mg/dm² and 20 mg/dm² for two week exposures in 300 and 360°C water, respectively. The actual increase in corrosion is probably greater than a factor of 6 to 8 increase. All of these samples are spalling, both in the form of powder and flakes. This is exemplified by the 3% Be-4% U samples which had gained 89.7 mg/dm² after two weeks exposure but had a net weight loss of 114 mg/dm² after four weeks exposure to 300°C water.

The exposure temperatures and water conditions used in this test are comparable to those in NPR. The results obtained in this test are probably representative of irradiated fuel end closures similarly contaminated with beryllium and uranium. These autoclave tests will be continued in order to determine whether the more dilute ternary alloys will slough on longer exposure.

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Examination of Zircaloy-2 Graphite Cooling Tube

Two Zircaloy-2 graphite cooling tubes were removed from N Reactor service for hydrogen analysis. Four 3 in. sections of tubing were cut from tube 0788 (2 upstream and 2 downstream), and two 3 in. sections from the downstream portion of tube 0899. Following crud removal, the sections were analyzed for hydrogen. Neither Zircaloy-2 tubes had increased significantly in hydrogen concentration over the preservice hydrogen concentration. The analyses for tube 0788 were from 10 and 12 ppm for the downstream portion and 12 and 15 ppm for the upstream portion. The two samples from tube 0899 analyzed 18 ppm. Analyses of similar Zircaloy-2 tube sections which have never been in reactor service have shown hydrogen concentrations ranging from 12 ppm to 25 ppm.

The crud levels measured on these two tubes were 19.8 mg/dm^2 for 0788 and 12.0 for 0899.

THERMAL HYDRAULICS STUDIES

Boiling Burnout Experiments for N-Reactor

Burnout experiments for N-Reactor co-producer fuel elements were resumed. The proposed co-producer element consists of a nonpower-producing cylinder of target material inside a fuel tube which in turn is centered inside of a pressure tube. Currently, the burnout experiments are being conducted with a model of the flow annulus between the target and fuel tube, consisting of an unheated tube inside an electrically heated outer tube. In this model, the dimensions of the coolant annulus are 1.775 in. OD and 1.220 in. ID which are not the same as the planned co-producer fuel dimensions but which are identical to those used in earlier experiments with both tubes heated. In later experiments, a model with the actual co-producer fuel dimensions will be used. The primary purpose of the present model experiments is to bridge the gap between the earlier two-heated surface model experiments and the experiments with the one-heated surface model having the co-producer dimensions.

These experiments will allow separation of the effects of the two variables - dimension changes and heated vs. unheated opposing surface. This information will be useful in estimating the effects of changes in co-producer dimensions which might be proposed in the future.

Experiments with the current model were interrupted earlier by physical burnout of the model after 14 sets of burnout conditions had been determined. These 14 sets did not provide enough information for establishment of good quantitative relationships. Consequently, the model was repaired and reinstalled in the heat transfer facilities of the Thermal Hydraulics Laboratory. Zero power runs at 16 combinations of coolant flow rates and temperatures have been completed with the repaired model. Data from these zero power runs will be used to determine instrument accuracies and heat losses for use in analyzing burnout experiment results. Experiments to investigate 14 more sets of burnout conditions have been scheduled and are ready to begin.

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Heat Transfer Experiments for K-Reactor

Laboratory experiments were completed for the first phase of a laboratory investigation of the capabilities of I&E fuel elements to operate at higher power densities in K Reactors. The objectives of this phase were to determine boiling burnout data and the pressure drop-flow characteristics of a full length fuel column positioned concentrically within a process tube at tube powers from 1000 to 3300 kw. The experiments were also meant to compare data with outlet fittings of the type in current use (standard outlet fittings) with data obtained with outlet fittings modified to give a minimum of hydraulic resistance (open outlet fittings).

The test section consisted of an electrically heated rod 0.470 in. ID, 1.512 in. OD, and 28.1 ft long. The length of the rod was made up of four different metal alloys to provide an approximation of a cosine power generation along the length with a peak-to-average specific power of 1.40. The rod was positioned inside of a 1.724 in. ID Zircaloy-2 process tube with short ceramic pieces which represented the "arch rail" type of supports in current use. The 0.090 in. height of these supports allows some clearance between the fuel and the process tube and thus the supports do not position the fuel pieces quite concentrically in the tube.

Two hundred thirty data points were obtained in the experimental program. A typical run consisted of maintaining a constant power generation while the flow was reduced in small steps from an initial condition where the exit water was subcooled to a final condition where film boiling existed at some point on the electrically heated test section. The onset of film boiling (referred to as boiling burnout in these tests) was detected by a rapid excursion of one of the surface thermocouples of at least 100°F (55°C). During each run, data were taken of flow, power generation, pressures, pressure drop, surface temperatures and water temperatures. The water temperature data included readings in the annulus at eight points around the circumference and three different locations in the downstream half of the test section.

A preliminary examination of the data obtained with the standard outlet fittings indicates that boiling burnout occurred when the flow rate was reduced to 47% of the flow required to produce a normal outlet temperature of 130°C for all tube powers between 1500 and 3000 kw. These flows are 6 to 8% higher than the burnout flow rates determined in previous tests with the rod supported by "bridge rail" supports. In addition, it was found that the amount of circumferential mixing as determined with the present test section using "arch rail" supports was not as great as the mixing found previously with "bridge rail" supports.

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When the test section was equipped with the open rear fittings, it was found that the flow reduction required to cause boiling burnout was less than that when the test section was equipped with the standard outlet fittings. The probable reason for this is that the more restrictive flow passage in the standard outlet fittings caused pressurization of the test section during two-phase flow. This would restrict the amount of steam formation over the heated surface and require a lower flow rate to cause burnout. On the other hand, the pressurization which took place in the standard outlet fittings makes that case much more susceptible to hydraulic instability than the test section equipped with open outlet fittings. In fact, it appears that instability could not occur in a tube using the open outlet fittings at tube powers below 3000 kw.

At the conclusion of this phase of the study, the test section was removed and the supports modified to allow the rod to be 54% eccentric within the process tube.

GRAPHITE STUDIES

Burnout Monitoring - N Reactor

The second burnout monitoring charge in 1648-N was removed on 10-8-65 after 142 operating days in the reactor. The majority of the monitors showed burnout rates below 0.10% per thousand operating days (%/KOD). The maximum burnout observed (one sample) was 0.20%/KOD. There was an apparent peak at ~5 ft from the reflector front face, and the burnout rate decreased thereafter with distance down the channel.

There were a total of 126 burnout monitoring samples in the channel (18 boats x 7 samples per boat). The monitors were all of TSX graphite 0.426 in. in diameter and of 3, 4, or 6 in. lengths. The lengths of approximately half of the samples were measured before charging into the reactor. The average length change on these samples during the operating period was -0.01%, with ~90 per cent of the samples falling in the range from 0.00 to -0.02%. Exposure information will be available after flux-monitor activities are determined.

Graphite Contraction

Tests have indicated that large bars used in reactor construction contract at a significantly greater rate than the small samples used in irradiation tests. A pertinent question is whether the ratio of contraction rates continues to all exposures, or whether the reversal of contraction (turn-around) which has been observed for the small transverse test samples occurs at a lower exposure for the large bars with the total amount of contraction being a fixed quantity. It is now possible to provide a partial answer to this question.

Central depression of the K Reactors has exceeded 7 in. without indication of a reduction in depression rate. This represents a contraction of the TSGBF central filler bars of more than 2.0% at an exposure of approximately

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4.5×10^{21} nvt ($E > 0.18$ MeV). In comparison, the maximum contraction at turn-around for small CSF transverse samples at 600°C is about 1.3% at an exposure approaching 10^{22} nvt. From the limited data available for TSGBF transverse samples, it appears that turn-around in TSGBF may occur at much lower exposures than for CSF, but the maximum contraction at turn-around may be the same or slightly less than for CSF. Thus, present data indicate that the ratio of contraction rates for large bars and small samples prior to turn-around will apply to the maximum amount of contraction at turn-around.

LiAl₂O₃ FUEL ENCAPSULATION

LiAl₂O₃ powder was densified and completely clad in aluminum in a single impaction by the pneumatic impaction process. This material was densified to a density 90% of theoretical and simultaneously fully encapsulated in aluminum. End closures were made by pressed aluminum powder bonded in position by the impaction. Excess aluminum was removed by machining, leaving the specified clad thickness. A high degree of dimensional control was maintained with this high energy rate forming technique.

A production test load was fabricated, which included a total of 18 fully clad pieces, assembled in six fuel elements approximately 12 inches long. A short flow test piece was also assembled.

Components were assembled in standard 2.5-inch thin wall stainless steel containers. Aluminum molds were machined to fit inside these containers. Fuel was pressed into the bore and capped with pressed aluminum caps. These molds were machined with a taper reducing the outside diameter at the top of the container. This taper compensated for the natural tendency of the core material to "neck down" or taper toward the top. Experience indicates that this tapering of the material is lineal and proportional in severity to the length. Several experimental shots would be required to eliminate this condition. However, in this series, X-rays and physical measurement indicates the core material was tapered less than .020 inches for the longer pieces.

The internal diameter of the aluminum molds was bored oversize, and a nominal reduction of .035 inches was noted. This radial compaction and the reduction in length of approximately .7 inches resulted in a compacted density 90% of theoretical.

The cylindrical cavity in the aluminum mold was loaded with LiAl₂O₃ powder and pressed to approximately 70% density. A layer of aluminum powder was pressed on top forming a cap closure. Conventional welding is not needed, and no interface can be found in the bonding region between the cladding and end cap.

The thin wall stainless steel container and can were heated to 550°C , while a vacuum was applied, and after 45 minutes at temperature, pneumatically impacted.

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As a final step, the aluminum cans were machined to diameter and end cans were faced. Final inspections included X-ray of all pieces and helium leak testing. All pieces, except one, were leakproof. This one piece was questionable, very likely caused from a surface entrainment or contamination at the interface of the closure.

The success achieved on this quantity of fully encapsulated fuels indicates a feasible production process.

THERMAL CONDUCTIVITY OF LiAlO_2

The thermal conductivity of LiAlO_2 (pneumatically impacted at 770°K) has been determined between 300 and 1200°K using a transient thermal diffusivity technique. The aluminate has a relatively low thermal conductivity varying from .0035 watts/cm- $^\circ\text{K}$ at 300°K to 0.018 watts/cm- $^\circ\text{K}$ at 1200°K . The data show a deviation from a $1/T$ relationship above 800°K with an enhanced thermal conductivity, presumably resulting from radiative heat transfer. A slight hysteresis in thermal conductivity at the lower temperatures occurred during the heating cycles. This may be caused by thermally induced phase changes in the aluminate.

X-ray diffraction and Debye-Scherrer powder patterns of the impacted LiAlO_2 indicate the presence of low temperature α - LiAlO_2 and high-temperature γ - LiAlO_2 in almost equal amounts. A detectable amount of γ - Al_2O_3 was also present. The patterns corresponded favorably with those of the as-received powder. Although the thermal cycle to 900°C during thermal conductivity measurements did not appear to alter the relative amounts of α and γ - LiAlO_2 , at least one of the unindexed lines disappeared after the heating cycle. A LiAlO_2 sample impacted after a vacuum heat treatment at 1200°C showed two phases, α - LiAlO_2 and a small amount of LiAl_5O_8 . Heating LiAlO_2 to 1200°C in a vacuum obviously results in preferential volatilization of Li_2O resulting in precipitation of the LiAl_5O_8 phase.

CERAMIC TARGET ELEMENTS

A total of 50, weighed and measured, target elements was delivered to N-Fuels Engineering during November. Twenty-one of the target elements were fabricated by pneumatic impaction - vibrational compaction, 22 by particle agglomeration - vibrational compaction, and seven by direct encapsulation. One of the target elements from each of the above groups was fabricated to an approximate six-inch length and the balance to approximately 12-inch lengths.

The six-inch targets are for destructive testing in hot water flow studies and the remainder for in-reactor and ex-reactor testing.

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Section A

PHYSICS AND INSTRUMENTS DEPARTMENT

REACTORS

(This section is issued separately as a classified document.)

N-REACTOR PLANT SUPPORT

Separate Target Exponential Experiments for N-Reactor

An exponential determination of supercell material buckling is under way in the N-Reactor exponential assembly. The loading that has been measured has a target-to-driver ratio of 1 to 3 using standard 0.948 w/o enriched driver clad in aluminum. Since there are only four supercells vertically in which useful measurements may be made, it is necessary statistically to obtain more information by taking additional horizontal measurements. This requires an analysis of counting data in two dimensions.

In attempting to do a detailed two-dimensional analysis using computer code ICEDT, program errors were discovered which have now been corrected. The results of three horizontal traverses which have now run correctly may be combined by hand with another point, and this should result in a good value for material buckling for the supercell system.

In addition to the present drivers and targets, bismuth and lithium-aluminum targets in a new geometry are available. Attempts are also being made to secure thorium targets and the most recent tube and tube geometry drivers.

System Simulations

Development of the N-Reactor secondary system simulation having parallel steam generator and parallel dump condenser models was completed. The parallel models are required to study transients in one of several units in use. External control loops are included for each model.

Use of a greatly simplified steam generator model allows study of two steam generators with only two-thirds of the computer equipment previously used for one steam generator model. The extra computer equipment thereby made available, plus other simplifications made to the export steam and dump condenser models, accommodates the second dump condenser model.

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N-REACTOR FUELS PLANT SUPPORTComponent Tests

Fabrication of the ultrasonic tester, which will be used as part of the receiving inspection program for Zircaloy components, is essentially complete. The electronic portion of the tester is complete. All of the mechanical parts have been fabricated and delivered to N Fuels personnel for assembly in the UT-8 tank.

Cap-to-Clad Bond Test

Conversion of the cap-to-clad bond test from pulse echo to through transmission is complete. Engineering specifications and quality control standards have been revised to be compatible with the new test.

Completion of this conversion consummates the production line tester improvement program. The combined efforts of TME and NRD personnel have resulted in a 50,000 dollar per year reduction in tester operating cost.

Development Co-Extruded Fuel Tester

Fabrication of a UE-2B co-extruded fuel core tester, which will be used by Testing Methods for developmental purposes, has been completed. The tester will be "debugged" as soon as time permits.

Lithium Aluminum Target Elements

An ultrasonic testing system was utilized to test the components for the fabrication of lithium-aluminum NFR target elements. Ultrasonic tests were made on the aluminum tubing before drawing and numerous flaws were detected. A red dye penetrant test was introduced in the test scheme to screen out the grossly defective tubing prior to drawing. Ultrasonic tests were also performed. Acceptable portions, in two-foot lengths, were used to fabricate the fuel elements. The zirconium tubes used to sheath the aluminum clad elements were also ultrasonically examined. All tubing was found to be acceptable.

DOUGLAS-UNITED NUCLEAR REACTOR PLANT SUPPORTKVE Experiments in the PCTR

Computer and hand analyses are being continued to determine the infinite medium multiplication factor and thermal utilization of wet and dry KVE fuel in a 7-1/2" graphite lattice. Detailed flux traverses are nearly completed for all cases, and this will be further corrected for the effects of flux peaking.

Corrections have already been applied to the data to correct for differences in foil geometry, gamma counting efficiency, and resonance absorption in the copper detector foils. The result will be values of k_{∞} and f for a graphite reactor uniformly loaded with KVE fuel under both wet and dry cold, clean, green conditions.

Pu-Al-Th Supercell PCTR Experiments

k_{∞} measurements are being made in the PCTR for a graphite lattice containing Pu-Al and Th in alternate cells with an 8" pitch. The measurements are being made with and without water in the coolant channels. The experimental phase of the dry case is completed, and the analysis of the data to yield k_{∞} is in progress. The experimental phase of the wet case is in progress now. The mass of copper required to poison the test cell to unity multiplication was found to be especially sensitive to the spectral mismatch of the buffer cells as measured by the cadmium ratio of gold detectors. Efforts are being made to evaluate the suitability of other detectors for spectrum matching measurements.

Vibration Analyses

Over 200 tape recordings of vibration data from 100-D Reactor were analyzed using a Spectral Dynamics sweeping filter and sweep oscillator. As a check on the accuracy of the data, eight of the recordings were analyzed using a parallel fixed filter technique on the analog computer. The results of the two methods, when normalized to the same magnitude, give excellent agreement. Cross check of actual magnitudes has not yet been done.

Detection of Hydride in Zircaloy Tubes

A prototype 5 Mc eddy current tester has been developed to detect hydride in Zircaloy tubes. During this reporting period three tubes were tested in-reactor and four tubes were tested ex-reactor. The data obtained will now be correlated with destructive analysis to be performed on the tubes. Laboratory experiments to develop a temperature stabilized probe are progressing.

Corrosion Thinning of Aluminum Process Tubes

Development is continuing on an ultrasonic system for use in detecting areas of localized corrosion thinning in aluminum process tubes. A lease agreement was arranged for temporary use of commercial ultrasonic test equipment.

Various preparations are being made to use the leased equipment at 103-B where a reactor tube mockup facility is located. Included in the preparations are resealing a nozzle adapter, checking out a light galvanometer recorder, and setting up for probe translation experiments.

Dimensional Variation Measurement of HCR's

Reactor Engineering's dimensional measurement tester was used to check the variation in horizontal control rod number two in 105-C Reactor. Measurements showed that the rod had not swollen significantly.

DOUGLAS-UNITED NUCLEAR FUELS PLANT SUPPORT

Bond Test for Mechanically Coupled Surfaces

A study to determine the technical feasibility of identifying unbonded surfaces in intimate mechanical contact is in progress. One approach under study uses highly sensitive liquid crystals to detect minute temperature variations. Results of tests employing on-site available crystals were not favorable. Some commercially available liquid crystal material has been ordered.

Two-frequency ultrasonic methods are being investigated for detection of unbonded surfaces in intimate surface contact. A low frequency, high power, ultrasonic system was used to try to separate the mechanical contact momentarily so that a high frequency system could detect it as an unbond. No significant results were obtained.

SEPARATIONS PLANT SUPPORT

Instrumentation and System Studies

Development was started on a Resin Exchange (REX) process model which is similar in many respects to the Aluminum Nitrate Nonahydrate (ANN) model previously developed. The REX process differs in that its PH is changed by addition to the process of an acid or base. The ANN process model appears usable for the REX process providing the variables are appropriately transformed.

Experiments demonstrated the usefulness of employing a neutron source and detector system to detect the interface between typical aqueous and organic solutions within a large stainless steel process pipe. Results warrant further experimentation with this method. Other approaches will also be investigated.

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A probe has been designed for improved detection of abnormal concentrations of Am^{241} in a specific process hood and the design was initiated on the associated instrumentation. The complete system will be termed an Am^{241} Breakthrough Monitor.

Engineering assistance aided in planning the establishment of a general purpose counting system, which will be assembled from quality, commercially-available, modular instrumentation.

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CHEMISTRY DEPARTMENT

IRRADIATION PROCESSES

Overheated N-Reactor Fuel Study

Two irradiated N-Reactor outer fuel tubes were tested for cladding integrity in a steam atmosphere below the melting point of uranium. Test conditions were to study the relationship of rupture to time and temperature after loss-of-coolant. The heating rate of the first fuel was programmed to reach 900 C in 5 min, then to increase 2 C/min to rupture. Separation of the cladding from the U started at the four visible supports at about 920 C, with the area of separation growing with time. Rupture occurred at 1010 C, approximately 53 min after the start of heating. The rupture was a 1 in. longitudinal crack starting at the upstream end cap. The fine crack occurred at one of the bubbles produced by the internal gas pressure.

The second tube was heated at 925 C until rupture 92.5 min after the start, or 87.5 min after reaching 925 C. The cladding sustained many pressure blisters. Cladding was missing from an irregular area about 1 in. in length starting from a point at one end to a width of about 1/4 in. at the other end. This occurred at the bottom near the downstream end cap. After the initial burst, fission product release was at a relatively slow, uniform rate. Oxidation and internal gas pressure appeared to have caused loss of cladding and rupture.

An irradiated inner N-Reactor element was heated rapidly until rupture occurred; the rate simulated that of the core material. The upstream end cap was blown out at 1055 C, 3.2 min after start of heating. Uranium extruded from the upstream support at about 3.4 min and at 1070 C. Cladding had sustained a large blister. Heating was stopped at 3.6 min and 1075 C when the U was oxidizing rapidly enough to raise the temperature above its melting point.

Noble Gases in Reactor Accidents

The rate at which dissolved noble gases would be released from a water basin was calculated for three idealized cases. These were: a single stage of a sweeping gas (air) contacting the liquid, a double stage of such contacting, and for the basin covered with a stagnant liquid film. The results indicated that methods could be developed for delaying and controlling the release of noble gases from a water basin.

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This indirect calculation approach was dictated by the fact that there is no known method for accurately predicting the rate of mass transfer of a gas from the surface of a solution within which the degree of mixing is variable or unknown.

SEPARATIONS PROCESSES

Solvent Extraction Recovery of Polonium

The first step in the solvent extraction process under development involves extraction of Po(IV) at 50 C from a 1M Bi(NO₃)₃-2.4M HNO₃ feed solution into dibutyl carbitol 1.6M in nitric acid. A miniature mixer-settler run was made under these conditions using feed containing Po at about 1/4 expected plant level. The losses to the aqueous raffinate were 0.03 and 0.49% with 12 and 6 extraction stages, respectively. The Bi separation factor was ca. 50 with 12 stages and ca. 110 with 6 stages. Batch contact experiments simulating the strip column indicated that Po is stripped more effectively with HNO₃-H₂C₂O₄ solutions than with oxalic acid alone. Greater than 97% of the Bi present in a 1.5M Bi(NO₃)₃-4M HNO₃ solution precipitated as a dense white solid (presumably BiONO₃) when the HNO₃ was destroyed with formaldehyde. This has been suggested as a desirable initial step in the recovery of Bi from solvent extraction process raffinates.

Associated pilot plant studies were made to demonstrate the extraction behavior of dibutyl carbitol in Redox-type packed columns. The flowsheet was modified slightly to provide optimum Po recovery at elevated temperatures. The flow-sheet utilizes 4 columns--IA, IS, IC and IO--performing successively the functions of Po extraction from the Bi(NO₃)₃-HNO₃ feed solution, Bi scrubbing from the Po-containing solvent, Po stripping and solvent cleanup. Nominal flow-sheet compositions and relative flow rates were as follows:

<u>Stream</u>	<u>Composition</u>	<u>Relative Flow</u>
IAF	1.67M Bi, 2.4M HNO ₃ , 0.1M NH ₂ SO ₃ H	0.3
IAX	1.6M HNO ₃	1.0
ISS	0.75M HNO ₃ , 0.5M NH ₂ SO ₃ H	0.125
ICX	0.2M HNO ₃ , 0.2M H ₂ C ₂ O ₄	1.0
IOS	0.5M Na ₂ CO ₃	0.2 - 0.5
IAFS*	1.23M Bi, 2.6M HNO ₃ , 0.07M NH ₂ SO ₃ H	0.425

*IA column feed after dilution with IS column aqueous effluent.

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The 3 in. diam columns used in most of these studies contained 19 ft of 3/4 in. stainless steel Rachig ring packing. They were operated with the aqueous phase continuous. The maximum stable capacities demonstrated for IA, IS, IC and IO columns were, respectively, ≥ 1730 , 1160, 1940 and 700 gal/hr(ft²), sum of flows, at 25 C and >2000 , ≥ 1360 , 1610 and 900 gal/hr(ft²) at 50-54 C. With this flowsheet the relative diameters of the columns would be 1.0, 1.08, 1.32 and ~ 1.4 to obtain a matched capacity.

Bismuth and HNO₃ stripping efficiencies were measured in the IS and IC columns, respectively, at flow rates within 80% of the maximum capacity. Over 99.8% of the extracted Bi was removed from the solvent and the overall transfer-unit height (HTU_{OR}) was ≥ 2.9 ft. The overall Bi decontamination factor, using the high temperature flowsheet conditions, was >5400 . Nitric acid HTU's were obtained in both 10 and 19 ft packed columns, using water as ICX. The values ranged from 2.8 to 3.0 feet in the shorter columns to ≤ 3.8 ft in the taller. The HTU values for both Bi and HNO₃ were about 2-fold higher than the corresponding HTU values measured in the earlier pulse column runs.

Although the dispersions obtained in the IA, IS and IO columns ranged from excellent to good, the dispersion in the IC column was quite poor at all flow rates. A few runs were made in a 9 ft pulse column containing 4 in. spaced, 23% free-area nozzle plates (3/16 in diam holes) to determine the performance of a pulse column at comparable high rates. The maximum capacity in this column was greater than 1860 gal/hr(ft²) with a 1 in. pulse amplitude and a pulse frequency of 65 cycles/min, and the nitric acid HTU values were in the range of 1.6 to 2.1 ft at 1530 gal/hr(ft²). The stability was rather poor in this column when operated with the aqueous phase continuous. A heavy buildup of solvent in the top half of the column was subject to cyclic coalescence into large, channeling globs over a wide range of pulse frequencies. The performance was more stable with the solvent phase continuous and this type of operation would probably be preferred.

Recent experiments, contrary to earlier indications, have shown that polonium does not extract from sodium hydroxide solutions. In the earlier experiments the Aliquot-336 was not pre-equilibrated with sodium hydroxide. The more recent experiments suggest that PoCl₅ is extracted at first but converts to the less soluble PoO₃ after a short period of time.

Effect of Impurities on Dose Rate from Recycled Bismuth

Calculations were made to establish the concentrations of various impurities which would result in the same dose rate as that due to the long-lived Bi isotopes arising from a single cycle Hanford reactor irradiation. The preliminary results showed Ag and Co to be quite important. The significance of various probable impurities is shown below:

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<u>Element</u>	<u>Nuclide</u>	<u>Concentration (ppm)</u>
Fe	Fe ⁵⁹	0.8
Co	Co ⁶⁰	0.0003
Ni	Ni ⁵⁹	>1000
Cu	None	-
Zn	Zn ⁶⁵	0.03
Sb	Sb ¹²⁴	0.02
Ag	Ag ^{110m}	0.0001
Mo	None	-
Ta	Ta ¹⁸²	0.002

Thorium Calcination

Work continued to demonstrate the applicability of an agitated trough calciner for the conversion of thorium nitrate to a ThO₂ with suitable properties for target element fabrication. The 4 in. diam calciner test section was installed in a containment hood within C-Cell of the 321 Building; the agitator drive assembly was connected; and the Th(NO₃)₄ feed control system installation was completed. The piping, electrical and instrumentation tie-ins to the calciner were 85% completed.

The purification of a supply of Th(NO₃)₄ for use in the calciner test program was started, using the A-C pulse column test facility. A modified Thorex acid flow sheet was selected to separate the Th from the gross quantities of Al and lesser quantities of F, Na, Fe, Ni, Cr and Hg.

Some difficulty was experienced initially with the A-Column interface detector which is a capacitance-type probe positioned in the bottom disengaging section of the column. This difficulty was encountered when the LAW losses were high and the second, heavy organic phase extended to the bottom of the extraction section and settled on the interface. Apparently, with these conditions, the dielectric strength of the aqueous phase (LAW) and the heavy organic phase were very nearly equal; consequently, there was very little response to the interface detector even though pronounced interface fluctuations were occurring. The erratic interface control aggravated the overall situation by causing erratic extraction section performance and thus a further instability in the interface solutions. Flowsheet modifications were made which reduced the LAW losses and stabilized the formation of the second organic phase at a higher level in the extraction section. The last two operating periods have been marked by stable operation and approximately 120 pounds of purified

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$\text{Th}(\text{NO}_3)_4$ have been produced. A sample of the product TNT solution was denitrated and the resulting oxide was then dispersed in dilute nitric acid (the standard sol gel process). Approximately 90% of the oxide dispersed and formed a stable sol.

Clean U^{233}

As a result of the difficulties encountered in applying the sol gel process to Purex process Th, other possible ways of preparing ThO_2 target elements were scouted. It was found that the spray-calcined Purex ThO_2 can be cold pressed and sintered to 87% of theoretical density by heating overnight at only 1000 C. Cause of this behavior is the presence in the recovered $\text{Th}(\text{NO}_3)_4$ of about 1.6 wt % Na^+ , which acts as a flux or sintering aid. Unlike sol gel ThO_2 , the sintered material appears to be extremely inert to water. Pellets 1.25 in. in diameter exhibited less than 0.0001 in. swelling on two weeks' immersion in cold water. Samples also appeared unchanged after 24 hr in boiling water. Rate of HNO_3 dissolution of the sintered ThO_2 is being determined, as is the value of a powdered aluminum dissolving aid (which would be incorporated in the pellets at time of manufacture).

Disposal to Ground

Ground water samples from well 199N-3, located between the 1301-N Crib and the river, showed two I^{131} peak concentrations which were spaced about 22 days apart. They appear to correspond to ruptures which occurred 23 days apart and which would require a travel time for the I^{131} from the disposal crib to the well of approximately 40 days. Riverbank spring samples showed an I^{131} peak corresponding to a different rupture; in this case the travel time from the disposal crib to the river was about 108 days. The ruptures detected at the well are, on this basis, expected to appear at the springs on December 20 and January 12.

Horizontal spread of waste in the ground beneath the Redox 216-S-9 (process condensate) crib appeared to be approaching a steady-state condition for present discharge rates. Scintillation logs of the two monitoring wells showed very slight changes in the areal extent or activity densities of the contaminated zones during the month. Analyses of water samples from the two wells showed that no significant contamination had reached the water table.

Waste-Soil Relations

Column studies of Pu and Am sorption from acid wastes were continued using soil materials selected from well samples at the Z-1A tile field site. Soils from 5 - 135 ft and from 160 ft to ground water neutralized ca. 0.43 meq acid/gram soil while the caliche zone at 135 - 160 ft neutralized 3.26 meq/gram. The work showed that about 1 column volume of waste could be infiltrated into the soil material from 5 - 135 ft and below 160 ft before the neutralization capacity of the soil was exhausted. Soil in the zone from 135 - 160 ft [REDACTED] neutralize as much waste as the rest of the profile; however, flow characteristics through this stratum are very poor and lateral spreading may occur. Work started with similar soil-waste systems differing in equilibrium H-ion concentration.

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WASTE MANAGEMENT AND FISSION PRODUCT RECOVERY

Effect of Complexants on D2EHPA Solvent Extraction Processing

A preliminary study was performed earlier on the effects of tartrate and HEDTA complexants on the behavior of Ce, Sr, Pm and certain impurity ions in the Phase-III (B-Plant) waste management flowsheet. This work has now been expanded to include two other candidate complexants, gluconic and citric acids, as well as the behavior of Fe and Pb and the effect of contact time in all four systems. The results will be compiled in an informal report.

Sludge Monitoring in Waste Storage Tanks

The sensitivity of an air-filled ion chamber for the detection of residual sludge deposits in waste storage tanks was determined using a $1/2$ Ci Cs^{137} gamma source. At about 3.2 r/hr the chamber read 1×10^{-11} amp and at 0.5 r/hr it read 4×10^{-12} amp without the $1/4$ in. diam Pb collimator fabricated for this purpose. This sensitivity will be compared with calculated detector dose rates for the expected source-to-detector distances and gamma emitter concentrations. Increased sensitivity is possible using a scintillation detector in place of the ion chamber if calculations so indicate.

Ion Exchange Contactor Demonstration

The testing of the multi-stage ion exchange apparatus was continued. The Ce product yield for Test No. 2 continued to increase and reached a peak value of ca. 10%. Decontamination factors reached average values of 5×10^2 with individual DF's as high as 2×10^3 . The test then deteriorated when a mold growth clogged the fine screens in the apparatus. An attempt to remove the mold with sodium hypochlorite additions was successful but the product yields dropped to near zero levels following the addition.

A third run was started with fresh resin in each stage preloaded with amounts of Cs and Ce tracer predicted by the DIXIT/CR computer program. The process then operated essentially at steady state as predicted.

Resin Column Liquid Level Detection

Two resin level measuring devices were selected for further study and development as plant prototypes - the incremental conductivity method and an adaptation of the ultrasonic depth sounding technique. Other methods evaluated included time domain reflectometry, capacitance, float devices, load cells, continuous conductivity and a vibrating reed viscosimeter. The time domain reflectometer and the capacitance devices failed to detect resin level due to the high electrical conductivity of the process solutions. Other methods and their limitations are as follows: vibrating reed viscosimeter, short range;

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continuous conductivity, short range or non-conducting vessel; load cells, accuracy and overload requirements; float types, operational in slurries but not in a settled resin.

EQUIPMENT AND MATERIALS

Stainless Steel Specifications Tests

Very unusual results were obtained in Huey tests of several lots of 1 in. diam 304-L stainless steel tubing destined for use in Purex concentrator tube bundles. All lots passed the test when the multi-sample tester was used but all failed when tested separately, one to a flask. In all previous experience, Huey test corrosion rates for 304-L stainless steel have been essentially independent of the test techniques used. Principal difference between the two techniques is the accumulation of corrosion products. The concentration of these is always low (ca. $10^{-5}M$) in the multi-sample tester but may amount to as much as $10^{-2}M$ at the end of a test period when individual flasks are used. The corrosion rate in simulated Purex 1WW was about twice that normally found. The material appears to be within specifications as to carbon content. Its micro-structure is typical of 304-L although the amount of "carbide" at grain boundaries in sensitized material appears to be greater than usual for its carbon content. Further chemical analyses and other tests are in progress in an attempt to resolve the problem.

Storage of Phosphoric Acid

Pursuant to the use of phosphoric acid in the processing of ThO_2 in the Purex plant, the corrosion of 304-L stainless steel in C.P., technical grade, N.F., and food grade 85% H_3PO_4 was determined at 25 and 40 C. Corrosion rate in all grades but technical was nil. The specimen exposed to technical grade H_2PO_4 at 40 C activated after about 48 hours, presumably due to fluoride impurity, and corroded thereafter at a rate in excess of 15 mils/mo.

Corrosion of Tantalum

Tantalum coupons exposed to boiling Zirflex decladding solution ($6M NH_4F-0.05M NH_4NO_3$) corroded at about 15 mils/mo in both the liquid and the vapor phase. Use of Ta in a demister exposed to Zirflex decladding solution off-gas appears to be questionable.

pH Measurement

The importance of accurate and reliable pH measurement in achieving proper performance of chemical separations and purifications processes has resulted in a return to this problem. Of particular concern is the performance and service life of pH instrumentation in high radiation fields. An experimental

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program was designed to determine radiation damage effects in pH electrodes in a gamma field of 10^6 r/hr; apparatus was designed and assembled for performance of experimental studies with a cobalt source. An improved pH sample cell was designed to provide easier and more rapid electrode replacement and with a lower initial cost.

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APPLIED MATHEMATICS DEPARTMENT

Chemical Processing Department

Separations R & D

Innovations - U233

A response surface design procedure was formulated to obtain experimental data which will form the basis for predicting the absorption of H_2O and CO_2 by each of three different types of thoria, as a function of surface area and relative humidity.

A detailed graphical presentation is being prepared on the results of a previously designed response surface study of the dissolution of thorium oxide by a 4-constituent solvent.

Irradiation Processing Department

Fuels R & D

Process Technology - Hot Die Sizing

A response surface analysis was carried out on experimental data obtained from autoclaved aluminum.

Fuel Development

A flux-oriented reactor loading (charging) plan is being formulated to facilitate meaningful comparisons among 52 types of fuel elements.

N-Reactor Department

Nuclear Safety

A decision theoretic reliability model was formulated for each of two operating policies. The models incorporate probabilities and costs of "reactor dumps" reflecting the availability of electrical power from three on-line alternate sources.

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RADIATION PROTECTION DEPARTMENTEnvironmental Studies

Samples of ground beef from a calf grown in the Riverview district of Pasco, and intended for a Zn-65 ingestion experiment were found to have an average Zn-65 concentration of 5.5 pCi/gram. The concentration was remarkably uniform, with a standard deviation of less than 13 percent for the samples analyzed.

A multiple regression calculation of whole body counting and diet questionnaire data on 2600 individuals gave estimates of the Zn-65 body burden contributed by various types of food. A planned modification of the program may alter the final regression coefficients, but the present results indicate that on the average 69 percent of the Zn-65 is obtained from drinking water, and 27 percent is obtained from seafood. Other contributors in order of importance are game birds, Columbia River fish, and fresh meat. Milk consumption continues to give a negative correlation with Zn-65 body burden.

A study of the flow regime in McNary reservoir was begun with temperature traverses above and below the Snake River and above and below McNary Dam. The reasons for extending previous studies are (1) to determine the mixing pattern of the main stream (containing reactor effluents) in McNary reservoir, (2) to determine time of travel through the reservoir, and (3) to establish a reliable continuous monitoring point at the dam.

A study of the radiological problems that may exist on the north slope of Rattlesnake Mountain if this area were opened up for farming was completed and a draft of the report was submitted. The study concluded that it is feasible to allow fruit and dairy farming and cattle grazing on the slope. Exposures of persons residing on the slope would be similar to those received by persons presently living in the perimeter areas where the maximum exposure occurs. Because of the slope's proximity to the Chemical Separations areas, the probability of contaminating crops to a level requiring confiscation is possibly greater than at other plant perimeter locations following a major accident in a separations facility under adverse meteorological conditions.

Environmental Evaluation

The small increases noted in I-131 concentration in milk, following increased I-131 release rates from the separations facilities in the latter part of October, continued into early November. The maximum value measured was 27 pCi I-131/liter of milk on November 9, 1965. Two other samples contained $\approx$ 20 pCi/l. Concentrations returned to values below 10 pCi I-131/l by mid-November. Other aspects of the radiological status of the environs were in the normally expected range for this season of the year.

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Hexavalent chromium ion and Cr-51 concentrations in the river below the reactors are decreasing as expected following the shutdown of the old reactors. Similar reductions in the concentrations of other radionuclides are not yet clear.

The teflon impregnated lithium fluoride discs were received and several exposures completed. The discs are 1/2 inch in diameter, about 0.015 inch thick, and have about 20 mg of Li^7F mixed with the teflon. The present TLD reader requires maximum current through the special disc planchettes to properly read out the teflon-LiF discs. A stainless steel screen is used to hold the discs flat against the heating surface. To reduce the amount of light stopped by the screen and increase the relative light reading, a hole was cut in the center of the screen. Enough screen was left to insure proper heating of the discs. Sets of five discs were exposed to 50, 100, 200, 400 and 500 mR of radium gamma radiation and readout with the special planchettes. The background reading is approximately 15 mR equivalent, and the system is adequate to detect doses of 100 mR or more. Energy response, fading, and other properties of the disc dosimeters remain to be determined. These dosimeters are being studied as possible replacements for the currently used finger ring film. They offer the advantages of a low detection limit, ease in handling and no darkroom requirements.

Degraded spectrums of alpha particles from uranium and plutonium sources were studied with polycarbonate plastics and cellulose nitrate. Alpha particle tracks were not observed in either the plastics or cellulose acetate until they were degraded with 5 mils of Lexon film. The degraded alpha particles have a higher linear rate of energy loss ($\frac{dE}{dX}$) which is above the critical rate of energy loss for track formation. Cellulose acetate, cellulose acetate-butyrate, polymethylmethacrylate, polyvinyl formal, and polysulfone plastics were exposed to alpha particles from uranium metal and etched in sodium hydroxide, nitric acid, and hydrochloric acid solutions of various concentrations. None of these plastics will record alpha tracks. The alpha particles cannot provide a linear rate of energy loss sufficient to produce a damage track.

Commercial nitrocellulose plastic contains camphor or other plasticizers which apparently cause small etch pits similar in appearance to alpha track etch pits. Dissolving the plastic in various solvents or treating the surface had no effect. By dissolving "Parlodion", a highly purified pyroxylin, in amylacetate or ethyl acetate and allowing a film to dry on a clean microscope slide, it is possible to make cellulose nitrate films which have a greatly reduced number of background tracks.

Fading of Co^{60} irradiated Bausch and Lomb "Low-Z" glass rod dosimeters during a three-month period was non-existent. In fact, during the three months, a slight increase in fluorescent reading was evident. However, most variations in glass rod readings are within the accuracy limits imposed by instrument stability.

The basic geometry and shielding requirements of a personnel dosimeter badge utilizing "Low-Z" glass blocks as radiation dosimeters were developed. Several glass dosimeter blocks with dimensions of 8 x 8 x 5 mm and 6 x 6 x 3 mm were ordered. The available literature indicates a 10 mR Co^{60} exposure can be measured with a 5% standard deviation using suitable readout instrumentation and the 8 x 8 x 5 mm glass dosimeter.

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B. DIVISION OF MILITARY APPLICATION: 03 PROGRAM

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SUMMARY OF ACTIVITIES

PHYSICS AND INSTRUMENTS DEPARTMENT

See Page B-3.

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PHYSICS AND INSTRUMENTS DEPARTMENTTensile and Compression Test Sample Heating Equipment

General design requirements, including specification of appropriate commercial equipment, were established for the development of experimental apparatus for obtaining tensile and compressional data on heated samples. The project involves the heating of test samples to temperatures from 100°C to 1000°C within a 10-second time period. Resistive heating, with currents to 2000 A, will be used along with controls to limit temperature overshoot to a few percent. Considerable modifications to the commercial furnace controls will be necessary to achieve the required operation on an automatic basis.

Thermal Diffusion Measurements

Completed development effort established the necessary instrumentation for conducting thermal diffusivity measurements on metallic samples. In essence, the instrument employs a bank of energy storage capacitors (160 μ F) and a high power flash tube to apply a fast heat transient to the test specimen. The capacitor bank is charged to 4000V. The remaining problem is that of providing an accurate measurement and recording of the sample temperatures.

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C. DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY -
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SUMMARY OF ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTPlutonium Recycle Program

Thirty-four Mark I-R series High Power Density fuel elements have been assembled since the start of fabrication in June; seven of these were fabricated this month. The pneumatic impaction of the core material, amounting to 544 kg of 2 wt% PuO₂-UO₂, has been completed. All zirconium components for the Batch Core loading are on hand.

Postirradiation examination of the three fuel rods from the High Power Density fuel element 6504, which failed in the PRTR Rupture Loop, is in progress. An ex-reactor test of a defective fuel rod similar to fuel element 6504 is being conducted under the same coolant conditions experienced by 6504.

A joint program has been initiated between Battelle-Northwest and Argonne National Laboratory for transient irradiation testing of UO₂ capsules in TREAT.

The first three bundles of Shippingport fuel for dosimetry measurements of high exposure plutonium have been received.

Conceptual design of a 1000 kg/day plant for mixed oxide pellet production has been completed and the information is being used in a computer program to calculate fuel cost.

Tensile data from specimens of an irradiated PRTR tube indicate an annealing-aging process occurred at 550°F (290°C).

The ID gage for the simultaneous measurement of six PRTR tube diameters performed well in monitoring the full length EDEL-1 loop tube. Outside diameter measurements of a PRTR tube were made which showed a 0.004-inch diameter increase in the core section of the tube.

Prior to resumption of normal PRTR operations the H₃BO₃ control ion exchange system will be relocated to avoid possible high radiation levels. Small amounts of rupture debris remaining in the moderator system after cleanup could be concentrated by the H₃BO₃ ion exchangers and cause a significant radiation problem.

Mechanical and flotation techniques for removing fuel rupture debris from the PRTR moderator and gas systems are being evaluated to provide candidate procedures for improving the efficiency of drain-flush operations. Vibration has been shown to increase Fe₃O₄ removal from bellows assemblies on the PRTR Pressure Tube Mockup. Flotation experiments are still in the preliminary stage; further work is required.

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Attempts to adapt the $\text{H}_2\text{SO}_4\text{-HNO}_3$ ceramic fuel dissolution process for use in systems containing aluminum and carbon steel were unsuccessful. Work is currently being directed toward the development of a two-step procedure for aluminum and carbon steel systems involving first UO_2 dissolution and then PuO_2 dissolution with concentrated H_2SO_4 .

The PRTR primary system decontamination process was successfully demonstrated in the G7/M3 Loop in the ETR at NRTS. The over-all loop decontamination factor was 12.6.

Development of the shroud tube removal and replacement equipment is continuing with reduced priority. Electrical and mechanical assembly of the 6-inch removal machinery has been completed; the weld cutting head has remotely cut a 6-inch tube mounted on the mockup.

An estimate was made of the size of the molten fuel region in the FERTF fuel element at the time of failure, September 29.

Nuclear Ceramics Research

The band gap for UO_2 was determined by electrical resistivity measurements to be 3.0 ev.

Hypostoichiometric mixed uranium-plutonium oxides are being synthesized to investigate the effects of stoichiometry on various properties. A ThO_2 induction furnace for measuring high temperature phase equilibria in oxidizing atmospheres was designed. Oxygen redistribution in oxide fuels is being studied using special postirradiation sampling techniques.

Feasibility of the carbothermic reduction scheme for mixed nitride synthesis has been demonstrated.

Swelling Studies

A general swelling capsule operating at a control temperature of 700°C (beta phase) and at a pressure of ~ 30 psi reached its goal exposure and was discharged. Two controlled temperature pressure capsules (550°C and 625°C at 500 psi) were discharged at a burnup of 0.16 at. %.

Irradiation Damage to Reactor Metals

Specimens of irradiated W, Ta, Mo, and Cb refractory metal alloys have been tested in tension at a temperature of 2000°F (1095°C). Almost complete recovery of irradiation damage was observed for the Cb and Mo alloys whereas Ta alloys tested showed marked increases in elevated temperature strength and decreases in ductility as measured by uniform elongation.

Zircaloy corrosion specimens in out-of-flux positions in the ETR G-7 loop were examined for weight gain, β/γ and α activity after a 48-day exposure.

Laboratory tests on a mockup of the seal between the pressure tube of the ATR Gas Loop and the columbium sleeve liner indicate the maximum clearance between the liner and the sleeve should not exceed 0.020".

Major effort during the period was concentrated on producing good electrical connection between tubular elements of the model gas loop heater. Sound metallurgical bond between Nichrome tubes and their connectors were produced. The same technique could not be applied successfully to molybdenum tubes and their connectors.

Nichrome has been found to oxidize according to a cubic rate law at 1020 and 1120°C over an oxygen pressure range of 0.14 torr to 85 torr. At higher temperatures, oxide loss by evaporation is suspected.

Nuclear Graphite

Porosity measurements show that the higher bulk thermal expansion observed for carbons with low graphitizing temperature (<1800°C) is not due to smaller open-pore void volume. It may result from smaller Mrozowski-type microcracks and/or higher elastic constants which are independent of crack size.

The oxidation rate of graphite has been varied by a factor of 80 with no apparent effect of rate on sonic modulus.

The yield of CO in the neon-sensitized radiolysis of a mixture of CO₂ and H₂ varied from 3.6 to 3.2 molecules/100 ev as the neon concentration was changed from zero to 96%. The CH₄ yield, though small, increased as the proportion of neon increased. This shows that energy is transferred very efficiently from the neon to the CO₂ and H₂, as has been found for He and Ar previously.

Metallic Fuel Development

Postirradiation annealing studies have shown that a submicron size dispersion of UC particles in uranium inhibits fission gas induced swelling up to temperatures of 800°C.

Irradiation of high exposure thorium-uranium fuel assemblies in the ETR-F7 Loop was re-initiated after a delay caused by a loop discharge incident.

Conceptual Reactor Design Studies

Burnup and sodium coefficient calculations for the nitride fueled reactor were completed.

Upstream Boiling Burnout

Analysis of the data was completed on two recent series of laboratory experiments. Definite indications were found that the cause of burnout occurring upstream of the outlet end is associated with a transition in flow patterns.

Fast Flux Test Facility

Preliminary experiments were performed to evaluate coining as a method for powder consolidation of cermet prior to extrusion. Several methods are under investigation for preparing spherical PuO_2 particles. A particle crush strength apparatus was calibrated.

High and low density stainless steel PuO_2 cermets were irradiated to burnups of 2.3, 5.0, and 7.5% of plutonium atoms. The high density cermets were in excellent condition. In the low density cermets the stainless steel matrix sintered and underwent extensive grain growth. The PuO_2 reacted with the stainless steel at temperatures estimated to be 1000 °C and higher. The reaction did not appear to impair cermet integrity.

Work on the sodium bonding facility is approximately 85% complete.

The 1000 hour run of the HTLTR mockup is continuing. More than 30 consecutive days at the goal core temperature of 1000 °C have been achieved.

Maximum burnup attained with impacted EBWR UO_2 -1.5 wt% PuO_2 fuel is 22,500 MWD/ton.

PHYSICS AND INSTRUMENTS DEPARTMENTPlutonium Recycle Physics

A series of measurements are being conducted in the PRCF to verify calculational methods used in the analysis of the HPD core for the PRTR. A brief summary has been written of the results of the PRCF experiments which have been conducted to date. Wherever possible, comparison is made to calculate parameters in order to assess the adequacy of calculational models used in the analysis.

Analysis of burnup data from the irradiation of UO_2 or PuO_2 - UO_2 fuels requires a knowledge of the fraction of all fissions resulting from the fission of U-238 by fast neutrons. Program GAMTEC-II has been used to calculate these values

for a cylindricized model of a PRTR 19-rod cell. Data from the destructive sampling of aluminum 1.8 w/o plutonium fuel rods have been obtained as a function of position in a given rod. The a/o Pu-239, a/o Pu-240, a/o Pu-241, a/o Pu-242 and the percent depletion, β , of the plutonium, have been determined. Analytic functions have been used to describe these data in order to deduce the average values of the above data per rod and per cluster, and to aid in the process of assigning uncertainties.

A series of improved calculations of plutonium-fueled light water reactor systems are in progress. The range of conditions (lattice spacings, fuel diameter, burnup, and void fractions) bracket the regions of interest for light water pressurized and boiling water power reactors. The results of the study are being summarized and will be reported in a formal document.

Experiments to measure critical masses, bucklings, reflector savings, and control rod worth, for 2 w/o PuO₂-UO₂ fuel rods moderated with H₂O are continuing in the Critical Approach Facility. Experiments using fuel containing 24% Pu-240 have been completed. In addition, an experiment has been completed using 1.8 w/o Pu-Al fuel rods containing 0% Pu-240.

A new program, HXXS, can now be used to compute cross sections using HRC and THERMOS, and to execute the reactor code HFN in one pass. A major option of Program HXXS is to have input cards punched for Program S.

The IBM-7090 computer programs, PLOTCH and LLESQS, have been converted for use in the UNIVAC-1107 computer.

The current version of the BNW Master Cross Section Library has been printed on duplimat paper for publication as report BNWL-CC-325. This report will be available for distribution in approximately one month.

The thermal neutron scattering law for glycerine has been calculated using the technique of Egelstaff and Schofield. The procedure used should give adequate results for the applications presently envisioned.

PRP Fuel Cycle Analysis and Specific Cycle Analysis

The following papers were presented at the ANS Meeting in Washington: Monte Carlo Calculations of Optimum Loading of Plutonium in Fuels; A New Fuel Configuration to Increase Plutonium Value; A Comparison of Fuel Cycles for Crossed-Progeny Fuels in Conventional and Novel Configurations; The Calibration of a High Speed Burnup Code for Nuclear Fuel Surveys; Conservation Economics and Reactor Technology.

Neutron Cross Sections

Slow-neutron scattering measurements during the month were hampered by unusual reactor leakage neutron levels and reactor outages.

A sample holder was prepared for use in the study of the inelastic scattering of slow neutrons from D₂O.

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The study of multiphonon effects in slow-neutron scattering was continued with further calculations of representative models of the scatterer. Further program deficiencies in the computer programs LEAP and ADDELT were uncovered in the course of these calculations.

Samples of several separated isotopes used for MeV total cross section measurements were prepared for return to ORNL.

Phoenix Fuel Program

The proposed MTR-Phoenix experiment was discussed with personnel of the NRTS at Idaho. On the basis of the requirements outlined, it would appear that a 3 x 9 core, 23.5" long with 0.117" water gaps would be potentially acceptable. Achievement of 40 MW in a core of this type is, however, quite problematical. As a consequence of the visit to NRTS a decision was made to abandon the previously selected 5 x 5 core. Current calculations are aimed at selection of a core concept by the end of the calendar year.

In connection with the current MTR-PHOENIX core design effort, an early set of approach-to-critical experiments utilizing the present Pu-Al disc inventory is under consideration. This information is needed to determine the material requirements for the proposed experiments, and to start preliminary hazards analyses.

FFTF Program

Three isotopes in the 26-group Russian fast cross section set have been updated using recent nuclear data. Knowledge of precise values for the high energy (~ 1 MeV) cross sections of sodium and iron is quite important for fast reactor calculations. Consequently, elastic and capture cross sections for these elements have been reinvestigated for the energy range of 0.01 to 10 MeV in order to update the BNW Master Neutron Cross Section Library. Group average calculations using the updated cross sections give closer agreement with those sets given by other laboratories (Bondarenko, Hansen-Roach, ANL-224).

The methods and models to be employed in calculating the energy release during a maximum credible accident (MCA) in the FTR were discussed with Nuclear Technology Corporation personnel. A research program was outlined.

Estimates of the buildup of Pu-238 in cermet driver fuel ($\text{PuO}_2\text{-SS}$) and in ceramic driver fuel ($\text{PuO}_2\text{-U}^{238}\text{O}_2$) in a 400 MW, 600 liter FTR reference core have been made.

Fuel cycle analysis of fast spectrum, oxide and carbide-fueled breeder reactors is continuing. Of particular interest is a comparison of systems in which the fuel is enriched in one case with plutonium, and in the other case with U-235.

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High Temperature Reactor Physics

The 1000 hour run of the HTLTR mock-up at 1000°C has been in progress for about 600 hours without indication of improper functioning of any of the reactor components under test. These include heater rods, control and safety rods with their actuators, thermocouples, moderator, graphite, thermal insulation, and the containment shell.

Experiments were performed in the PCTR to determine the worth of the HTLTR control rods and driver fuel rods. The measurements confirm earlier design calculations.

Contacts with vendors to investigate the feasibility of using a Pu²³⁸-Be startup source in the HTLTR indicate major objections to their use. There is evidence that recrystallization in Pu-Be sources at temperatures as low as 500°C causes swelling and rupture.

Review of the HTLTR Safeguards report continued.

A design review was completed on several specifications. As a result, specifications will be significantly altered on the neutron generator, panel mounted instruments, radiation monitors, and the criticality monitor.

Installation of the reactor by J. A. Jones Construction Company has been started. Plaster work on all offices, hallways, control room, and change room was completed.

Plutonium Criticality

Installation is proceeding on a new slab tank for criticality experiments incorporating a variable thickness adjustment feature that will permit the determination of critical thicknesses of bare and reflected "infinite" slabs of Pu solution in the thickness range of 3-9 inches.

The aptness of multi-group diffusion theory for computing criticality of Pu solutions in simple geometry (spheres) was further demonstrated by comparisons of measured and computed criticality. Good agreement was obtained with the early experimental results at Hanford.

Subroutines have been written for the Code MISFIT III for the analyses of data from pulsed neutron source experiments. In addition to the determination of the prompt neutron decay constant (α), subroutines have been incorporated for determining subcritical reactivity making use of the Gareis-Russell theory.

Crack Propagation Detection

A program was initiated to investigate the behavior of cracks propagating in stressed pressure vessels by detection and analysis of sonic energy liberated by the growth and propagation of defects. Program objectives are to develop methods of annunciating imminent catastrophic failure. Early tests have detected sharp pulses of energy in notched tensile specimens which occurred before metal yielding.

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Nondestructive Testing

Differences in transmission of some Fourier components of the multiparameter eddy current signal have been overcome by changes made on low pass filters in the demodulator output circuits.

Data from a computer program to calculate ultrasonic interactions at a water-stainless steel interface are presently being analyzed and appear to be in agreement with experimental results.

A peak temperature readout and phase detector triggering circuit has been developed for the thermal transducer system.

Development of prototype eddy current instrument which measures the propagation of cracks in stressed metallic specimens has been completed. Early experimental results reveal the nature of crack propagation mechanisms to be complex, and show main energy propagation to be substantially slower than the sonic velocity.

CHEMISTRY DEPARTMENT

A report on the behavior of about 15 radionuclides released to the PRTR containment system was completed. About 99% of the radioactivity was removed by the air cooler system.

Measurement of the coefficients of thermal expansion (CTE) of phosphate glasses indicated that no problems should exist because of differential contraction between the glass and stainless steel containers.

Calculations of the Pu^{238} in mixed oxide FFTF fuels indicate that only 10% of the isotope present will be produced from n, 2n chain involving U^{238} .

In studies of radoruthenium removal from acid condensate wastes, pulvisite column studies showed that for waste at pH 5.1 less than 10% breakthrough occurred at 400 column volumes and 20% at 1250 column volumes.

Preliminary scanning of 46 melts of interest to the waste program indicated none with the desired properties.

The prototype spray calciner design verification tests gave satisfactory performance on Purex wastes without sulfate by using air atomization. All waste required adjustment of the melting point below 1000 °C to successfully fill receivers. Operation with sulfate-bearing waste has been only partially successful to date.

Closure of the CSE containment vessel was completed and leakage rate tests were initiated. A comprehensive, detailed program outline was prepared and distributed for comment. Tests in the Aerosol Development Facility emphasized demonstrations of CSE prototypic sampling equipment and a study of generation methods for cesium, tellurium, and barium aerosols.

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Experimental design planning was initiated for a new program to study the removal of methyl iodide from reactor containment systems.

Columbia River sediment samples have been obtained in the reach extending from Ringold to McNary Dam.

APPLIED MATHEMATICS DEPARTMENT

Work continued on the keyboard executive and the scan programs for the PDP-7 control computer for the HTLTR. Outlining of the display program for the colored alpha-numeric display tube is well underway. It will be required to display continuously the position of the horizontal control rods, the status of the vertical control rods, current time, maximum temperature, and flux, as well as alarm signals and operator demands.

The EDPM program for the investigation of the generalized Snell's law for visco-elastic media, and the tabulation of reflection factors as a function of incidence angle has been extensively modified to provide data more compatible with laboratory needs. Theoretical investigations are in progress on the nature of power propagation at a plane boundary joining two dissimilar media.

Physical Metallurgy and Applied Mathematics personnel have formed a team effort to document the several statistical techniques developed in the last three years by Applied Mathematics to quantitatively evaluate electron microscopy photographs of irradiated uranium. The effort will include, in addition to the documentation, the writing of individual computer programs to summarize and statistically evaluate Ziess particle size data from photographs. The end result is to be a magnetic tape of quantitative metallographic programs which will be available upon request to Battelle-Northwest personnel.

RADIATION PROTECTION DEPARTMENT

Studies to determine the radiation properties of power reactor grade plutonium blends were initiated. The feasibility of promptly progressing on this program by blending readily available plutonium from Savannah River Project and Hanford to simulate power reactor plutonium met with only moderate success. As a result of this effort, the AEC is preparing an inventory of available plutonium to be considered as input materials to substantially improve the simulated material.

TEST REACTOR AND ENGINEERING SERVICES DEPARTMENTPlutonium Recycle Program

There were no operating periods during the month. All activities were directed toward recovery efforts from the FERTF fuel failure of September 29, 1965, and toward preparations for chemical decontamination of the primary system.

Significant maintenance items accomplished were the removal of 1550 bellows and top shield pieces, removal of the dew cell refrigerator and moisture detection panel, cutting of the 20 dry gas system sample lines, removal of six shim rods and installation of dummy rods, shielding fabrication of several types, and fabrication of many cleaning and remote handling tools. A total of 2,372 craft manhours was utilized during the November outage as follows:

FERTF Fuel Failure	980
Decontamination	570
Repair	538
Modification and Improvement (Mainly Capital Work)	657
Operations Assistance	200
Preventive Maintenance	<u>625</u>
	<u>2,372</u> manhours

FAST FLUX TEST FACILITY PROJECTConceptual Design

Conceptual design studies are being actively pursued in reactor vessel arrangement, heat removal systems optimization, instrumentation, and building arrangement. The text of Volume 2, BNWL-CC-175, and various drawings were revised to conform to the most recent concept.

A survey was begun to ascertain the advantages (and disadvantages) of decreasing the core lattice size to achieve a more subdivided core assembly of more numerous fuel positions.

The FFTF design and construction project was cost-estimated by Bechtel Corporation with the assistance of Atomics International. The Bechtel-AI estimate was quite similar to the budget cost estimate for the same work scope prepared at Hanford by BNWL-FFTF and Vitro personnel.

A general method has been devised for optimizing the design of the FFTF. Relative values are estimated for the useful range of each irradiation testing variable. Trade-off factors are established, values are normalized, and a composite value function is derived.

Analyses were conducted to explore the costs of various power levels and fuel configurations for the FTR cermet-fueled system. Optimum fuel cycle costs were found to lie in the range of 250 to 400 MW.

The two-dimensional DDK transport theory code was used to calculate the reactivity for the Fermi Core B critical assembly using the RUSS II cross section set.

The methods and models to be employed in calculating the energy release during a maximum credible accident in the FTR were discussed with Nuclear Technology Corporation personnel. A program was outlined to evaluate the presently available computer codes and analytical models.

Preliminary estimates have been made of the buildup of Pu^{238} in cermet driver fuel ($\text{PuO}_2\text{-SS}$) and in ceramic driver fuel ($\text{PuO}_2\text{-U}^{238}\text{O}_2$) in a 400 MW, 600 liter FTR reference core.

The reactor heat removal system has been divided into two segments: (1) heat removal and (2) heat dissipation. Four leading concepts for the heat removal system are under detailed study for a reactor power level in the range of 400 MW.

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Atomics International has accelerated work on alternate fuel handling machine configurations in an attempt to reduce the height of the machine by devising new fuel transfer techniques that will require a shorter reactor tube.

Preliminary engineering studies were continued to define the requirements for the prototypical sodium facilities. A suitable location for this complex appears to be the basement of the 324 Building.

Design Analysis

The SEAL-SHELL 2 code for thermal and mechanical loadings of axisymmetric vessels is now operating on the 1107 computer.

After a review of experimental work by the Naval Ordnance Laboratory for Enrico Fermi reactor, the present FFTF design is considered to have more than adequate protection against any initial shock wave (microsecond phenomena). This protection is provided by the multiple layers of steel cylinders and piping around the core.

TIGER 5, a transient thermal digital computer program, has been modified to calculate temperature profiles in an FFTF fuel bundle. A secondary program has been added to TIGER 5 which calculates nodal heat generation rates based on any radial and axial flux profile in the core.

The hydraulic characteristics of the fuel element (conceptual design) were analyzed. For a sodium flow of 57 lbm/sec, the fuel element pressure loss will be between 100 and 130 psi. The leakage flow between channels for an annulus pressure loss of 70 psi amounts to 6.4×10^{-3} percent of the annulus flow. The grid spacer strips will withstand the forces due to a sodium velocity of 50 fps, with a large safety factor.

Work has begun on the detailed analysis of the temperature profiles of single FFTF fuel pins under steady state and transient conditions. Due to the tricuspid shape of the coolant passages in a rod bundle, the surface heat transfer coefficient and the coolant sink capacity vary circumferentially around a fuel pin.

Development of the FFTF dynamics model continued in the areas of reactor and primary loop components. The responses of logarithmic and linear, step-wise reactor startup models to selected input disturbances were compared.

Program Control Management

Several PERT computer programs were located and a test run was made on each of the following: NASA PERT "Charlie", NASA PERT Time I, and USAF PERT/Cost. The NASA programs ran satisfactorily but the USAF PERT/Cost program failed to run due to memory restrictions in the available computer systems.

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ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTPLUTONIUM RECYCLE PROGRAMPlutonium Fuels DevelopmentPRTR-HPD Fabrication

Since the start of fabrication on June 8, 1965, a total of 34 Mark I-R series fuel elements have been assembled. Seven of these elements were assembled during the month of November. Twenty-eight more UO_2 - PuO_2 elements and one ThO_2 - PuO_2 element are scheduled for completion by December 31, 1965, in order to meet the batch core loading requirement for the Molten Core Experiment in the PRTR.

Except for about 50 plenum chamber springs, enough hardware has been accumulated to meet the December 31 schedule. The plenum chamber springs are expected to be delivered by November 29, 1965.

Installation of the post-decontamination fuel rod dryer has increased the loading rate from 20 rods per day to 20 to 30 rods per day. About 275 loaded fuel rods are in process and about 300 rods need to be fabricated. At the present loading rate, it is expected that the December 31 schedule will be met.

Pneumatic Impaction of Mixed Oxide

The pneumatic impaction of the HPD core loading for the PRTR has been essentially completed. A few kilograms of recycle material will be processed in early December as it becomes available. This concludes a program started in the early part of the summer wherein the total amount of material scheduled for processing was a little over two tons. The impacted material has been generally of excellent quality and presented little difficulty in loading by vibrational compaction in the fuel rods.

Zirconium

All tubing required to meet current PRTR fuel fabrication schedules has been received during this past month. All certifications have been received and are available.

PRTR High Power Density Core

Postirradiation examination of three fuel rods from the High Power Density Fuel Element 6504 containing UO_2 -4 wt% PuO_2 is in progress. This examination will assist in determining the mechanism by which failure occurred during irradiation in the FERTF. The three rods being examined include the ruptured rod, a non-defected rod adjacent to the rupture (FB-00), and a non-defected replaceable rod (FB-10).

A piece of cladding approximately 3 inches long and almost 1/2-inch wide at the widest point is missing from the ruptured rod. The missing clad apparently contains the 1/16-inch diameter hole originally drilled to defect the rod. There is visual evidence of localized overheating of the cladding in the rupture region, thus indicating that failure was due to burnout. Except for the ruptured region, there were essentially no diameter changes along the length of the rod. More than 700 grams of UO_2 - PuO_2 fuel material, or about 39 per cent of the total amount in the rod is missing. In the immediate vicinity of the rupture, approximately 5 inches of fuel is completely missing.

Examination of the fuel below the rupture shows the presence of once-molten fuel, a central void, and the expected fuel structures in the graded enrichment region near the bottom end cap. Examination of the fuel above the rupture, however, shows a considerably larger central void than expected and indicates that the once-molten fuel is missing completely. The large central void extends uninterrupted over the entire fuel region above the rupture. In fact, the void extends through the depleted pellet at the top end of the fuel column and into the plenum region. Apparently, the depleted pellet has been eroded by the molten fuel column. Some fuel material has moved into the gas plenum, probably by a vapor deposition mechanism. The UO_2 -1 wt% PuO_2 fuel material used as part of the graded enrichment seems to have become mixed into the UO_2 -4 wt% PuO_2 fuel material, perhaps causing higher than expected fuel temperatures in the top end region.

Preliminary examination of the Zircaloy cladding in the rupture region shows that only the material near the break was affected. A thick oxide layer has formed on the cladding near the break and, in one instance, on the fractured surface indicating the clad was fractured in this region before the process tube rupture occurred. The cladding has not yet been examined to determine the extent of hydriding.

Examination of the rod adjacent to the ruptured rod indicates that it was unaffected by the presence of the rupture. There were no dimensional changes in the rod and gas release (217 ml) was as

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expected. The expected fuel structures were formed at the maximum power generation of 27 Kw/ft, i.e., a central void; melting to 50 per cent of the radius; and columnar grain growth, representing fuel temperatures in excess of 1800°C, to 80 per cent of the radius in the plane of the rupture.

There were no dimensional changes in the non-defected replaceable rod and no measurable change in the balance point, thus indicating that the operating conditions did not cause swelling or gross axial fuel movement.

The fuel structures formed in the ruptured rod and the non-defected rods indicate that, for some unknown reason, the ruptured rod operated at a higher fuel temperature. It appears that the molten fuel may have been rapidly ejected from that portion of the fuel rod above the rupture when the pressure tube failed and that the gas plenum may have played a role in the failure mechanism. Whether the localized overheating of the cladding in the ruptured region was caused by classical boiling burnout or by contacting molten fuel is not known at this time. Localized overheating as a result of rod swelling and flow blockage would cause rapid deterioration of the cladding by hydriding. Although the precise failure mechanism is still unknown, the rupture appears to be related to the defect in the rod and not to molten fuel operation per se.

Evaluation of Coolant Conditions for Element 6504

Establishment of environmental conditions of temperature, time, flow, pressure, pH, oxygen content and conductivity of water coolant was completed for loop testing of a defected fuel rod similar to the UO_2 -4 wt% PuO_2 HPD element (6504) tested recently in the FERTF. The ex-reactor test is designed to simulate coolant conditions of FE-6504. Examination of the ex-reactor rod will be made at the end of the test for possible failure mechanisms such as oxidation-swelling of fuel, water logging of fuel, or water-vapor trapping in gas plenum.

TREAT Experiments in Support of PRTR High Power Density Core

A joint program has been initiated between Battelle-Northwest and Argonne National Laboratory for the transient irradiation of UO_2 capsules in TREAT. The irradiations will furnish information of interest to the Reactor Safety Program at ANL as well as needed data directly related to the safeguards analysis of the Molten Core Experiment in PRTR. The capsules will be fabricated by BNW, irradiated in TREAT by ANL, and jointly evaluated by BNW and ANL.

Two transient irradiations, each involving four capsules, will be performed in TREAT during the first part of January. The 1/2-inch diameter Zircaloy-clad capsules contain a vibrationally-compacted impacted UO_2 (21-23 per cent TD) fuel column 4.2-inches long capped on each end by a depleted UO_2 pellet 0.22-inches long.

The UO_2 fuel contains 15 per cent U^{235} . There is a 2-inch long gas plenum at one end of the fuel column to simulate the PBTR fuel design. The four capsules included in one of the transient irradiations will contain He at atmospheric pressure. The four capsules in the second irradiation will be pressurized with He at 1500 psi. to approximate the conditions that exist in a high burnup fuel element. The capsules are to be contained in a water-filled autoclave and subjected to a transient burst of neutrons in TREAT. A transient energy of 460 cal/gm of UO_2 will guarantee melting of the UO_2 , reaction with the Zircaloy cladding, and fragmentation of the capsules. From past experience, ANL has found that 280 cal/gm melts UO_2 .

After the irradiation, the hydrogen evolved (measured by mass spectrometer) provides a means of computing the extent of clad-water and/or UO_2 -water reaction which occurred. The particle size distribution of the residue from the fragmented capsules will indicate the nature of the UO_2 water reaction. These data will help evaluate the consequences of injecting molten UO_2 - PuO_2 fuel into the water coolant in the event of a catastrophic fuel element failure.

Shippingport Fuel Elements

Three, 120 rod bundles from Shippingport Fuel Element 045 are being brought to the Radiometallurgical Facility for disassembly. These are the first three bundles to be analyzed for the high exposure plutonium radiation measurements program.

Specific fuel rods will be removed from each of the bundles and plutonium extracted for isotopic analysis and Beta-Gamma Dosimetry Studies. The exposure of these rods will vary from 13,000 MWD/TU to 34,000 MWD/TU.

High Pressure Encapsulation

As part of the TREAT irradiation tests in support of the PBTR Molten Core Experiment it will be necessary to irradiate pressurized capsules.

A method to effect a high integrity seal on capsules containing gas pressure of 1500 psig was investigated. Two approaches were used to effect the seal on a 1/8" diameter x 1/32" wall helium pressurized Zircaloy tube. The first approach consisted of effecting a temporary mechanical seal by compressing the tube wall onto a specially designed mandrel, cutting off the feed end of the tube, and forming a fusion weld as the final seal. This approach was successful; however, fine finish on the mandrel and tube inside diameter is required. The second approach consisted of mashing the pressurized tube flat with electrodes and effecting a spot weld, in a single operation, as the final closure. This approach was successful and will be applied to current TREAT (Transient Reactor Experiment) capsule fabrication because of its apparent advantages in part preparation (finish and cost). This method looks applicable to many types of encapsulation in which the gas or pressure (or both) make fusion welding impractical.

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Fuel Cycle Economics

The conceptual design of the 1000 kg per day plant for mixed oxide pellet production has been completed and the information fed into the computer program to calculate fuel costs. A summary report of this study and the one on vibratory compaction is being written.

Corrosion and Water Quality StudiesPRTR Pressure Tube Evaluation

The PRTR reactor safeguards program includes study by destructive examination of the effect of reactor environment upon the Zircaloy-2 pressure tubes. Work in progress is composed of measurement of elastic strain on unirradiated tube specimens while under different loading conditions, time-temperature effects on irradiated tube material, and crack propagation tests at various temperatures. The respective objectives are to describe elastic behavior of an anisotropic tube, to measure the effects of annealing on as-irradiated strength for reference to stress rupture testing, and to provide crack propagation data. This work is continuously improving the basis for judgment of irradiated specimens.

An annealing-aging reaction was indicated by the data obtained from tensile testing of ring specimens after they were held at 550°F (290°C) for one, two, three, and four thousand hours. The section of PRTR tube from which these specimens were cut had received 5×10^{20} fast nvt at a temperature between 185°F and 276°F (85 and 180°C). The first 1000 hours at 550°F (290°C) produced some annealing of irradiation damage which was indicated by a decrease in tensile strength. At greater than 1000 hours, aging of the alloy was indicated by an increase of strength. At 4000 hours the strength was greater than at the start of the test.

These data suggest that stress rupture testing at 550°F (290°C) may be complicated by opposing reactions, strength reduction from irradiation damage annealing and strength enhancement from aging.

ID Measuring Gage

The ID gage for measuring simultaneously six PRTR tube diameters has been used to monitor the full length tube in the EDEL-1 loop in the 314 Building. A complete analysis of the data has not been completed; however, there appears to be fair correlation between repeated traverses through this tube. Evaluation is continuing.

A gage was fabricated for measuring the crack tip opening displacement (CTOD). This gage was attached to a section of PRTR tubing with a machined slot. The curve of CTOD versus internal pressure was obtained. The gage functioned well but was destroyed by the water force when the tube failed. Additional gages are being fabricated for continued testing.

The failed rupture loop process tube from the PRTR was measured with a circumference tape to determine the amount of straining that had occurred during reactor operation. The average tube diameter was plotted versus tube length and clearly showed a 0.004" diameter increase within the core section. It is planned to examine about thirty additional PRTR tubes when they are removed during the conversion to the short core. These measurements are made in the PRTR storage basin using a "pi" tape and telescope.

Crevice Corrosion in Lithiated Water

The out-of-reactor test program to investigate crevice corrosion of Zircaloy-2 in 293°C, pH 10 lithiated water was terminated because of insufficient funds for test facility operation. Tests consistently resulted in accelerated attack in crevice regions of electrically heated test specimens. The attack was attributed primarily to the concentration of lithia, by boiling, in the restricted flow area of the heated crevice regions.

As indicated in the report for October 1965, visual examination had revealed severe attack on the last two specimens tested (800,000 Btu/hr-ft² surface heat flux specimens with exposures of only 36 and 102 hours). Metallographic examination showed that corrosion product deposits extended halfway through the 0.030 inch cladding (where bands had formed heated crevices) of the assembly with the 36-hour exposure. Maximum cladding penetration found in crevice regions of the assembly with the 102-hour exposure was 0.004 inch; the cladding pin-hole evident in one of the corroded regions of this assembly was the result of heating element failure.

The 0.015 inch cladding penetration was the maximum observed in the test program, but metallographic examination also verified the presence of breakaway corrosion on heated surfaces away from crevice regions. In general, penetration away from crevice regions was much less than that in the crevice regions, but in prior tests with lower heat fluxes, corrosion was limited solely to crevice regions.

Chemical Shim Studies - Corrosion and Water Quality

The presence of even small amounts of fuel debris in the moderator system when operation is resumed would cause the H₃BO₃ control ion exchangers to become grossly radioactive. It will therefore be necessary to relocate the existing anion exchange units inside the shielded ion exchange vault and to consolidate the system into a single unit to conserve space and to simplify control functions. A single exchanger would be satisfactory if it is sized to remove the required amount of boron and is designed to operate effectively over the relatively wide range of flow rates that will be encountered. The size of the exchanger is determined primarily by the final permissible boron concentration in the moderator but is also affected by the presence of other impurities in the moderator. Calculations based on anticipated operating conditions during the batch-core experiment indicate the following relationship between required exchanger volume and permissible final boron concentration in the moderator.

<u>Final Boron</u> (ppm)	<u>Ion Exchanger Volume</u> (ft ³)
0.5	79
1.0	40
1.5	26

The over-all control system design is currently being reviewed to determine the optimum exchanger size with respect to possible modes of operation.

Work is currently in progress to procure an automatic boron concentration meter for use at PRTR. There has been considerable difficulty in getting the meter manufacturer to agree to furnish a process instrument that will meet PRTR performance requirements. The major obstacles have involved sample handling, selection, and identification procedures and equipment. It now appears that the manufacturer will not supply the complete analysis system--just the analyzer portion. The analyzer will then have to be assembled into an integrated sampling and analysis system after it arrives on-site.

General Corrosion Studies

Recommendations were prepared outlining water treatment procedures for the various PRTR cooling systems. No significant changes from normal conditions are required in any system except the moderator and dry gas systems while cleanup operations following the FERTF failure continue and nonstandard conditions are in effect. At the present time drain-flush operations to remove fuel debris and fission products have top priority in these two systems. This work is proceeding using deionized and degassed flush water with an air atmosphere present in both the aluminum moderator system and the carbon steel dry gas system. An inert gas atmosphere would be more desirable in the dry gas system; however, the cost of providing such a system and the probable inability to maintain effective gas blanketing during cleanup and maintenance operations indicated the possible benefits did not warrant the efforts involved to provide for blanketing. Corrosion inhibitor addition to the systems and/or the flush water has not been employed to date pending evaluation of the desirability of this action. At present there are no plans to add inhibitors until cleanup operations are completed, and the systems are placed in wet layup.

One of the potentially severe corrosion problems exists in the shim recess wells if crevices and galvanic couples are formed between the aluminum shim wells and the Inconel rods. To minimize this problem, 14 of the shim assemblies have been removed from the reactor. Of the four remaining assemblies, it appears that three are located in the recess wells. These shims are presently inoperative so they cannot be removed from the wells. These assemblies are scheduled for removal at the earliest possible opportunity.

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Some of the PRTR cooling systems utilize process water as the coolant. Pressure drops have gradually increased in these systems due to corrosion and scaling of the carbon steel piping until now it is difficult to maintain the required coolant flow rate. To remove the accumulated debris obstructing these cooling lines, it was recommended that mechanical procedures be tried first followed by chemical cleaning if necessary. Mechanical procedures under consideration include air sparging and possibly addition and circulation of an abrasive material (e.g., diatomaceous earth). No decision has yet been made concerning the recommended chemical cleaning procedure pending analysis of mechanical cleaning results.

FERTF Cleanup Studies

Drain-flush operations have been moderately successful in reducing activity levels at PRTR moderator and dry gas systems. Although some additional benefits could probably be obtained by continuing these operations, it is apparent that they will not be completely effective in removing the fuel debris. Work is therefore in progress to evaluate methods of increasing the efficiency of these operations and to devise alternate methods for activity reduction.

Vibration is one possible method for increasing flushing efficiency. Vibrations can be initiated mechanically in the system materials, or electrically to produce ultrasonic vibrations in the fluid. One exploratory test was performed in the PRTR Pressure Tube Mockup (EDEL-1) to determine whether vibration would be effective in assisting flushing operations in the bottom shield-inlet bellows assembly region. The test was performed by pouring about 50 cm³ of powdered Fe₃O₄ (magnetite) into this region and then flushing with and without vibrating the assembly to check Fe₃O₄ removal. Only about 10% of the Fe₃O₄ was removed by plain water flushing, while an additional 50% was removed after 15 minutes of vibration at 20 cps.

Flotation is a candidate method being evaluated for removing particulate fuel debris. The flotation studies are being conducted with particulate Fe₃O₄ (sp.gr. ~6); if a suitable process is found for Fe₃O₄, work will then begin with UO₂-PuO₂ powders (sp.gr. ~11). To date the flotation studies have not been successful since the particles are simply too dense to remain suspended in foam-water systems. Work is continuing in this area to try to find additives that will promote additional bubble formation on particle surfaces. Recently work was begun on methods for entrapping particulate matter in chemical adhesives that can in turn be removed by flotation or other means. An example of the type of process under study involves addition of a water soluble polymer to an organic system (e.g., alcohol or acetone) to cause further polymerization. Loose particulate matter present in the organic system can be entrapped during the polymerization process. If the polymer can then be removed by flotation, flushing, or other means, the loose activity can be removed. Efforts to date are exploratory, and no conclusions can be drawn yet on whether the process has any definite potential for use at PRTR. Work will continue in this area.

Evaluation of Reagents for Dissolving Ceramic Core Material

Since the PRTR gas and moderator systems have been found to be grossly contaminated with fuel element rupture debris, efforts are being made to adapt ceramic fuel dissolution processes for use in systems containing aluminum and carbon steel. The first phase of this adaptation work was centered on the $\text{H}_2\text{SO}_4\text{-HNO}_3$ process. According to the literature, mixtures of sulfuric acid, nitric acid and water, where either the nitric acid or sulfuric acid concentrations are high or low, cause only mild corrosion on 2S aluminum at room temperature. At elevated temperatures ($>100^\circ\text{C}$) laboratory tests, however, showed a very aggressive attack on aluminum. Results of these tests follow:

<u>Composition</u>	<u>Temp.</u>	<u>Reaction</u>
55 vol% conc H_2SO_4 -5 vol% conc HNO_3	137°C	Totally dissolved 1100 Al in 24 hrs.
95 vol% conc HNO_3 -5 vol% conc H_2SO_4	115°C	Totally dissolved 1100 Al in 24 hrs.
40% H_2SO_4 - 20% HNO_3	120°C	Totally dissolved 1100 Al in 24 hrs.

In all cases carbon steel was dissolved by these solutions.

The second phase of this work was directed toward the removal of ceramic fuel debris in two steps: (1) the dissolution of UO_2 , and (2) the dissolution of the residual PuO_2 . Such an approach would permit the possible use of HNO_3 and H_2SO_4 successively to effect $\text{UO}_2\text{-PuO}_2$ removal. Very concentrated HNO_3 is not unduly corrosive to aluminum according to the literature. Corrosion scanning tests with concentrated H_2SO_4 at 100 to 200°C show that corrosion losses are less than 0.5 mil/day on 1100 aluminum alloy.

Further tests are being run to more fully define the corrosion behavior of ceramic fuel dissolution reagents on gas and moderator system components.

PRTR Decontamination Studies

Recently the G7/M3 Loop in the ETR at NRTS was successfully decontaminated using the process that has been recommended for PRTR primary system decontamination. Following 2-hour treatments with alkaline permanganate at 230°F (110°C) and an oxalic acid-ammonium citrate solution at 180°F (82°C), efficiency measurements on the loop showed a decontamination factor ranging from 2.3 on pressurizer heaters to 23.2 on a temperature control valve. The over-all loop decontamination factor was 12.6.

~~NO USE ONLY~~

Reactor Components DevelopmentShroud Tube Removal

Since the FERTF incident at the PRTR apparently did not rupture the shroud tube, completion of the removal machine is continuing with reduced emphasis. Electrical and mechanical assembly have been essentially completed. Preliminary testing of the rotational drive, translational drive, and cutter drive has been completed. Some difficulty has been encountered on the stepping motor to cutter feed drive system. The problem is apparently due to voltage drop in the wiring from the pulse translator to the stepping motor. Remedial action is pending completion of further investigation.

The six-inch weld cutting head has been calibrated and installed, and has satisfactorily cut a tube which was installed on the top of the mockup. This test was not fully prototypical because the spindle was withdrawn from, and clamped to the top of the housing so that the cutter could work on top of the mockup where it could be observed. However, the test did demonstrate the ability of the cutter to machine aluminum at design feeds and speeds.

Fabrication of parts for the pad cutting version of the machine and for the 4-inch shroud removal equipment is proceeding on a low priority basis.

Development of a method to replace PRTR shroud tubes is also continuing with reduced priority. Prints for two insertion concepts are complete, and layouts for three other concepts are done. Measuring equipment is designed and built. Bench tests confirm that rolling is acceptable for the bottom but unacceptable for the top of the shroud. Clearances using the chill-expanding method continue to be less than expected, resulting in problems in both handling and sealing. Tight joints were obtained with wedge sealing, but reactor insertion problems remain unsolved. Bench tests are continuing with modifications of the above tests and tests on new concepts planned.

Design Analysis

The recent failure of the PRTR rupture loop pressure tube prompted a more detailed analysis of the power distribution and temperature profiles in the fuel bundle for operating conditions at the time of the failure. This analysis, which included determination of surface heat flux and molten fuel regions, has been completed.

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The fuel pins in the rupture loop at the time of failure were composed of $\text{UO}_2 + 4 \text{ wt\% PuO}_2$ (85% theoretical density), having 1 wt% PuO_2 and depleted uranium at the rod ends. Tube power was 1780 kw, coolant outlet temperature was 490°F (255°C), and the flow rate was 67,000 lbs/hr. Estimated average rod power for an outside rod was 24 kw/ft, 11.7 kw/ft for a rod in the 6-rod ring, and 9.7 kw/ft for the center rod. The axial power peaking factor was about 1.165, the maximum being near the midpoint of the rod.

The steady state heat transfer program (STHTP) was used for calculating the molten fuel zone and fuel surface heat flux. Results were obtained for each fuel rod in the bundle and at several axial locations. In this manner the complete molten region was established. Calculations revealed that molten fuel would be present only in the outer rods. The maximum molten radius will extend to about $2/3$ of the fuel radius in a direction toward the side of the fuel rod nearest the pressure tube. On the opposite side of the fuel rod, toward the bundle center, the molten zone will extend to about $1/2$ of the fuel radius. The beginning of melting at the entrance end of the fuel rod is difficult to define because of the end effect. It is estimated, however, that the molten zone will extend to within an inch or so of the depleted uranium zone. Calculations show that the molten zone will extend to about $8/10$ of the fuel rod length.

NUCLEAR CERAMICS RESEARCH

Electrical Resistance of UO_2

The band gap for UO_2 determined by electrical resistivity measurements was estimated to be 3.0 eV. Electrical resistivities of three UO_2 single crystal specimens measured by three separate experimental procedures agree within experimental error. The activation energy for the intrinsic range ($> 1000^\circ\text{C}$) appeared to change slightly with temperature. Variations between specimens in the low temperature extrinsic ranges ($< 1000^\circ\text{C}$) are attributed to differences in impurity concentrations.

Phase Studies

Measurements of the oxygen dissociation pressures over americium sub-oxides were completed during the past month. The total temperature and pressure exchange interval studied now extends from 800° to 1100° and from below 10^{-6} atm to 1 atm of oxygen. AmO_2 is quite unstable with respect to reduction relative to PuO_2 . For example in 1 atm oxygen and at 1100°C , a composition of $\text{AmO}_{1.98}$ is attained. High equilibrium oxygen pressures were obtained both through use of the cupric oxide reduction equilibrium and measured physical additions of oxygen. The results, using both methods of oxygen addition, were in good agreement. Heat and entropy of solution values were in accord with earlier results.

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Electron Microscopy Phase Studies

In-microscope studies of the phase changes of PuO_{2-x} were initiated in August. Interpretation of the results was made difficult because of oxidation of the samples during the observations. The oxidation was due primarily to leaks in the special equipment needed for the experiment. These difficulties have been corrected, and arrangements have been made to allow H_2 to flow over the surface of the specimen during the experiments. Further investigations should be underway within a few weeks.

Hypostoichiometric Mixed Uranium-Plutonium Oxides

Hypostoichiometric mixed uranium-plutonium oxides are being synthesized to investigate the effects of stoichiometry on various properties pertinent to reactor fuels applications. The oxygen to metal ratio is varied by PuF_3 additions to PuO_2 - UO_2 mixtures. A constant plutonium-uranium metal ratio of 1:4 is maintained at each oxygen to metal ratio, which is varied between $\text{O/M} = 1.75$ -2.00. The raw materials are ground and mixed in a ball mill for 24 hours, followed by pressing and sintering operations. The pressed pellets are sintered at 1700°C for several hours in a dry nitrogen - 5% hydrogen atmosphere.

Effects of Oxygen Partial-Pressure on Oxide Melting Points

Interpretation of fuel heat transfer capabilities by correlating melt radii with heat ratings requires knowledge of the melting points of the materials under study. These data have been obtained for the fissile oxides only under inert or reducing atmospheres which promote dissociation to lower oxygen/metal ratios than are typical of the same material in the closed system represented by a fuel rod. Equipment and techniques for measuring melting temperatures and equilibrium compositions for hyperstoichiometric and stoichiometric ceramics, i.e. UO_{2+x} and $\text{PuO}_{2.0}$ are being developed. The basic apparatus consists of a 10 mc, 25 kva induction generator and a ThO_2 susceptor tube furnace. The ThO_2 furnace tube is 1/2 inch ID and is segmented into 3/4 inch lengths to prevent severe longitudinal cracking. The tube is surrounded by ThO_2 powder for thermal insulation and an outer quartz jacket. An automated hearth moves a specimen into or out of the vertical heat zone from below. Ignition of the furnace can be achieved most simply by inserting a SiC rod into the heat zone and preheating the thorium until it becomes sufficiently warm to couple to the 10 mc field (1400 - 1700°C). The SiC rod is then removed. The facility will operate in any atmosphere and should be capable of temperatures to approximately 3000°C (ThO_2 melts at 3400°C).

Stoichiometry Gradients in Oxide Fuels

Analytical studies indicate that radial stoichiometry gradients established during irradiation can account for the discrepancy between ex-reactor measurements of UO_2 thermal conductivity and the lower conductivity values deduced from observations of melt and grain growth radii in irradiated fuel rods. These gradients indirectly affect other temperature dependent fuel characteristics such as fission gas retention and swelling.

In-reactor studies were initiated to determine the effects of varying bulk fuel stoichiometry and heat rating on post-irradiation oxygen distribution. Several 1.373 inch diameter uranium oxide fuel cores with O/U ratios from 2.00 to 2.25 will be irradiated for short times (one day maximum) at heat ratings sufficient to cause center melting. Following irradiation each core will be sectioned and the extent of grain growth and center melting correlated with initial stoichiometry. Diamond core drills will be employed to remove samples from selected radial positions in the irradiated cross sections. These will be analysed coulometrically for O/U ratio and the results correlated with fuel heat rating.

The initial experiment includes eight specimen capsules fueled with UO_{2+x} ($0 \leq x \leq 0.25$) and eight standard capsules fueled with sintered $\text{UO}_{2.00}$. One specimen and one standard will be irradiated simultaneously in a duplex holder. Each capsule will operate at 30 to 35 kw/ft (which will insure center melting) for twelve hours. The sequence of eight such irradiations will be performed in as short a time as is practicable; hopefully during a single reactor cycle.

Techniques for fabricating integrally clad UO_{2+x} specimens were developed. Pneumatic impaction was selected as the only available process which retains the desired high O/U ratios in a high density, regularly shaped body. Containers were designed, experimentally tested, and ordered from an off-site vendor. Specimen fabrication, assembly, and irradiation are scheduled for December.

A more simple, qualitative technique for determining UO_2 stoichiometry is to observe the color of the streak obtained by abrading the oxide with a porous ceramic plate. As the O/U ratio increases above 2.00, the streak color changes from red-brown to darker shades until it becomes black at a composition of approximately $\text{UO}_{2.05}$. To apply this method to irradiated UO_2 fuels, low density HgO rods of the same diameter as the fuel were prepared with flat, ground ends. These rods will be rotated against the polished surface of cross-sectioned fuel cores to produce a circular streak pattern. Radial color gradients in this pattern will be indicative of corresponding stoichiometry gradients.

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Mixed Nitride Synthesis by Carbothermic Reduction of Mixed Oxides

Further attempts have been made to synthesize a mixed U, Pu nitride by means of carbothermic reduction of mixed oxides. In one case, involving temperatures to 1800°C and static atmospheres of nitrogen and hydrogen, nearly complete conversion was attained. An X-ray diffraction pattern of the material indicated the presence of a nitride solid solution with but faint indication of an oxide phase. Unfortunately, this material was contaminated with tantalum hydride during final heat treatments and is of little use for further analysis. Use of nitrogen hydrogen mixtures and of pure hydrogen atmospheres during heat treatment have led to reductions in carbon content from as much as 4% excess to approximately 1000 ppm.

The following conclusion may be drawn on the basis of these preliminary experiments: significant conversion of mixed oxide to solid solution nitride can be obtained through carbothermic reduction, even in static atmospheres, and excess carbon, which may be initially desirable to cause complete reduction, may be removed by appropriate treatment with hydrogen. A more systematic and detailed investigation of this synthesis scheme appears to be justified.

Plutonium Development ProgramPlutonium Ceramics Irradiations

Irradiation of wafer specimens of various plutonium compounds continued. Status of specimens currently in-reactor is summarized below.

Current Plutonium Ceramics Irradiations

<u>GEH-14-</u>	<u>Material</u>	<u>Goal Exposure, Fissions/ ccx10⁻²⁰</u>	<u>Exposure, Fissions/ccx10⁻²⁰</u>		<u>Projected Discharge Date</u>
			<u>During 11/65</u>	<u>Cum. to 12/65</u>	
405	PuN, impacted	172.5	4.5	106.0	5/67
407	PuN, impacted	77.5	2.0	47.3	5/67
409	PuC, arc cast	150.0	2.7	81.3	1/68
411	PuO ₂ , 95% TD sintered	166.0	4.0	109.5	1/67
413	Pu ₂ O ₃ , 95% TD sintered	116.0	2.8	78.3	1/67
538	PuN-50 vol% W, impacted	124.0	2.8	65.8	8/67
540	PuN-50 vol% W, impacted	62.5	1.6	37.8	8/67
549	PuN-24 vol% Pu, arc cast	67.0	1.6	37.8	8/67
551	PuN-15 vol% Pu, arc cast	32.0	1.0	17.5	8/67

Ceramic Bonding During Impaction

A study proposal for effects of high energy impaction of UO_2 was prepared by J. L. Daniel and B. Ishii of Mitsubishi Atomic Power Industries. The objectives of this proposal are to: a) study the bonding mechanism of UO_2 during compaction of powders, b) study diffusion characteristics of samples of different O:U ratios, and between UO_2 and other additives, c) evaluate the effects of annealing and impaction temperatures, and d) identify the significant processing variables. Results of this study will be compared with similar studies on hot swaging and isostatic pressing to be conducted by Mitsubishi.

Ceramic Pneumatic Impaction Molds

Several Al_2O_3 and ZrO_2 pneumatic impaction molds were prepared. These molds have been loaded with metal powders to evaluate their potential in shape forming with pneumatic impaction. Potential advantages for using ceramic molds as opposed to metal molds are:

- a. rapid and economical means of preparing shapes
- b. reduced heat losses during pneumatic impaction
- c. impacted assemblies can be quickly removed by fracturing the unbonded ceramic mold.

Trial impactions are being performed.

Materials and Information Exchange

UO_2 single crystal specimens are being prepared to meet requests from both foreign and domestic sites including: Mitsubishi Atomic Laboratory, Japan; The Hebrew University of Jerusalem, Israel; Harwell, England; and ORNL. In addition, a request was completed to obtain PuO_2 spheres (sol-gel) from ORNL.

BASIC SWELLING STUDIES

The purpose of this program is to characterize fissionable materials and understand their behavior during irradiation. The theories developed provide a basis for the engineering exploitation of fissionable fuel materials in advanced reactor applications. Specimens under study at the present time include uranium with and without dilute alloy additions.

Irradiation Program

Two controlled temperature-pressure capsules and a controlled temperature (low pressure) capsule reached their goal exposure and were discharged. The former were irradiated in tandem (both in a single test hole) at respective control temperatures of 550°C and 625°C and at a common pressure of 500 psi to a burnup of 0.16 at.%. The latter was controlled at 700°C and irradiated to a burnup of 0.09 at.%. All three capsules contained two types of high purity uranium (6 ppm C and 484 ppm C), British "adjusted" uranium and a U + Fe - Si alloy. Several different heat treatments were employed for each composition. The pressure capsules complete the 500 psi series. The 700°C irradiation will provide additional extremely valuable information on the effect of burnup in the beta phase and the effect of minor alloying additions on fuel stability at this high temperature.

A controlled temperature-controlled pressure capsule and a controlled temperature capsule are under construction for long-term irradiation. The high pressure capsule contains high purity uranium and alloy material whereas the low pressure capsule contains only alloy specimens. The alloy specimens were obtained from several sources and their performance will be compared directly.

Postirradiation Examination

Metallography is in progress on several high purity uranium specimens recovered from Capsule 24 which was controlled at 625°C and irradiated to 0.13 at.% B.U.

Analysis of the data obtained from uranium specimens irradiated in the series of three capsules controlled at 1000 psi and at 450°C, 575°C, and 625°C, respectively, shows that pressure reduces the amount of swelling by about a factor of 5 in all materials. Two types of high purity uranium were employed in this series of capsules in addition to a U + Fe - Al - Si alloy and a U + Fe - Si alloy. One type of high purity uranium (T-3) contained <10 ppm C while the other type (PT3-NA) contained about 480 ppm C. Other impurities were low and at about the same level for both types of high purity uranium. The two alloy materials each contained 400 - 500 ppm C. The high carbon - high purity uranium (PT3-NA) swelled less than the low carbon - high purity uranium.

(T-3) by about a factor of three. Furthermore, the PT3-NA material behaved identically to the U + Fe - Si alloy at 1000 psi. It would appear, therefore, that carbon is a much more important additive than the 140 ppm Fe - 95 ppm Si employed in this alloy. The U + Fe - Al - Si alloy (400 ppm Fe, 640 ppm Al, 85 Si) was definitely superior to the other materials indicating that positive benefits are derived from the higher Fe and Al additions. Low pressure data has not yet been obtained with the high carbon - high purity uranium. The 1000 psi capsule controlled at 450°C contained no low carbon - high purity uranium. Data at this pressure are needed and will be obtained for low carbon - high purity uranium.

Capsule P-5 controlled at 450°C and 500 psi during irradiation to 0.16 at.% B.U. in the specimens was opened in Radiometallurgy. Examination of the specimens has been initiated. Sample distortion was severe due to irradiation growth. Density measurements were difficult to make because of extensive surface roughening. Preliminary observations indicate that 500 psi was not particularly effective in reducing the swelling that occurs at low pressures. This is strong evidence supporting the previously advanced concepts that mechanical processes rather than fission gas agglomeration are responsible for swelling in uranium and its alloys. If fission gases were responsible for swelling, then swelling behavior should be a relatively simple, inverse function of pressure.

IRRADIATION DAMAGE TO REACTOR METALSAlloy Selection

Two new nickel-base alloys are being evaluated to determine the effects of irradiation upon their mechanical properties. Specimens of PDRL-120 and low boron Hastelloy X-280 materials are presently being fabricated. These specimens will be irradiated in the ETR G-7 loop facility to various exposures up to 1×10^{21} nvt.

Specimens of irradiated W, Ta, Mo, and Cb refractory metal alloys have been tested in tension at a temperature of 2000°F (1095°C). Analysis of the data indicates that most of the damage incurred by the irradiation is recovered by testing at 2000°F (1095°C) for the Cb and Mo alloys. The Ta-10 W alloy shows a marked increase in elevated temperature yield strength and decrease in elongation. More important is the difference in fracture mode of the irradiated specimens. Immediately upon yielding, plastic instability occurs and the specimen fractures after experiencing a large reduction in area. Unirradiated specimens, while reaching an early ultimate load, experience a lower rate of instability and a larger total elongation prior to fracture.

Irradiation Effects in Structural Materials

Preliminary investigations have been carried out to determine the adaptability of the DCB specimen for fracture testing of graphite. Excellent behavior was seen for specimens oriented along the extrusion direction. Values of K_{Ic} were obtained for crack initiation and arrest at room temperature. Specimens having a beam depth of 0.25 inch and beam width of 0.25 inch with a 0.070 inch groove displayed a marked variation of K_{Ic} with crack length, the values of K_{Ic} varying from 1.10×10^3 psi $\sqrt{\text{in.}}$ to 1.86×10^3 psi $\sqrt{\text{in.}}$ for crack lengths of 2.7 and 4.3 inches, respectively. Larger specimens of beam depth 0.500 inch, beam width 0.500 inch with a 0.150 inch side groove showed very little dependence of K_{Ic} on crack length. The average values of K_{Ic} obtained for crack initiation and arrest, at room temperature, were 2.13×10^3 psi $\sqrt{\text{in.}}$ and 2.10×10^3 psi $\sqrt{\text{in.}}$, respectively. These tests demonstrated that the DCB specimen is ideally suited for fracture mechanics studies of graphite.

Irradiation Effects on Nickel-Base Alloys

The purpose of this program is to determine the effects of modified microstructures on the irradiation stability of nickel-base alloys. Structural modifications are made by pre-irradiation thermal and/or mechanical treatments and are screened by tensile tests, stress-rupture tests, and metallographic examination.

Stress-rupture testing at 1350°F (730°C) in air has just commenced at BNL-Columbus on 10 specimens of Hastelloy X-280 which were given pre-irradiation

thermal/mechanical treatments and irradiated at 540°F (280°C) to approximately 1×10^{20} nvt. Correction of furnace malfunctions, grip misalignment, and equipment calibration has consumed the last two months.

The design of a shielded facility for creep-rupture testing of irradiated materials is nearing completion. Components for a prototypical, remotized creep-rupture unit will be assembled shortly for a system checkout.

Damage Mechanisms

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

The results of strain rate sensitivity tests at room temperature on Ferrovac E (vacuum-melted) and Battelle (zone-refined) iron samples which had been purified for 4 hours and 24 hours, respectively, in the ZrH_2 system were compared with previous results obtained on "unpurified" specimens. The Ferrovac E sample had a $\Delta\tau$ of 0.42 Kg/mm² compared to 0.34 Kg/mm² for an unpurified sample tested previously under the same conditions. The $\Delta\tau$ showed little dependence on strain. The Battelle sample had a $\Delta\tau$ value which was strongly dependent on strain and increased from 0.42 at 0.125 true strain to 0.77 Kg/mm² at 0.9 true strain. These values and trends are in contrast to previous tests where $\Delta\tau$ decreased with strain from 0.66 to about 0.4 Kg/mm². Further tests are necessary to substantiate these observations.

A furnace for growing iron single crystals has been assembled and is in operation. No single crystals have been grown as yet, but one sample containing grains up to 1-1/2 inches long has been produced. It is thought that the conditions necessary for growth of large grains have been defined. Growth of iron single crystals by zone melting has also been investigated. Some large-grained samples have been successfully produced by this technique.

Examination of Out-of-Flux Zircaloy Specimens from ETR G-7 Loop, Cycles 73 and 74

During ETR cycles 73 and 74 (two quadrants, 12 coupons each) Zircaloy corrosion specimens were exposed in-flux at estimated flux levels of 2×10^{13} and 1×10^{14} nv. The exposure time was 48.2 full power days with 36 days at 280°C nominal. The oxygen level was approximately 1 ppm with 4×10^5 dpm/cc water of Cs-138 (U-235 fission product). In the Bettis ETR loops, Cs-138 concentrations occur in the range 10^4 to 10^7 dpm/cc water. All position B coupons were exposed in the as-etched condition. In position A (1 sec out-of-flux) one-half of the coupons were autoclaved (7 to 9 mg/dm²) and one-half were etched. Out-of-flux coupons were examined for surface activity (β/γ and α).

After three decontamination cycles, alkaline permanganate followed by ammonium citrate, the activity was reduced from 11.5 to 3.0 millirads (zero mr) on a coupon from position A and from 3.0 to 1.5 millirads (zero mr) on a B coupon. Beta/gamma activity on a stainless steel spacer was a factor of 10 higher than the activity on Zircaloy specimens.

Alpha activity was approximately equal, in the range 25 to 240 dpm/cm², for out-of-flux coupons. The alpha activity on the stainless steel spacer was 300 dpm/cm². Decontamination (3 cycles) of the Zircaloy coupons reduced the alpha activity less than a factor of 2. Prefilming did not appear to affect beta/gamma or alpha contamination in the downstream positions.

Weight changes on three of the six B coupons were invalidated by obvious fretting corrosion. Comparison of the remaining upstream coupons with corresponding coupons in A positions showed no significant differences in weight gain for Zircaloy-2 and Zircaloy-4 coupons. Comparison of Zr-3Nb-1Sn coupons shows a higher (~15%) weight gain for the coupon in the A position.

WAPD out-of-reactor data are comparable to Zircaloy-2 weight gains in the out-of-flux positions; however, weight gains for Zircaloy-4 in both positions are ~25% higher than the out-of-reactor data would predict. Weight gains on prefilmed coupons are consistent with the out-of-reactor data. Data now available show no consistent acceleration of corrosion in out-of-flux positions.

ATR Gas Loop Studies

Columbium Sleeve Leak Test

The specimen train and the 2000°F (1095°C) helium in the test section of the ATR Gas Loop will be separated from the in-reactor pressure tube of the loop by a replaceable columbium sleeve heat shield. A labyrinth seal at the top of the sleeve must reduce helium flow in the annulus between the liner and the pressure tube to 25 lb/hr or less at a sleeve temperature of 2000°F (1095°C) and a pressure drop of 30 psi; otherwise heat loss to the water jacket surrounding the pressure tube would be excessive. The nominal diametral clearance between the sleeve and the seal, tested previously, was 0.010 inch as compared to a design clearance of 0.020 ± 0.010 inch. Tests were run by varying the sleeve temperature to as much as 1000°F (540°C). A pressure drop of 35 psi at 1000°F (540°C) and a flow through the seal of 8.9 lb/hr extrapolates to 30 psi, 2000°F (1095°C), and 5 lb/hr, respectively.

Model Gas Loop Heater Construction and Piping Modifications

The tubular molybdenum heating elements in the model heater form a series electrical circuit. In order to minimize connector heat losses, a thermally stable good fit between tube and connectors is required. Locking taper and press fit samples of tube connector joints were heated for a week at 2000-2100°F (1095-1150°C) in vacuum. No evidence of desired diffusion bonding was observed. Metallography indicated a very large grained structure in the connector and evidence of recrystallization and grain growth in the tubing. Other samples of tubing were induction annealed at 2100°F (1150°C) for 10-15 minutes. Expansion of the tubes into the connectors by use of an expanding mandrel was then attempted. Only partial success was achieved.

A sample of Nichrome tubing, such as will be used in the first stage of the heater, was rolled, with an expanding mandrel, into a nickel connector. This joint was heated for a week at 2100°F (1150°C) in vacuum. Subsequent metallography indicated good diffusion bonding.

Corrosion Support

Nichrome samples typical of ATR Gas Loop heater elements have been oxidized in 140 μ , 25 torr and 85 torr oxygen at 1020, 1120, and 1220°C for times to 7000 minutes. Oxidation rates were found to be slower than those previously reported for Hastelloy X-280 and Haynes Alloy 25. The Hastelloy and Haynes alloys were reported to oxidize according to a parabolic rate. Average rate constants for the Nichrome oxidation are:

1020°C in 140 μ replenished O ₂	2.8×10^{-5} (mg/cm ²) ³ /min.
1020°C in 25 torr static O ₂	3.8×10^{-5}
1020°C in 85 torr static O ₂	4.6×10^{-5}
1120°C in 140 μ replenished O ₂	1.7×10^{-4}

At 1220°C in 140 μ replenished O₂, the Nichrome sample oxidized to the extent of 1 mg/cm² in 2500 min. The sample then started losing weight. The weight loss is probably due to the evaporation of the chromium oxide. X-ray analysis indicates the oxide retained on cooling to consist of both chromium oxide and nickel oxide.

Metallographic examination reveals that void formation, similar to that seen in the Hastelloy and Haynes alloys, occurred in all samples tested. No evidence of subsurface oxidation was found.

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NUCLEAR GRAPHITE STUDIESRaw-Coke Graphites

From a single lot of high-volatile Texas Lockport raw petroleum coke a series of five different experimental grades of graphite was produced. One grade was processed directly from the raw coke itself using no additional material as binder. A second was processed in the conventional manner, i.e., from the fully calcined coke and a standard coal-tar-pitch binder. The remaining three grades were produced from cokes of intermediate volatile content obtained by heat treating the high-volatile raw petroleum coke. Appropriate amount of coal-tar derivatives were employed as binders in these three grades. No impregnations were included in the processing of any of the experimental graphites. All samples were graphitized at 2500°C.

The samples have been discharged after an initial exposure of 2760 Wd/At at 550 - 750°C. Length changes were in the range +0.004 to +0.008 for both orientations in all five sets; weight changes ranged from -0.025 to -0.059%; changes in coefficient of thermal expansion varied from -1.4 to +6.7%. Crystallite changes were: c , +10 to +34%; L_c , -35% to -47%; a , +12 to +20%; L_a , -73 to -88%. At this low exposure, no trends are apparent in the data for the different materials with volatile-matter contents of 0.2, 6.3, 9.9, 12.0, and 15.6%.

Cracks and Pores in Graphite

Porosity measurements with a mechanical pycnometer show that the total pore volume is 23.5 ±5% for CSF and TSX graphites and 30 ±2% for baked needle coke (900°C) and BMI graphitization-series molded graphites heat treated to 1370 and 2700°C. The open-pore void volume is 16 to 20% for CSF and TSX and 17 to 21% for the baked and molded graphites.

These results show that the higher bulk thermal expansion observed with graphites of low heat-treatment temperature (less than 1800°C) compared to better graphitized material is not due to smaller open-pore void volume. The higher thermal expansion can be caused by smaller Frozowski-type microcracks and by higher elastic constants which are independent of the crack size.

The Effect of Oxidation on the Sonic Modulus of TSX Graphite

As reported previously, the rate at which the graphite is oxidized has little influence on the decrease in sonic modulus with oxidation. A further demonstration of this phenomenon was obtained by alternately oxidizing given TSX graphite samples at two different temperatures. A

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transverse sample was alternately oxidized at 621°C and 730°C; at the higher temperature, the oxidation rate was increased by a factor of about 14. A parallel sample was alternately oxidized at 621°C and 724°C; at the higher temperature, the rate was approximately 12 times that at the lower temperature. In both cases, the modulus decreased in a regular manner with increasing extent of oxidation, with all of the experimental points falling on a smooth curve irrespective of the temperature and hence the rate of oxidation.

Previous measurements on another transverse sample yielded similar results for alternate oxidation temperatures of 577°C and 621°C. At these temperatures the oxidation rates differed by a factor of about 6. Taken together, the results on the two transverse samples imply that the oxidation rate can be varied by a factor of approximately 30 with no apparent effect on the sonic modulus decrease.

The Oxidation of Boronated Graphite

The reaction of boronated graphite with oxygen in the temperature range 580 to 778°C occurs by two competing reactions:

The formation of B_2O_3 not only decreases the O_2 concentration but also serves as a barrier to O_2 diffusion through sample. Although physically small samples are employed in this investigation, rate measurements indicate that diffusion is influencing the results, particularly at the higher temperatures.

To demonstrate that diffusion effects are occurring, three small samples, 6.6 mm wide and 15 mm long but of different thicknesses were prepared from a small piece of boronated graphite containing a nominal 5 wt % boron. These samples were then reacted with helium containing 1 percent O_2 at a temperature of 730°C. A sample 1.2 mm thick displayed a rate of 5.15×10^{-3} /hr. Doubling the thickness decreased the rate by a third for each of two steps. Employing an approximate treatment of diffusion control due to Wheeler [Adv. in Catalysis III, 249 (1951)], these data yielded an approximate depth of oxidation of 0.35 mm., a small fraction of the sample thickness even though initially the concentration of B_2O_3 should be small. While this result is only semiquantitative, it does indicate that under the experimental conditions employed, diffusion of the reactant into the sample is strongly influencing its oxidation rate.

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Neon-Sensitized Radiolysis of a Mixture of CO₂ and H₂

As part of the program to study the inert-gas sensitization of radiolysis reactions, the reaction of CO₂ with H₂ has been studied in the presence of up to 96% neon. The gas mixtures were irradiated in a gamma flux of 1.3×10^7 R/hr at 350°C.

The CO yield ranged from 3.6 to 3.2 molecules/100 ev as the neon concentration was changed from zero to 96%. The CH₄ yield, though small, is increased as the proportion of inert gas increases.

METALLIC FUEL DEVELOPMENTIrradiation of Metallic Uranium with Submicron Uranium Carbide Dispersion

The irradiation performance of uranium containing a submicron dispersion of uranium carbide is being investigated. Postirradiation annealing of irradiated samples from a uranium rod containing a submicron dispersion of UC, and from a uranium rod containing normal sized UC particles has been completed for temperatures up to 800°C. The observed density changes after irradiation and after postirradiation annealing are summarized as follows:

<u>Element Number*</u>	<u>Annealing Temp. °C</u>	<u>Percent Total Density Change from Preirradiation</u>
32	As-irradiated	-13.4
2	" "	- 3.4
32	625	-14.7
2	625	- 5.7
32	710	-16.2
2	710	- 9.6
32	800	-16.7
2	800	-28.7

*Element No. 32 contains submicron UC, Element No. 2 contains normal UC.

As reported previously the submicron containing rod sustained a brittle failure of the 0.025 in. Zr-2 cladding during irradiation and the loss of restraint resulted in the large as-irradiated density decrease. However, the density decrease of the uranium containing normal UC particles is greater at each annealing temperature than for the uranium with the submicron UC dispersion and at 800°C the observed decrease is almost twice that observed with the submicron dispersion. The density results from these postirradiation annealing tests support the conclusion that the submicron carbide particles inhibit fission gas swelling. Metallography of the samples annealed at each temperature is in progress.

Irradiation of Thorium-Uranium Fuel Elements

The irradiation of six tubular Zircaloy-2 clad thorium - 2.5 wt.% uranium 1.0 wt.% zirconium fuel elements continues successfully in the ETR P-7 and M-3 loops. The two highest exposure fuel elements (currently 12,500 Mwd/T and 11,500 Mwd/T) which were subjected to a loop-discharge incident at the end of cycle 74 were recharged after an examination revealed no damage.

CONCEPTUAL REACTOR DESIGN STUDIESNitride Fuel Cycle Studies

Further work on nitride fuel cycle costs is being initiated with analysis of the reactor burnup calculations which provide a fissile materials balance for the reactor. Calculations are expected to be complete by the end of the month.

During the reporting period, burnup studies were completed for the nitride reactor and the carbide reference core (the C.E. carbide reactor normalized to our calculated critical mass requirement). The DDK temperature coefficient study is also nearing completion.

Four nitride reactor cases have been completed with the two dimensional code DDK. For the reference nitride core a k of 0.9795 was obtained with DDK, for a case where a k of 1.0 was obtained with DTK. Other cases gave sodium void coefficients indicated in the following table.

Sodium Void Coefficients in the Nitride Fuel
Reactor Calculated with DDK

<u>Portion of Core Voided</u>	<u>$\frac{\Delta k}{k}$</u>
Inner core only	+ 1.1%
Total core and axial blanket	- 1.9%
Total reactor	- 2.4%

The current effort with DDK is a study of the sodium void nucleation effect, i.e., to examine the reactivity effect of voiding a small section at the center of the reactor. To study this it was necessary to tighten the convergence criteria of DDK.

After completing the reference core calculation with DDK, it was found that the average flux in the radial blanket was lower than that calculated by DTK. Thus, radial blanket conversion was recalculated with PYRE, using the new value of the average flux. Final region burnup results are given in the following table for the nitride and our reference carbide cores.

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Equilibrium Breedings Ratios (instantaneous B.R. at 50,000 Mwd/ton)
For the Nitride and Carbide Reactors

<u>Region No.</u>	<u>Breeding Ratios</u>	
	<u>Nitride Reactor</u>	<u>Carbide Reactor</u>
I - Inner Core	0.861	0.926
II - Outer Core	0.650	0.695
III - Radial Blanket	3.32	3.25
IV - Axial Blanket	5.06	4.94

A description of the carbothermic-reduction method of producing uranium and plutonium nitrides from oxides, and a report of current status of development, was completed by Ceramics Research and Development Section. This will be used as source material in preparing the final report on the nitride fuel study.

Site Study for Large Organic Component Test Facility

At the request of the AEC-RLOO, a preliminary study was made of Hanford as a site for the Large Organic Component Test Facility (LOCTF). This facility is to be constructed to permit testing of major HWOCR components. In general, the test loop in the facility is designed to accommodate a 90,000 gal/min pump producing a 330 psi head. It will consist of 42-inch OD piping designed for 750°F (400°C) and 400 psig operation, and will have a calibrated flow measuring device, an air-cooled heat exchanger, and a flow throttling valve. Connections are provided for installing large test valves, steam generating components, and flow measuring devices for performance and prooftesting. Thermal shocking capability is also a part of the system design. A conceptual design for the LOCTF is given in the Atomics International - Combustion Engineering document AI-CE-MEMO-1, "Large Organic Component Test Facility (LOCTF) - Conceptual Design and Cost Analysis," October 22, 1965.

There are several locations at the Hanford site which are attractive for the LOCTF. The 189-F Building in the 100-F Area appears to offer the most favorable siting for the minimum over-all cost to the Atomic Energy Commission. The 186-D Building in the 100-D Area also provides substantial capital cost savings and has the required power and supporting services. Another site would be in the 300 Area, where the LOCTF could be constructed without savings in initial capital costs.

An important factor common to all Hanford plant locations is the availability of inexpensive and reliable electrical power at or near the building sites. The cost of electrical power for a facility at Hanford is determined by several factors, including whether the power contracted for is firm or interruptible, the Hanford complex peak load, and whether the facility contributes to the peak load. These variables prevent an

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exact determination of LOCTF power costs. The estimated annual power cost for the LOCTF for each of the first two years of its proposed operating mode (20% load factor for the first two years, 10% thereafter) would probably be between \$88,000 to \$176,000. The latter figure assumes that LOCTF would be part of the peak Hanford load for (an unlikely) six months of each year. Optimum programming of the LOCTF operation would probably reduce the annual power cost to less than \$50,000 for each of the first two years, and \$25,000 for each subsequent year. These costs compare to an annual cost of \$315,000 for the first two years at a site having 10 mill power.

Thermionic Conversion

A review was made of the status of major nuclear reactor-thermionic converter development programs being carried on at other sites, to determine the suitability or incentives for plutonium or U-233 fueling of this type of reactor. Since these programs are presently concerned with in-core thermionic converters combined with the fuel element, the reactor size and design is influenced by converter design parameters, and the potential advantages of plutonium or U-233 fuel would also be affected. A more detailed study of several power ratings would be required to support conclusions regarding the merits of alternate fuels.

UPSTREAM BOILING BURNOUT

Analysis of the data has been completed on the two recent series of upstream burnout tests with water in the high pressure apparatus. Conditions in these tests were 6 ft and 12 ft heated length, 0.435 in. ID test section, mass velocities of 3, 5, and 7 x 10⁶ Btu/hr-sq ft, 1000 and 1500 psig pressure, 400 to 900 Btu/lb inlet enthalpy and 0.22 to 1.47 x 10⁶ Btu/hr-sq ft burnout heat flux. From these data, the following tentative hypothesis has been developed to explain the upstream burnout and related behavior. Further work will be required to confirm this hypothesis.

The most significant observation which must be explained was a marked transition in behavior which took place when the quality at the point of burnout was about 10%. At the lower enthalpies, the burnout heat flux fell sharply with increasing enthalpy and burnout always occurred at the downstream end. At higher enthalpies, the burnout heat flux either fell only slightly with increasing enthalpy (at $G = 3 \times 10^6$ lb/hr-sq ft) or actually rose (at $G = 5$ and 7×10^6 lb/hr-sq ft). Where the burnout heat flux rose with increasing enthalpy, the initial burnout occurred near, but not at, the upstream end of the heated section. In the intermediate transition region, the burnout heat flux and burnout enthalpy stayed nearly constant as the inlet enthalpy was increased, while the location of burnout moved upstream. When the burnout heat flux continued to fall with increasing enthalpy (i.e., at $G = 3 \times 10^6$ lb/hr-sq ft),

the burnout location remained at the downstream end. (Upstream burnout was, however, obtained at much higher enthalpies, i.e., over 40% quality at $G = 3 \times 10^6$ lb/hr-sq ft.) Before the transition and again well after it, the pressure drop across the heated section increased in a regular manner as the heat flux was increased at constant inlet enthalpy. However, in the transition region at all mass velocities studied, the pressure drop decreased over a portion of the range of increasing heat flux.

The above behavior may be explained by postulating that at sufficiently high values of enthalpy and heat flux, a transition in flow pattern occurs to a type of flow for which the pressure drop is lower and for which (at the higher mass velocities) the burnout heat flux increases with enthalpy. Thus, the burnout heat flux passes through a sharp minimum at the point of flow transition, and the burnout location is where the enthalpy corresponds to this minimum. However, the burnout location does not move all the way to the start of the heated length because, it is postulated, transient flow behavior inhibits burnout over the initial foot or two of the heated section. The flow transition occurs first at the downstream end of the test section and then moves progressively upstream as the heat flux and enthalpy are increased. As this happens, the normal tendency of the pressure drop to rise as the result of increased vapor formation is counteracted by the fact that more and more of the tube is in a flow region with a significantly lower pressure drop. Thus, the pressure drop increases less rapidly and in many cases decreases on increasing heat flux.

Although the above can explain the upstream burnout encountered at the minimum in the graph of burnout heat flux vs. enthalpy, further explanation is required to account for the upstream burnout obtained at 40-60% quality at $G = 3 \times 10^6$ lb/hr-sq ft where no such minimum existed. There may be an additional length effect operating here, such that the susceptibility to burnout is greatest at the end of the entrance region and/or where the transition occurs from one type of flow to the other. More study is required before an adequate explanation of this behavior can be offered.

Further hypotheses can be offered as to the nature of the flow transition. It is suggested that at low enthalpy and low heat flux, the flow is characterized by liquid next to the wall in which vapor bubbles form. As the heat flux is raised, the bubbles coalesce to form a continuous vapor blanket. This transition need not result in burnout. With sufficiently high turbulence (linear velocities were on the order of 100 ft/sec at the transition in these tests), an adequate supply of small liquid drops moves through the vapor layer to maintain a wet surface. In this flow region, then, the wall is covered by a very thin liquid film with evaporation taking place from its outer surface. This evaporation creates a constantly replenished vapor layer between the liquid film and the liquid-vapor mixture in the bulk fluid. Slip in this vapor layer results in a lower pressure drop than when liquid is near the wall. Burnout then occurs when the transport of liquid drops across the vapor layer is no longer sufficient to carry off all the heat from the wall. The supply of liquid drops to the wall is favored by a high linear velocity, and hence the rate of supply and the burnout heat flux can be greater at higher qualities. In the unheated entrance region, there is no vapor formation and hence a liquid rather than a vapor layer is present near the wall. A certain time and distance are required for this liquid layer to be broken up by vaporization, and thus burnout never occurs in the initial portion of the heated length.

A new test section was designed for further tests of upstream burnout in the high pressure apparatus. It will have a 0.065 in. nickel wall which will have a much lower temperature drop than the present 0.190 in. Inconel wall and will thus permit better measurement of changes in the heat transfer coefficient to boiling water. This test section will be used for further study of variation of the burnout heat flux along the length and around the circumference of the tube. It will also include pressure taps at 1 ft intervals and thus permit study of the pressure drop as a function of position as well as enthalpy and heat flux and enable one to follow flow transitions along the length of the test section.

A nonuniformly heated test section having alternate bands of high and low heat flux was installed in the low pressure apparatus. A short test series will be run, when a break in the laboratory schedule permits, to determine whether upstream burnout can be produced in this system.

FAST FLUX TEST FACILITY

Spectral Analyses of ATR Loop

Neutron spectral analyses of an ATR loop both with and without sodium-cooled fast-reactor fuel-pin experiments have been completed as reported last month.

The three cases analyzed were: 1) an unperturbed loop; 2) a sodium loop with a 7-pin fast-reactor fuel experiment (the fuel pins consisted of a mixture of 80% natural uranium oxide and 20% plutonium oxide); and 3) a sodium loop containing a single fuel pin but surrounded by a boron stainless-steel filter and an enriched U^{235} converter.

Large differences in the neutron spectra were found for the three cases. The thermal-to-fast ($E > 0.18$ MeV) flux ratio in the center of the loop for the unperturbed case was 3.33. The thermal-to-fast flux ratio for the center of the center pin of the 7-pin cluster was reduced to 0.003. Similarly, in the center of the single-pin experiment the ratio was 0.003. However, the thermal-to-fast flux ratio near the fuel-pin outer diameter was considerably higher, 0.014.

As reported previously the ratio of the thermal fission rate to the total fission rate from the outer ring of the 7-pin cluster to the center of the central pin varies from 0.743 to 0.211. The value of the same ratio across the single pin (utilizing the thermal-neutron filter and converter) varies from 0.481 near the outer edge of the fuel pin to 0.277 in the central zone of the fuel pin.

These results give further evidence that fast-reactor fuel-pin irradiations in a thermal-reactor loop such as the ATR would not be typical of a fast-reactor irradiation environment.

FFTF Core and Fuel Design

During the current report period an FFTF oxide core design and economic analysis was programmed and checked out on the analog computer. Current physics, hydraulic and economic information was also put into the cermet design and economic analog.

Preliminary parameter sensitivity studies were run on both types of cores. Since core operating parameters such as neutron flux, transient overpower, fuel temperature at maximum overpower and pressure drop through the core are set for the oxide design, it appears from early results that the oxide core will have to be much larger than the reference cermet design. The oxide design analog must be rescaled to cover larger core volumes before a realistic oxide core design that meets FFTF operating criteria can be established.

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The preliminary cermet design results showed little change in the fuel cycle cost but a smaller pin size and higher throughput of fissile material than previous results.

A survey of fuel cycle costs for cermet fueled reactors, ranging in power from 100 to 600 MWt, was made using the MIDAS version of the design analog. The results are now being analyzed.

Engineering Development and Test Facility

Study effort to further define facility scope and criteria were continued. Preliminary PERT charts were prepared. Studies were initiated to define the scope and scheduling of test programs required in support of FFTF R&D and design.

Coining Process for Extrusion Billets

Preliminary experiments were performed to evaluate coining as an economical and technically feasible method for powder consolidation prior to extrusion. Initial cold-pressed densities of 85% theoretical density have been achieved for 30 vol% UO_2 -stainless steel (304-L - 325 mesh) powders. A subsequent coining (or repressing) process might be capable of producing densities for powder compacts in the range of 90 to 95% of theoretical density. These densities would be suitable for extrusion.

Powder compacts of 30 vol% UO_2 -stainless steel (304-L - 325 mesh) were produced by cold pressing at pressures between 35,000 psi and 70,000 psi. Some compacts were sintered in hydrogen and the remainder were sintered in helium at 1300°C for two hours. The sintered pellets were then coined (or repressed) in the original die at a pressure of 125,000 psi. During each step the densities were measured by the geometric method. These densities varied from 66 to 74% as a function of the cold pressing pressure and varied from 85 to 90% after coining. Metallographic examination shows voids in the stainless steel matrix to be partially filled with a third phase. This seems to be an oxide phase and also occurs in the unfueled stainless steel powder compacts which were processed as control samples with the fueled pellets.

Observations made on extruded fuel pins whose extrusion billets were prepared by other means (i.e., Dynapak, cold pressing) show fine particle alignment in a string-like configuration. Some of these particles are UO_2 fines; however, some appear to have a tendency to deform or break up quite easily and flow with the metal matrix during extrusion in a manner uncommon to UO_2 . The stringered particles may or may not be identical to the soft phase which is present in the sintered pellets.

Additional pellets were pressed at 100,000 psi to 140,000 psi to observe UO_2 particle behavior and cold-pressed densities as a function of pressure. The densities varied from 79 to 84% as a function of pressure. The UO_2 particles appear intact with very little fracturing as a result of cold pressing. The complete data for these pellets have not been obtained.

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Preparation of PuO₂ Spheroids

Preliminary work using ceramic grade UO₂ (bonded with PVA, "dry" pressed, then granulated) produced particles that were dense and tough enough to be rounded into spheres by jet milling. An initial preparation of PVA bonded PuO₂ (calcined plutonium oxalate) indicates that granules suitable for jet milling can be obtained with this type of oxide. Sintering characteristics of the "dry" pressed PuO₂ remain to be determined.

A 230-gram batch of PuO₂ was spheroidized by adding carbowax binder to the powder while mixing in a Hobart mixer. Forty-five grams of -35 +100 mesh (as green particles) were obtained. Approximately 50% of the particles were -35 +60 mesh, and the other one-half were -60 +100 mesh. These will be sintered in hydrogen at 1650°C.

Particle Tester

The particle crush strength apparatus was delivered and calibrated. A batch of UO₂ particles (nominally 100 micron diameter) was found to have a mean crush strength of 200 grams. In order to avoid contaminating the hood atmosphere, the tester is now driven by nitrogen pressure instead of air.

Cermet Irradiations

Irradiation tests were performed on (1) cold pressed, low density cermets, (2) pneumatically impacted, high density cermets, and (3) extruded, high density cermets. The cermet compositions tested included the following:

- stainless steel - 20 vol% PuO₂
- stainless steel - 30 vol% PuO₂
- stainless steel - 20 vol% β -Pu₂O₃
- stainless steel - 30 vol% UO₂.

The high density cermets were irradiated at maximum temperatures up to ~1200°C. The low density cermets were irradiated at maximum temperatures up to and beyond the melting point of the stainless steel matrix (>1400°C). Both high and low density cermets were irradiated to burnups of 2.3%, 5%, and 7.5% burnup of the heavy atoms.

In all cases the high density cermets appeared to be in excellent condition after irradiation. No dimensional changes occurred in extruded cermet pins (0.25" dia. x 12" long) irradiated to 5% BU. The core-clad metallurgical bond formed during extrusion was retained throughout the irradiation. A reaction between PuO₂ and stainless steel was noted; however, the reaction was confined to the immediate vicinity of the individual PuO₂ particles and did not appear to degrade the matrix. No reaction was evident between UO₂ and stainless steel.

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Some of the low density cermet operated at temperatures sufficiently high to melt the stainless steel matrix. In the region surrounding the molten zone extensive grain growth and densification occurred in the matrix. At cermet temperatures estimated to be between 1300 and 1400°C, suboxides were formed in the PuO_2 . As with the high density cermet, the PuO_2 reacted with the stainless steel but did not appear to impair cermet integrity. No reaction between UO_2 and stainless steel was detected in the low density cermet. At cermet temperatures estimated to be 1000°C and higher, sintering of the fuel particles and the stainless steel increased the cermet density.

The low density stainless steel 20 vol% $\beta\text{-Pu}_2\text{O}_3$ cermet was irradiated at an estimated maximum temperature of 900°C. The $\beta\text{-Pu}_2\text{O}_3$ appeared to have undergone more of a reaction with stainless steel than did the PuO_2 . The $\beta\text{-Pu}_2\text{O}_3$ appears to have diffused into the matrix--"fingers" of plutonium oxide extending from the fuel particles into interfaces between the surrounding stainless steel particles. In spite of this reaction, the cermet appears to be in good condition.

Sodium Bonding Facility

Instrument racks have been installed beside the bonding facility and instrument wiring has been completed. All electrical work is now completed except the installation of heating cord around sodium lines.

Sodium lines have been put in place and all joints are being welded. Lines will now be leak tested and welded joints will be radiographed. All tanks and traps have been installed. Work is now approximately 85% complete.

PHOENIX FUEL

Initial planning of the thermal hydraulics analysis for the MTR core is underway. Four cores are being considered: a 5x5, 3x9 symmetrical, 3x9 asymmetrical and a 3x3 partial core with 3x3 uranium loadings at each end. Thermal and hydraulic limits will be evaluated using power distributions calculated by Engineering Physics Unit. A thermal hydraulics computer code, written especially for the general MTR core configuration, is being planned to expedite this work.

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HIGH TEMPERATURE LATTICE TEST REACTORHTLTR Large Mockup

Run Number 4 is in progress. The graphite core temperature of 1000°C has been maintained for 30 consecutive days. Rate of decrease of gaseous contaminants in the core atmosphere has slowed. Present concentrations of H₂O, CO₂, and CO in the blower discharge gas (mockup inlet) are 80 vppm, 106 vppm, and 850 vppm, respectively. Mockup outlet gas concentrations of H₂O, CO₂ and CO and H₂ are presently 200 vppm, 98 vppm, 870 vppm and 800 vppm, respectively. The cleanup system exit gas stream water concentration has been less than 20 vppm for the past month.

IRRADIATION TESTING OF EBWR PROTOTYPE FUEL RODS

Three capsules containing short lengths of vibrationally-compacted EBWR fuel rods attained a burnup of $\sim 5.6 \times 10^{20}$ fissions/cm³ (22,500 MWD/MT of fuel) and are being irradiated to higher burnups. The fuel rod sections contain UO₂-1.5 wt% PuO₂. A maximum burnup of 4.7×10^{20} fissions/cm³ was obtained on the 29 capsules already discharged.

Excellent irradiation performance has been obtained in the PRTR from the 21-rod bundle (FE-6501) containing replaceable production-run EBWR rods. Maximum burnup as of September 29 is 0.7×10^{20} fissions/cm³. Irradiation of the fuel bundle will continue. Goal exposure is 2.7×10^{20} fissions/cm³. One rod of this bundle was removed and replaced. Irradiation data on this rod are included in the table.

Continued satisfactory performance is being observed with the three-rod cluster irradiation tests. Goal burnups for the clusters (36 rods total) are 0.4-, 1.4-, and 6.8×10^{20} fissions/cm³. The production-run rods are 1.07 cm OD by 148 cm long.

PHYSICS AND INSTRUMENTS DEPARTMENTPLUTONIUM RECYCLE PROGRAMExperiments with D₂O ModeratorCritical Experiments with 1.8 w/o Pu-Al Fuel

Calculations of results to compare with experimental results obtained for Pu-Al-D₂O loading in the PRCF have continued with Program HFN. A value for the thermal absorption cross section of a control rod region was chosen such that the thermal flux shape matched the experimental BF₃ traverse and the worth of moving the control rods radially matched the experimental value.

The calculated k_{eff} for a loading of seven clusters and the moderator level at 103.25" is 4% above the value determined experimentally. Calculations of fluxes were made both radially and axially in order to compare calculated and experimental BF₃ and Au traverses. Both of the calculated traverses agree with the experimental data to better than 3%.

HPD Core Experiment in the PRTR

A series of measurements are being conducted in the PRCF to verify calculational methods used in the analysis of the HPD core for the PRTR. The fuel elements used in the PRCF experiments will be used later in the HPD core of the PRTR. Thirteen clustered fuel elements have been loaded in the PRCF, and 4.8 ppm B¹⁰ boron has been added to the moderator to reduce the effective neutron multiplication to unity.

The temperature coefficient of reactivity as a function of temperature is being measured over a range of temperatures between 20°C and 70°C. Near 20° the coefficient is slightly positive. It becomes negative near 30°C and is nearly constant with increasing temperature. The temperature coefficient measurements are about 50% completed.

The worth of a one-inch annular void around a central fuel thimble relative to 4.8 weight ppm B^{10} in D_2O was obtained from subcritical multiplication measurements. The worth of the void plus a 6" O.D. aluminum thimble was -13.5 mk, and the thimble worth in borated D_2O was -2.56 mk. The reactivity difference of the one-inch annulus when filled with borated D_2O and pure D_2O was 2.05 mk as measured by positive periods. Removing the pure D_2O from the 3.75" fuel thimble reduced the excess reactivity by 0.563 mk.

Analysis of data from reactor noise recordings on magnetic tape indicated the desirability of obtaining results for lower frequencies than was possible with the frequency analyzer. By copying the magnetic tapes at a speed to give an increase of a factor of eight in the recorded frequencies, the lower frequencies were brought into the range of the analyzer.

Frequency analyses of the power fluctuations of a critical core with seven fuel elements and 0.75 ppm B^{10} in the moderator and of a core with 13 elements and 4.80 ppm B^{10} has revealed values of β/λ of $\sim 5.0 \text{ sec}^{-1}$ for the former and 9.40 sec^{-1} for the latter concentration. Reduction of the data obtained with a ten-channel frequency analyzer is accomplished using Program NOISE. A multi-group noise subroutine for Program FLEX will be used to obtain a least squares fit of the analytical power spectrum to the data. This will provide a better estimate of β/λ and a measure of the error associated with it.

A brief summary has been written of the results of the PRCF experiments which have been conducted to date. Wherever possible, comparison is made to calculated parameters in order to assess the adequacy of calculational models used in the analysis of the HPD core in the PRTR.

PuO₂-UO₂ Burnup Experiments

Analysis of burnup data from the irradiation of UO₂ or PuO₂-UO₂ fuels requires a knowledge of the fraction of all fissions resulting from the fission of U^{238} by fast neutrons. In general, this fraction is obtained from Δ = ratio of U^{238} fissions to fissions caused by slow neutrons and the latter ratio depends upon the geometry and fuel composition of the cell under consideration. Lack of experimental data of the quantity Δ for the PRTR 19-rod element requires that it be estimated by some computational scheme. The computer code GAMTEC II is formulated in such a manner that Δ is calculated directly. Program GAMTEC II has been used to calculate values of Δ for a cylindricized model of a PRTR 19-rod cell. The cell was cylindricized into 14 regions (3 of which are fuel regions) by conserving nuclear densities. Values of Δ which resulted from the

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calculations for various fissile materials in a given U^{238} density are given in Table I.

TABLE I

<u>Fissile Material</u>	<u>Density of U^{238} (atoms/barn-cm)</u>	<u>Δ Values</u>		
		<u>Center</u>	<u>Middle</u>	<u>Outer</u>
U^{235}	0.020415	0.064	0.061	0.049
Pu^{239}	0.020415	0.083	0.077	0.056
Pu^{241}	0.020415	0.085	0.080	0.060

Center, middle, and outer refer to the three fuel regions (i.e., center rod, 6-rod ring, and 12-rod ring, respectively. The value of Δ center is larger than that for Δ outer because the center rod is partially shielded from slow neutrons but not from fast neutrons.

In addition, the dependence of Δ upon the fissile material depends primarily upon the average number of neutrons per fission and to a lesser amount upon the fission neutron spectrum. As is expected, for this cell, Δ is directly proportional to the nuclear density of U^{238} and is independent of the nuclear density of the fissile material and Pu^{240} concentration.

Average Values from Plutonium Burnup Data

Data from the destructive sampling of aluminum-1.8 w/o plutonium fuel rods have been obtained as a function of position in a given rod. The a/o Pu^{239} , a/o Pu^{240} , a/o Pu^{241} , a/o Pu^{242} and the percent depletion, β , of the plutonium, of each sample have been plotted versus sample position. In the past, smooth curves have been drawn in by inspection to determine average values. Analytic functions have since been used to describe these data in order to deduce the average values of the above data per rod and per cluster and to aid in the process of assigning uncertainties. The average values from the fits are summarized in Table II.

TABLE II

AVERAGE VALUES FROM PLUTONIUM BURNUP DATA

Element and Rod Identification		β	a/o Pu ²³⁹	a/o Pu ²⁴⁰	a/o Pu ²⁴¹	a/o Pu ²⁴²
Element 5051 (13.8 MWd)	Outer Ring Rod	5.71	90.39	8.773	0.8114	0.0353
	Middle Ring Rod	4.88	90.84	8.389	0.7396	0.0310
	Center Rod	4.34	90.97	8.273	0.6889	0.0307
	Avg. for Cluster	5.37	90.56	8.625	0.7823	0.0337
	Exposure (MWd)	14.1				
Element 5103 109.5 MWd	Outer Ring Rod	40.03	62.40	30.06	6.348	1.177
	Middle Ring Rod	37.49	65.06	28.85	5.199	0.880
	Center Rod	36.35	65.92	28.34	4.933	0.819
	Avg. for Cluster	39.03	63.42	29.59	5.911	1.064
	Exposure (MWd)	104.2				

An equation in the form

$$y = A_1 + A_2 \exp(-|A_3x - A_4|^n) + A_5x$$

has been used where six or more samples are available from a rod. Six fuel rods (three from element 5051 and three from element 5103) with sufficient data are available. The function which is the normal or Gaussian distribution, when $n = 2$, was fit to the data weighted inversely as their uncertainties by the method of least square using the code MISFIT III. This n was adjusted higher to flatten the curve and lower to increase the peak. The "goodness of fit" of the function as n was varied was based on the chi-square test⁽¹⁾ applied to the minimized sum of the square of the residuals, Q , and the T test⁽²⁾ applied to the parameters, A_1 , of the function. Each fitted function was then integrated over the length of a fuel rod by Simpson's rule to determine the average values. Average values for a cluster were found by assuming the twelve outer ring rods in the cluster are identical to the outer ring

(1) Cziffra, Peter and Michael J. Moravcsik. A Practical Guide to the Method of Least Squares, UCRL-8523 Rev. June 5, 1959.

(2) Volk, William. Applied Statistics for Engineers, McGraw-Hill Book Co., New York, 1960.

rod analyzed and the six inner or middle ring rods are identical to the middle ring rod analyzed. The heat output or exposure of the cluster was then determined using the average depletion β . This value can be compared with the coolant flow-temperature rise determination of exposure during reactor operation. Both values are shown in Table II and are seen to be in good agreement.

In addition to the values determined above, the fitting procedure is a basis of comparing uncertainties⁽³⁾ assigned experimentally to those assigned by the least squares analysis. In particular, it can be concluded that the uncertainty in β (4.1% at 1% depletion which decreases to 2.1% at 50% depletion) is realistic and is conservative. The uncertainty in the atom percents is, generally, realistic and conservative, the only exception being Pu^{242} atom percent. However, in this case the T test indicates one or more parameters of the function are not significant and another function should be tried.

Pu^{240} Resonance Studies Using Iridium Foils

Studies have begun to determine the feasibility of using iridium foils to measure the departure of the epithermal neutron spectrum from the 1/E form in the region of the 1.054 eV Pu^{240} resonance. Iridium appears to be ideally suited for this measurement because it consists of two isotopes: Ir^{191} , which has a large resonance at 0.4654 eV producing a radioisotope with a 74-day half-life; and Ir^{193} , which has a large resonance at 1.303 eV producing a radioisotope with a 19-hour half-life. By having both activities in the same foil problems of source geometry, absolute flux, irradiation time, etc. are eliminated.

Preliminary calculations have indicated that useful data can be obtained by this method. Iridium foils covered with 20-mil cadmium have been irradiated in the Thermal Test Reactor (TTR) to measure their behavior in a 1/E neutron spectrum. These foils are currently being counted and the data analyzed. It is expected that additional irradiations will be made in the Plutonium Recycle Critical Facility (PRCF).

(3) Cziffra, Peter and Michael J. Moravcsik. A Practical Guide to the Method of Least Squares, UCRL-8523 Rev. June 5, 1959.

H₂O Moderated SystemsMixed Oxide Reactor Survey Calculations

A series of calculations which were made earlier have been extended in order to gain a better understanding of plutonium-fueled light water reactor systems. The improved methods include cross section updating and resonance self-shielding for 18 rods instead of only 6 rods which was used initially. The extensions include recalculations using improved methods and calculations for a higher Pu to H₂O ratio.

The fuel selected for the study is 3/8" diameter rods, 2 w/o Pu₂-UO₂ in natural UO₂, clad with 25 mils of Zircaloy. The plutonium contains 8 w/o Pu-240. The range of conditions bracket the regions of interest for light water pressurized and boiling water power reactors.

The initial study was for lattice spacings of 0.60, 0.70, and 0.80" in a hexagonal array. A 0.55" lattice spacing was added to broaden the undermoderated range of moderator-to-fuel volume ratios. The temperatures selected cover the ranges: Fuel 20°C to 500°C, and moderator 20°C to 260°C. To simulate boiling water systems, void fractions covering the range from 0 to 30% voids have been chosen.

Infinite multiplications for all systems were determined along with parameters which can be used to account for leakage such that the appropriate critical mass for any of the systems can be determined for any specific geometry. Few group parameters were calculated with HRG for the fast group and THERMOS for the thermal group. Temperature, void, and Doppler coefficients were determined along with the effects the plutonium has on each.

Calculations of the burnup of the fuel have been started with ZODIAC in order to determine the physics characteristics as a function of exposure. Thus far the burnup to a 12,000 MWd/t exposure has been calculated for the 0.60, 0.70, and 0.80" lattices at 30% void (boiling water environment), and the 0.70" lattice at 0 voids (pressurized water environment).

The results of the study are being summarized and will be reported in a formal document.

Comparative Studies of Scattering Kernel

The comparative study of scattering models for thermalization calculations, using the THERMOS code as reported last month, has been extended to include

the graphite-moderated H_2O cooled PCTR experiments. Whereas the calculated results of η and f , reported last month, agree very well with experimental values for the air-cooled lattices, the calculations have an $\sim 2.5\%$ discrepancy in f for the H_2O -cooled lattices, independent of whether the gas model or the Nelkin model is used for the H_2O scattering kernel. Thus, the Nelkin kernel is inadequate for the H_2O and D_2O systems studied if the criterion for validity is matching the reactor parameters, η and f .

CAF Experiments with Plutonium Fuel

Experiments to measure critical masses, bucklings, and reflector savings of 2 wt% PuO_2 - UO_2 fuel rods, one-half inch in diameter and 36 inches long, moderated with H_2O are continuing in the Critical Approach Facility (CAF). Experiments using fuel containing 24% Pu^{240} have been completed for a hexagonal lattice spacing of 1.05 in.

In addition, an experiment has been completed using 1.8 wt% Pu -Al fuel rods containing 6% Pu^{240} for a hexagonal lattice spacing of 0.80 inches.

Experiments with Control Rod Materials

Measurements of reactivity worths of neutron absorbing rods in the Critical Approach Facility were made in two loadings. Both loadings contained 2 wt% PuO_2 - UO_2 fuel rods in a hexagonal lattice with a spacing of 1.05 in. The two loadings differed only in the amount of Pu^{240} in the fuel, one containing 16% Pu^{240} and the other 24% Pu^{240} .

Physics Code Development

1. Computer Code LLESQS

The code LLESQS, a linear least squares fitting program which was written for the IBM-7090, has been converted for use with the UNIVAC-1107. In addition, the program has been revised to calculate and print out the intercept on the x-axis for the best fitting curve. The error for this intercept for a given confidence interval is also calculated and printed.

2. Program HXXS

The two main programs for cross section calculations have for some time been HRG and THERMOS, for epithermal and thermal cross sections, respectively. In order to speed up the cross section calculation process, HRG and THERMOS have been combined with a diffusion theory code HFN into

Program HXXS. Program HXXS is written for use with the UNIVAC-1107.

Program HXXS can be used to compute cross sections using HRG and THERMOS, to store the cross sections on drum and execute HFN in one pass. More than one cycle may be made through Program HRG and THERMOS before the HFN calculations are made. In such a case cross sections from each pass through HRG-THERMOS will be available to HFN. All cross sections may be stored on a tape for later use at the option of the user.

Another major option of Program HXXS is the ability to have input cards punched for Program S. Microscopic cross sections are computed and punched as well as a transmutation matrix.

3. ZODIAC3

The feasibility study for ZODIAC3, a revised version of the one-dimensional burnup code, ZODIAC2, is complete. Plans for ZODIAC3 call for greater flexibility of operation and decreased volume of input. The primary target, however, is a setup which will be able to meet the anticipated demand for additional capabilities.

It has been determined, by means of a dummy code simulating the size and features of ZODIAC3, that the EXECII monitor system of the UNIVAC-1107 has adequate facility to accommodate the code as planned. With the chain of dummy codes in operation, their real counterparts will be substituted one by one. The HFN portion is now being checked out. A realistic target date for completion of the entire chain is not yet available.

4. PLOTX

The IBM-7090 computer program PLOTX has been converted for use in the UNIVAC-1107 computer. PLOTX is used to plot the neutron cross section data calculated by the computer program BARNS II from data contained in the BNW Master Library (BNWL-CC-325). The output from PLOTX is plotted on the Benson-Lehner plotter. The normal output is log sigma vs. log energy and is plotted on 3 x 5 cycle log paper; however, an option has been added that will produce a linear plot of sigma vs. energy.

This linear plotting option will allow neutron cross section data to be presented graphically in much finer detail than is possible using a logarithmic plot.

5. BNW Master Cross Section Library

The current version of the BNW Master Cross Section Library has been printed on duplimat paper for publication as report BNWL-CC-325. This report will be available for distribution in approximately one month. The current report is about 1000 pages long and has a format that will allow substitution of pages of updated isotope records for old pages. Revised cross section data will be collected and distributed periodically or upon special request.

Theory DevelopmentCalculation of the Thermal Neutron Scattering Law for Glycerine

In a recent paper (BNWL-149), D. R. Skeen has compared, measured, and predicted foil activation in a critical system with a special detector whose contents include glycerine ($C_2H_5(OH)_3$). His reported calculations assumed a hydrogen gas scattering kernel for glycerine, but a more precise analysis will require a better kernel for glycerine.

The scattering law for glycerine has been calculated using the technique of Egelstaff and Schofield, who assume that the intermediate scattering function $\chi(\underline{k}, T)$ can be represented as a Gaussian distribution in $\underline{k}$:

$$\chi(\underline{k}, t) = e^{-\underline{k}^2 \gamma(t)}$$

where $\gamma(t)$ is related to the mean square displacement of an atom of the scatterer away from its initial position. The Fourier transform of the second derivative of $\gamma(t)$,

$$\rho(\omega) = \int_{-\infty}^{\infty} e^{-i\omega t} \gamma(t) dt,$$

has the character of a density of states for the scatterer, and a knowledge of $\rho(\omega)$ is sufficient to determine the scattering law for all energy transfers ($\hbar\omega$) and momentum transfers ($\hbar\underline{k}$).

Larsson and Dahlborg have measured $\rho(\omega)$ by slow neutron scattering experiments for $0 < \omega < 0.1$ eV, an energy range corresponding primarily to translational and rotational states of the molecule. The values of $\rho(\omega)$ for $\hbar\omega > 0.1$ eV, which correspond to internal vibrations of the glycerine molecule, are at present inaccessible to neutron scattering experiments but can be inferred from infrared absorption spectra and a knowledge of typical interatomic bond frequencies of carbohydrate

molecules. Using bond frequencies given by Herzberg, 26 of the normal mode frequencies of glycerine have been estimated and are found to agree closely with observed peaks of the infrared absorption spectrum. The remaining ten modes are estimated to have frequencies with $\hbar\omega < 0.1$ eV, and thus to contribute to $\rho(\omega)$ only in the range measured by Larsson and Dahlborg.

In determining contributions to $\rho(\omega)$, each normal mode frequency has been weighted according to its degree of degeneracy. While not exact, this procedure should give adequate results for the applications presently envisioned. The LEAP II and ADDELT codes are being used to calculate the scattering law from $\rho(\omega)$.

Isotopic Analysis of PRTR Samples

Isotopic analyses were performed on 18 burnup samples of PRTR-irradiated fuel elements. These included 5 uranium and 8 plutonium samples from UO₂ fuel element No. 1097, 1 uranium sample from PuO₂-UO₂ fuel element No. 5185, and 4 plutonium samples from PuO₂-UO₂ fuel element No. 5217.

EBWR PROGRAM

The startup tests for the plutonium experiment in the EBWR have begun. Preliminary values of unrodded, cold clean critical mass of plutonium and boric acid requirements for the plutonium zone have been received from Argonne personnel. The number of elements for critical is ~ 22 (or 792 rods), and the amount of boric acid required for a 36 element Pu loading is ~ 5 gms of H₃BO₃/gal H₂O. The temperature coefficient experiment for the plutonium zone is presently under way.

PLUTONIUM RECYCLE FUEL CYCLE ANALYSIS

American Nuclear Society Papers

During this report period, essentially all efforts were expended on preparing papers, listed below, for the ANS Meeting in Washington, D. C. A formal document containing these papers presented by members of the Fuel Cycle Analysis Section will be issued forthwith.

1. E. T. Merrill, "Monte Carlo Calculation of Optimum Loading of Plutonium in Fuels," BNWL-SA-278A.
2. D. E. Deonigi and J. H. Nail, "A New Fuel Configuration to Increase Plutonium Value," BNWL-SA-277A.

3. D. E. Deonigi, "A Comparison of Fuel Cycles for Crossed-Progeny Fuels in Conventional and Novel Configurations," BNWL-SA-275A.
4. W. W. Porath, D. E. Deonigi, and E. T. Merrill, "The Calibration of a High Speed Burnup Code for Nuclear Fuel Surveys," BNWL-SA-274A.

Fuel Cycle Code Development

An initial control configuration based on the epithermal flux is now operational on an 1107 version of ALTHAEA. The initial control approximation is adjusted in an iterative loop to approach a flat power situation prior to calculating a burnup. The method has been successful for UO_2 and $\text{UO}_2\text{-PuO}_2$ fuels tested to date in water reactors.

HWO CR Reactivity Coefficient Comparison

A brief comparison of the reactivity coefficients for the heavy water organic cooled reactor as calculated via the ALTHAEA chain was made with similar calculations made by ORNL, AI, and CE. The initial results are shown in Table III.

TABLE III

REACTIVITY COEFFICIENT

<u>Reactivity Coefficient</u>	<u>ALTHAEA</u>	<u>ORNL</u>	<u>AI</u>	<u>CE</u>
Fuel Temp., $\Delta K/K^\circ F(10^{-6})$	-4.44	-5.96	-6.85	-8.95
Coolant Temp., $\Delta K/K^\circ F(10^{-6})$	+19.9	+6.14	+8.13	+9.90
Coolant Void, $\Delta K/K$	+0.0213	+0.0542	+0.0471	+0.0421
Moderator Temp., $\Delta K/K^\circ F(10^{-6})$	-2.85	-3.71	-6.44	-2.48

One notes that in all cases, the spread is quite large for the ORNL, AI, and CE results, ranging from 25% to 90%. This trend is further accentuated when the ALTHAEA results are included. It should be noted that all of the ALTHAEA results are conservative from a safety point of view except the void coefficient. It is not clear at this time why the coolant temperature coefficient is the largest and the void coefficient the smallest. This seems somewhat inconsistent and may indicate the treatment of voids is inadequate. This will be checked in the future.

SPECIFIC FUEL CYCLE ANALYSIS PROGRAMFuel Utilization Studies

The effect of plutonium holdup, or "immobilization," in the nuclear fuel cycle can affect the amount of uranium required in an expanding economy by more than a factor of 2. The holdup period is defined as the time which lapses between the formation of plutonium in the reactor and its subsequent recharge after irradiation, discharge, separation, chemical conversion, refabrication and shipping. A current mean value for this holdup is about three years. If a 3-year holdup is assumed, a heavy water reactor-fast breeder system is calculated to require about 720,000 tons of natural uranium between now and the year 2020. If a 5-year holdup is assumed, 1,100,000 tons will be required, and if a one-year holdup is assumed, only 280,000 tons will be required. These are shown in more detail in Figure 1. The difference is a direct consequence of the ability to build plutonium fueled breeders sooner when the holdup is shorter. This means that fewer uranium using thermal reactors will be required to provide our energy demands. For example, as shown in Figure 2, in 1990, a total of 200,000 breeder megawatts are installed if a one-year holdup period is assumed, but only 30,000 megawatts are installed if a 5-year holdup is assumed. However, after the year 2020, the total breeder megawatts are about the same, no matter what holdup is assumed. Along with these differences in uranium requirements we have differences in separative duty requirements, shown in Figure 3. The longer holdup periods result in larger separative duty requirements. The maximum separative duty requirements for holdups of one, three, and five years are about 2.2, 5.5, and 10.0 million kg/yr, respectively. The cusp in the curve around 1980 represents the switch of thermal reactors from plutonium recycle to uranium-only operation as breeders are introduced. The differences in these separative duty requirements curves result from differences in thermal reactor capacity. The curves flatten out when breeder reactors are supplying all the plutonium needed for the increasing breeder inventory requirements, and plutonium becomes an excess commodity. Breeders will also be exporting plutonium to thermal reactors, which will then be using as much of the excess plutonium as they can use economically.

A draft of the final report on the VESTA code, a report which includes the result of several parametric studies of reactor system in an expanding economy, is nearing completion.

Application of Linear Programming

A procedure for determining the minima of nonlinear curves, to any desired degree of accuracy, by means of linear programming techniques has been devised. However, the method requires acceptance by the LP program of a

EFFECT OF TIME PERIOD LENGTH ON URANIUM REQUIREMENTSFIGURE 1

INSTALLED BREEDER CAPACITY FOR VARIOUS TIME PERIOD LENGTHS--HWR

FIGURE 2

EFFECT OF PERIOD LENGTH ON SEPARATIVE DUTY REQUIREMENTS- HWRFIGURE 3

segmented curve representing a logarithm table, and the application of LP to segmented curves (which was reported in October) is currently not satisfactory for the general case.

Earlier attempts to locate the minima of segmented curves with LP were highly successful in a series of test cases and were so reported. Subsequent extension to more difficult curves showed that the point identified by the LP program as a minimum was actually determined by other factors, and that the point so located fortuitously coincided with the minimum in all of the earlier cases. Two new techniques for presenting a segmented curve in a form which is compatible with the LP programs have been tried, in addition to the three which were initially tried, without success.

To facilitate trial runs, the IBM-7090 program which produces LP-1107 input was converted to a UNIVAC-1107 program. The program permits the generation of input card images, processing of multiple cases by LP, and manipulation of the LP output, in one computer run.

Code Development

The JASON code has been revised so that a more accurate calculation of effective heat transfer surface can be made when the clustered fuel option is used which ultimately considers the burnup as occurring at a point. A "heat transfer utilization" for each fuel region is now calculated as follows:

$$\text{Heat Transfer Utilization} = \frac{\text{HTS} \times F(K)}{V(K)^2}$$

where

HTS = heat transfer surface of fuel region

F(K) = thermal utilization of fuel region (that is,
the relative fission rate)

V(K) = volume of fuel region

American Nuclear Society Paper

The following paper was presented at the Washington ANS meeting during November:

W. I. Neef and E. D. Jones, Jr., "Conservation Economics and Reactor Technology," BNWL-SA-276A.

NEUTRON CROSS SECTION PROGRAM

(Partially funded by Division of Production - Reactor Programs)

Neutron Scattering from H₂O

The data obtained during October on the scattering of 0.3 eV neutrons from water using the Rotating-Crystal Spectrometer were reduced to partial cross sections and scattering law formulations. As expected the results were of poor quality due to high and variable backgrounds of reactor neutrons from an improperly shielded test facility. A further measurement was performed after the backgrounds were reduced but the data have not yet been analyzed.

Rotating-Crystal Spectrometer

A single crystal of copper was cut into five thin slabs which will be used to fabricate a composite crystal for the RXS monochromator. The new dual channel delay has been put into operation and performs satisfactorily.

Neutron Scattering from D₂O

A thin-walled sample holder has been constructed for use with D₂O samples. Sample transmissions are designed to be 90% or greater.

Triple-Axis Spectrometer

Additional measurements were made on the elastic scattering from vanadium in the study of systematic errors in the 3-axis system.

Multiphonon Calculations

Work has continued on the theoretical calculations of representative models to demonstrate the nature of multiphonon processes in the interpretation of partial and total neutron cross section measurements. Extensive use of the LEAP/ADDELT computer programs in these calculations has revealed further program deficiencies which are now being investigated. Modifications have been made so far to the restart feature of LEAP and the card punching feature of ADDELT in the 1107 versions.

Fast-Neutron Cross Sections

Preparations are under way for a measurements run early in December. Samples of Ca⁴², Ca⁴⁴, W¹⁸², and W¹⁸⁶ have been removed from their cans in preparation for return to ORNL. Improvements have been made to the 1107 computer program TAPEWORM/RAW to give a drastic improvement in running time. Tests

have been made on the Least Squares Fit of total cross section data which have determined that the use of a second order polynomial is sufficient.

PHOENIX FUEL PROGRAM

Objectives and Limitations of the MTR-Phoenix Experiment

At the request of the Idaho Operations Office, AEC, a trip was undertaken recently to acquaint NRTS personnel with the scope of the proposed MTR-Phoenix experiment, to define problem areas that might exist, and to explore possible solutions.

The nature of the proposed MTR-Phoenix experiment is affected by two different programs. The experiment should advance the technology of the "Phoenix Concept." At the same time, however, it must also not interfere with the present physics research programs carried out at the MTR, and with the proposed conversion of this reactor. Often, the requirements of these two programs tend to conflict, and satisfactory compromises must be sought.

The main objectives of the MTR experiment, as set by the Phoenix Program, are:

1. The MTR test core should approximate, as closely as possible, a typical Phoenix core configuration.

This requirement implies the use of high Pu^{240} content fuel ($\sim 20\%$), high fuel loadings (.1 to .2 kg/lit), high metal content ($M/W \sim 1$).

2. High fractional fuel burnups should be achieved.

Assuming no restrictions were placed on the length of the experimental run, high fractional burnups require highly loaded, high endurance cores.

3. The test results should be amenable to analysis.

Analysis is always simplified if core configurations and mode of operation (i.e., rod motions) have a high degree of symmetry. For instance, a four row core using the present MTR control rod configuration is less attractive than more symmetrical configurations (i.e., fuel box arrangements of 5×5 , 3×9 , 5×9 , etc.).

4. The MTR-Phoenix test should be inexpensive.

MTR modifications should be kept to a minimum; testing and other R/D preceding the experiment should be minimized, and the core size should be kept small to reduce fabrication charges. Partial Pu loadings may be of interest.

The continued operation of the MTR as a physics research tool and the scheduled conversion to a research reactor dictate additional requirements:

1. The MTR-Phoenix experiment must be run between September 1967-July 1968, and the actual experimental run should last about 4 months.

To meet this rather tight schedule places emphasis on a rather conventional design with as few innovations as possible. This approach will also simplify the hazards analysis. In fact, it may be desirable to keep close to cycle 108 configurations and reduce Pu inventories by employing a partial Pu loading.

If only four months is allocated for the test run, the use of large, highly loaded cores becomes less attractive if high fractional burnups are to be achieved.

2. The thermal flux levels in the side beam tubes and in the transverse HT-1 tube should not be affected adversely.

High reflector flux levels require high volumetric power densities in the core, i.e., 40 MW operation in small cores. Furthermore, to assure flux patterns similar to those presently available, a core 9' units long appears to be desirable

On the basis of these requirements, it would appear that a 3 x 9 core, 23.5" long with .117" water gaps would be potentially acceptable. Previous calculations* have shown that a core loading of ~ 1 kg/lit with a M/W ~ 1 would have an endurance of about 3-4 months and the fuel inventory could be accommodated using 20 w/o Pu in Al plates. Achievement of 40 MW in a core of this type is, however, quite problematical. Furthermore, the shutdown capability of this core with conventional MTR control rods has not yet been ascertained. Using 8 rods, symmetrically placed, will not only increase the shutdown margin but might also improve the power distribution.

* Holeman, R. H. and P. L. Hofmann, "Nuclear Design Calculations for a High Exposure Plutonium-Fueled MTR," 12/15/64, HW-84575, Unclassified.

Other alternatives are to fuel-zone the core or to simply operate at reduced power (lower flux) until the power peaks have burned down.

The primary consequence of the visit to NRTS is the decision to abandon the previously selected 5 x 5 core. It would adversely affect the experiments in the HT-1 tube for the duration of the Phoenix experiment. Four alternate cores, shown in Table IV, are now under investigation.

TABLE IV
CORE CONFIGURATIONS

<u>No.</u>	<u>Configuration</u>	<u>Fuel Type</u>	<u>Fuel Ldg.</u>	<u>M/W</u>	<u>No. of Rods</u>
1	3 x 9	H x Pu	~.1 kg/lit	~1.0	4
2	3 x 9	H x Pu	~.1 kg/lit	~1.0	8
3	3 x 9	U + HxPu	~.1 kg/lit	~1.0	4
4	3 x 9	U + HxPu	~.1 kg/lit	~1.0	8

The Pu zones would have 0.080" fuel plates consisting of 0.040" Pu-Al with a 0.020" Al clad on each side. The 16 fuel plates would be contained in a standard MTR fuel box and a near standard water channel would result.

The U²³⁵ fueled zones characteristics for the partial Pu cores have yet to be determined. Calculations are under way to determine the power profiles for all cores. The resultant peak to average power ratios will to a large extent define the potential power level of the core. As noted above, a high power level is necessary to assure continuity to the existing MTR physics research programs. The U²³⁵ content of the partially Pu loaded core will be selected to minimize power peaks. The partial Pu cores have the advantage of lower cost, larger neutron delay fraction, and possibly a more easily attainable 40 MW core. The disadvantages include a more complex experiment, hence a more difficult analysis, and perhaps a non-typical flux spectrum in the Pu zone.

Current calculations are aimed at selection of a core concept by the end of the calendar year.

MTR Phoenix Critical Approach Facility Experiments

In connection with the current MTR-Phoenix core design effort an early set of approach-to-critical experiments utilizing the present PuAl disc inventory is under consideration. Calculations are under way on the proposed

cylindrical experimental geometry. The objectives of these calculations are: 1) To define critical cylinder dimensions, and 2) to determine the change in reactivity upon removing the water top reflector, the core water and finally the bottom water reflector.

This information is needed to determine the material requirements for the proposed experiments and to start preliminary hazards analyses. The calculations are approximately 50 percent complete at this point.

Cycle-108 Analysis

Analysis of the Cycle-108 MTR configuration is being continued. Most of these calculations are being carried out by means of the 3-D WHIRLAWAY code. A control rod model has been established, and some comparisons of calculated and measured flux distributions have been made. The initial results are quite encouraging.

MTR-Phoenix Critical Experiments

Experiments are being planned for the Critical Approach Facility and the Plutonium Recycle Critical Facility which will use unclad discs of Pu-Al moderated by H₂O. Arrangements have been made for Coolant Systems Development Unit to perform corrosion tests and gas evolution measurements of prototype fuel assemblies. These assemblies will consist of a stack of Pu-Al discs (20 w/o Pu containing 8 w/o Pu-240), aluminum discs, and aluminum spacers in aluminum cans filled with water. Enough test data should be available by the end of December to permit final design of critical experiment hardware.

A few preliminary calculations have been made using the MIDAS (Modified Integral Digital Analog Simulators) program in preparation of its use for nuclear safety estimates.

Pu-Al Polyethylene Experiments in the PCTR

The paper, "Analysis and Measurement of Pu-Al-Polyethylene Systems in the PCTR," by D. G. Albertson and G. J. Busselman, was delivered at the 1965 Winter Meeting of the American Nuclear Society in Washington, D. C., November 17, 1965. It is anticipated that the material contained in that paper, along with additional material derived from another experiment and further calculations, will appear in a formal report early next year.

~~CONFIDENTIAL~~

Fast Reactor Cross Section

Three isotopes in the 26-group Russian cross section set have been updated using recent nuclear data. The fission cross section of Pu^{239} and U^{235} was lowered and the capture cross section of Fe increased in the fast neutron region. The original Russian set gave a lower critical mass for the FTR than other principal cross section compilations, but the new set (RUSS II) tends to agree with other sets, especially the latest Argonne set (ANL-224).

To check some of these modifications, the two-dimensional ODK transport theory code was used to calculate the reactivity for the Fermi Core B critical assembly using the RUSS II cross section set. Results of similar calculations using the ANL-224, Hansen-Roach, and the original Russian set were reported last month, but are included for comparison. All computations were performed with 4 energy groups derived from the above sets using the CRUNCH group collapsing code.

CALCULATED k_{eff} - FERMI CORE B

<u>Cross Section Set</u>	<u>k_{eff}</u>
ANL-244 Set	0.985
Hansen-Roach Set	1.018
Russian Set	1.032
RUSS II Set	0.988

Sodium and Iron Cross Sections

Knowledge of precise values for the high energy (~ 1 MeV) cross sections of sodium and iron is quite important for fast reactor calculations. Consequently, elastic and capture cross sections for these elements have been reinvestigated for the energy range of .01 to 10 MeV in order to update the BNW Master Neutron Cross Section Library.

Computer Program BWN has been developed and used to obtain the resonance parameters including potential scattering which characterize the energy dependence of these cross sections as given from the latest experimental and evaluated data. This program which uses BNW Program MISFIT III is designed to fit the data with Breit-Wigner single level expressions and has the options of including interference between potential and resonance scattering and contributions from neighboring resonances and sums of several resonances.

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Group average calculations using BNW Program BARNs have been made with the original and updated cross sections with the result that the updated ones give, in general, closer agreement with those sets given by other laboratories (Bondarenko, Hansen-Roach, ANL-224).

FTR-MCA Analysis

The methods and models to be employed in calculating the energy release during a maximum credible accident (MCA) in the FTR were discussed with Nuclear Technology Corporation personnel. NTC has been engaged as a consulting firm for the FFTF project. A program to evaluate the presently available computer codes and analytical models was outlined. The following topics will receive major attention:

- a) The mechanism which initiates the accident and the means by which the core becomes supercritical,
- b) The effect of distribution of fuel in the core,
- c) The geometry of the supercritical core,
- d) The Doppler temperature coefficient,
- e) The type of fuel in the core, and
- f) The material parameters (i.e., equation of state) of the reactor constituents.

The results of this investigation should indicate the sensitivity of the calculations to various assumptions.

Pu-238 Buildup in the FTR Driver Fuel

Processing of FTR driver fuel could be affected by the presence of Pu^{238} . To assess the severity of this problem, some very preliminary estimates of the buildup of Pu^{238} in cermet driver fuel ($\text{PuO}_2\text{-SS}$) and in ceramic driver fuel ($\text{PuO}_2\text{-U}^{238}\text{O}_2$) in a 400 MW, 600 liter FTR reference core have been made. Since the available cross section data are quite uncertain, some rather crude assumptions had to be made.

The results of the calculations are tabulated as follows for several irradiation intervals.

Pu²³⁸ BUILDUP IN FTR DRIVER FUEL

	<u>PuO₂- SS Driver</u>	<u>PuO₂-U²³⁸O₂ Driver</u>
<u>Time (days)</u>	<u>ppm Pu²³⁸/Pu</u>	<u>ppm Pu²³⁸/Pu</u>
0	0	0
40	18	20
80	35	41
120	53	64

Fast Reactor Fuel Cycle Calculations

Fuel cycle analysis of fast spectrum, oxide and carbide fueled breeder reactors is continuing. Of particular interest in this work is a comparison of systems in which the fuel is enriched in one case with plutonium, and in the other case with U²³⁵. Power cost components covering only charges for inventory (at 10% use charge) and burnup of fissile fuel (with credit for bred fuel) have been computed for two such oxide systems. Averaged over burnup to 120,000 Mwd/MTM, this cost fraction was found to be 0.39 and 1.39 mills/kWh for the plutonium and U²³⁵ enriched systems, respectively.

The breeding ratios were computed from one-dimensional calculations for the oxide systems (patterned after the Allis Chalmers 1000 MWe Reference Design). Initial breeding ratios are 1.037 and 0.604 for the plutonium and U²³⁵ enriched reactors, respectively. These values must be augmented by 0.06 due to breeding accomplished in blanket positions not included in the one-dimensional model. Information in this regard was obtained from two-dimensional (R-Z), four energy group, diffusion theory calculations using the 4-K ANGIE program. The relatively lower value of the breeding ratio (plutonium-enriched system) compared to the breeding ratio reported for the original Allis Chalmers design is a direct result of the 14% higher initial fissile load assumed for the present study.

Analysis of the carbide system Reference Design⁽⁴⁾ proposed by Combustion Engineering is in progress.

(4) Liquid Metal Fast Breeder Reactor Design Study, CEND-200, Volumes I and II, January 1964.

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Instrumentation, System Studies and Digital Systems

The dynamic responses of a continuous multidecade analog simulation of the FFTF reactor kinetics equations were determined for comparison with similar runs made on a digital computer code. For rapid reactivity excursions, the short neutron lifetime (5×10^{-7} seconds) is a significant factor in the power level response. The need to simulate both this extremely short neutron lifetime and the relatively long delayed neutron decay constants results in large dynamic range requirements.

To date, attempts to use lifetimes of less than 10^{-5} seconds on analog computer simulations have not been successful; consequently, neutron lifetimes of 10^{-5} and 10^{-4} seconds were used for the analog model.

Good agreement was obtained between the analog and digital computer results for reactivity ramps up to about 50 cents per second, using $\beta^* = 10^{-5}$, and up to 10 cents per second for $\beta^* = 10^{-4}$. These tests assumed zero temperature coefficient feedback and low initial power levels (minimum critical power level).

Good agreement was obtained for a reactivity ramp of one dollar per second using temperature feedback, with the initial power level in the power range (2000 watts/c.c.). Cases using faster ramps have not been obtained from the digital computer code to date.

A simplified model of an electromagnetic brake was developed and added to the FFTF dynamics analog simulation model. It will provide means for studying the feasibility of using EM brakes to reduce primary loop post-scam thermal stresses. The effects of primary coolant inertia were also added to the model.

A report describing the FFTF dynamics analog simulation model and its use to date has been drafted. Derivations of the model equations, calculations of process constants, and typical results are included.

A linear process model was simulated on the Donner 3500 analog computer for use in checking the programming and operation of the G. E. 412 digital computer as a direct digital controller. The model consists of a pure deadtime cascaded with one to three series-connected first order lags. Consultation and operational assistance was given to Digital Systems Development Unit in preliminary testing of the direct-digital-controlled (DDC) process.

Oxide fuel and cermet fuel FFTF design optimization models were demonstrated to FFTF personnel and to AEC visitors. A meeting was held with interested plant and FFTF personnel to establish needs for the study and to solicit information concerning experimental value for an improved performance measure. A MIDAS program for the oxide fuel is being written and the oxide analog simulation is being reviewed for possible scaling changes.

Following conceptual revisions to the basic reactor design, the problems involving fuel element rupture monitoring and neutron flux monitoring were carefully reviewed and new design concepts were prepared. Major effort has to date involved designing and calculating, with extensive experimental work to follow as the schedule requires. Shielding calculations were also performed for the reactor top shield.

The four segments of the analog controller simulation program were debugged and combined to form the complete program. All the necessary cabling and computer connections for controlling a process and recording the input to and output from the G. E. 412 were made. A portable analog computer was connected to simulate a second order process with a variable transport lag. This simulated process, with lag set to zero, was controlled automatically using the simulation program. Preliminary data show that good control is possible. Some bugs in the program at the time of the test made it impossible to use all the features of the program.

The regular and the special timing chains in the Process Control prototype were set up during the month. The necessary cables and parts to operate the rest of the system have now arrived and are being installed.

HIGH TEMPERATURE REACTOR PHYSICS PROGRAM

The High Temperature Lattice Test Reactor (HTLTR), now under construction, is to be used in the subject program to measure the variations in reactor physics parameters of a variety of reactor lattices with temperature. The reactor will provide a facility for measuring various nuclear cross sections as a function of neutron temperature as well as allowing reactor design experiments to be performed on mock-up cores of large, high power reactors.

HTLTR Mockup and Materials Studies

The 1000 hour run of the HTLTR mock-up at 1000°C has been in progress for about 600 hours without indication of improper functioning of any of the reactor components under test. These include heater rods, control and safety rods with their actuators, thermocouples, moderator graphite, thermal insulation, and the containment shell. During the run the contaminant gases, primarily H₂O, and CO, have been gradually lessened by purging and by the use

of water removal towers. Their concentrations are now at 200 and 800 vpm, respectively. This is below the 300 vpm water that has been considered a reasonable lower limit to aim at in the reactor. From the data on H₂O concentration in the mock-up and the observed corrosion rates of samples placed in it, a summary has been made of the expected corrosion in the reactor. This is to be used in evaluating the need for including a water removal system in the construction of the HTLTR.

Control Rod and Fuel Element Worth

An experiment was performed to determine the worth of the HTLTR control rods. A 12-inch mock-up of one section of a control rod was used. The section was placed in a flux environment in the PCTR similar to that to be expected for the HTLTR. At this location the change in reactivity from the most reactive to the least reactive configuration was found to be 9.8 cents. This is about 1/6 of what can be expected for one complete, 7½ foot long HTLTR control rod. There will be 8 in the reactor. A subsequent chemical analysis of the mock-up fuel sections indicated that the actual control rod worth will be larger than this by as much as 30%. This is because the uranium density in the element tested is about half of that which the actual control rod will have. Therefore, in the actual control rod, the outer fuel section will be blacker, causing a greater change in reactivity than that measured when the inner fuel cylinder slides under it.

At the same time a worth of 26¢ was measured for a 10¼-inch length of a fuel element having the approximate diametral dimensions and composition of an element for the reactor.

Pu-Be Source Availability Investigation

Contacts have been made with vendors to investigate the feasibility of using a Pu²³⁸-Be startup source in the HTLTR. Although initial contacts indicated that this seemed an attractive possibility, later contacts revealed several major objections to it. These were: (1) There is increasing evidence that recrystallization in Pu-Be sources at temperatures as low as 500°C causes swelling and rupture, (2) Pu²³⁸ - Be sources having yields of about 10⁸ do not hold dimensions due to internal heating even at room temperature, and (3) the amount of Pu²³⁸ required for sources much larger than 10⁸ requires extremely specialized handling due to melting and even spontaneous ignition caused by internal heating.

Safeguards Report

Review of the HTLTR Safeguards report continued. A short addition was made to the analysis of self-limited excursions, in which the effects of such an

excursion on the control rods were studied.

HTLTR Construction and Procurement

Design review was completed on the following specifications as a result of which each specification will be significantly altered. The specifications describe:

1. Neutron generator
2. Panel mounted instruments
3. Radiation monitors
4. Criticality monitor

Information about costs, operating expense, and availability was obtained for AEC-RL00 on neutron sources for HTLTR. Chemical sources and two types of particle accelerators were included.

The J. A. Jones Construction Company has moved into the 318 Building. Installation of the reactor has been started with the welding in place of the base plate and the erection of the prefabricated shell. Finish plaster work on all offices, hallways, control room, and change room was completed. Some excessed capital equipment is being obtained for use in Building 318.

Preliminary drafts of the Acceptance Test Procedures for HTLTR equipment and the specification for assembly of the core were reviewed.

Instrumentation

Assistance was given Vitro Corporation in the formulation of Acceptance Test Procedures for the Programmed Measurement and Control System (PMACS). Preparations were made to teach an introductory course on the PMACS. These courses, to be taught in the near future, will be available to interested personnel involved in HTLTR work.

Performed experimental irradiations determined that saturation properties of a commercial BF_3 -filled ionization chamber were significantly affected by elevated temperatures. During heating to about 200°F in the Hanford Test Reactor at a neutron flux density of about 10^7 n/cm²-sec, the chamber output current increased approximately 2% per 100°F increase in temperature. This is an undesirable coefficient, unacceptable for the application, and further chamber investigations will be made to attempt to achieve a solution. More chambers have been ordered.

PLUTONIUM CRITICALITY STUDIES PROGRAMBasic Criticality Studies with Plutonium Nitrate Solutions - Slab Geometry

A slab tank for criticality experiments incorporating a special variable thickness adjustment feature, together with its control and safety rod system, and a slow fill device for adding Pu solution was received from Mechanical Equipment Development. The installation of these components in Hood 1 of the Critical Assembly Room is now proceeding. This criticality vessel will permit the determination of critical thicknesses of bare and reflected "infinite" slabs of Pu solution throughout the thickness range of 3-9 inches; variations in critical thickness will be measured as a function of nitrate and Pu concentration in aqueous solutions

Correlation Between Computed and Measured Criticality of Pu Solutions in Spherical Geometry

The aptness of multi-group diffusion theory for computing criticality of Pu solutions in simple geometry (spheres) has been further demonstrated by comparisons of measured and computed criticality. The early experimental criticality data obtained by F. E. Kruesi, et al., at Hanford in 1950-51 on water reflected spheres of Pu nitrate solutions was compared with the results obtained from multi-group diffusion theory with 18 energy groups. The multi-group constants were obtained with the GAMTEC-II Code utilizing cross sections normalized to the Sher data for ^{239}Pu at 2200 m/s. The results of these calculations are presented in the following table. The computed values of k_{eff} should be unity to be in agreement with the experiments. The average computed k_{eff} for the 39 critical experiments considered is 0.9998 ± 0.0060 . The uncertainty represents the deviation of the computed k_{eff} values with the exception of one case (bare 16-inch sphere with Pu concentration of 38.3 g Pu/l) where k_{eff} was computed to be 12.7 milli-k above unity.

Similar calculations were previously made for the more recent criticality experiments performed in the Pu Critical Mass Laboratory that also showed good agreement between theory and experiment for clean spherical assemblies of Pu solution.

Application of Code MSIFIT III to Pulsed Neutron Source Experiments

The usual output of pulsed neutron data is a printout from a multichannel time analyzer in terms of counts per channel. Computer reduction of this output data is desired to obtain information on the neutron lifetime and the reactivity of the system being pulsed. Subroutines have been written for the computer code MSIFIT III that employ the Garelis-Russell theory for determin-

COMPARISON BETWEEN COMPUTED AND MEASURED
CRITICALITY OF PU SOLUTIONS IN SPHERICAL GEOMETRY

<u>Pu, g/l</u>	<u>NO₃, g/l</u>	<u>H₂O, g/l</u>	<u>n+</u>	<u>²⁴⁰Pu, %</u>	<u>Computed k_{eff}</u>
<u>12.0-INCH WATER REFLECTED SPHERE</u>					
50.34	133.0	941.8	1.57	3.12	0.9977
51.00	163.0	931.2	2.20	3.12	0.9967
56.20	207.0	902.9	2.49	3.12	0.9969
59.93	237.0	889.7	3.20	3.12	1.0008
63.75	270.0	877.8	3.62	3.12	1.0037
70.22	322.0	844.9	4.07	3.12	0.9960
77.22	399.0	828.6	4.72	3.12	1.0021
<u>13.0-INCH WATER REFLECTED SPHERE</u>					
33.17	86.2	964.4	1.02	1.76	0.9940
34.81	116.0	943.3	1.46	1.76	0.9976
37.99	132.0	939.0	1.64	3.12	0.9990
38.83	156.0	931.1	2.01	3.12	1.0006
41.13	205.0	905.8	2.90	3.12	1.0014
44.64	269.0	873.2	3.71	3.12	1.0020
36.27	93.1	960.5	1.71	3.12	0.9994
<u>14.0-INCH WATER REFLECTED SPHERE</u>					
26.33	77.3	964.7	0.99	0.54	0.9974
26.77	107.0	951.7	1.42	0.54	0.9962
27.91	138.0	935.2	1.85	0.54	1.0000
28.63	188.0	912.8	2.72	0.54	0.9974
27.95	109.0	950.1	1.44	1.76	0.9967
28.78	87.4	959.7	1.02	3.12	0.9968
30.15	109.0	951.3	1.39	3.12	1.0039
30.33	143.0	933.5	1.92	3.12	0.9983
31.79	208.0	903.3	2.90	3.12	0.9968
35.82	308.0	851.1	4.39	3.12	1.0007
39.62	408.0	797.4	5.75	3.12	0.9980
29.94	87.5	959.3	1.01	4.05	0.9964
30.75	119.3	945.2	1.44	4.05	0.9985
34.07	210.7	901.9	2.92	4.05	1.0032
36.38	272.4	870.1	3.69	4.05	1.0031
38.72	335.2	837.8	5.46	4.05	1.0059
41.12	384.9	809.1	6.40	4.05	1.0013
31.04	127.0	942.4	1.71	4.40	0.9970
32.41	158.0	927.9	2.22	4.40	1.0006

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15-INCH WATER REFLECTED SPHERE

<u>Pu, g/l</u>	<u>NO₃, g/l</u>	<u>H₂O, g/l</u>	<u>H+</u>	<u>²⁴⁰Pu, %</u>	<u>Computed k_{eff}</u>
25.02	116.0	945.3	1.60	3.12	0.9980
25.83	147.0	931.9	2.08	3.12	0.9999
27.05	212.0	898.8	3.07	3.12	0.9983

16-INCH BARE SPHERE

34.80	104.4	955.8	1.12	4.17	1.0013
38.31	163.0	933.3	2.79	4.17	1.0127
43.20	281.8	866.6	4.18	4.17	1.0005

ing reactivity; the prompt neutron decay constant is determined from a least squares analysis of a single exponential plus a constant.

The computer code MISFIT III was utilized because of the flexibility available in over-riding the built-in subroutines with subroutines written specifically for the analysis of pulsed neutron source data. The desired results represent a more accurate determination of the prompt neutron decay constant, and, similarly, a more accurate determination of the subcritical reactivity in a given experimental determination. The use of MISFIT III has other advantages over the code GLEX, previously used, in that the data cards need be punched and run only once to obtain both the values for the prompt neutron decay constant, α , and the subcritical reactivity, ρ . Also, a graph plotting subroutine is available in code MISFIT III.

Consulting Services on Nuclear Safety - Criticality Hazards

a. Nuclear Safety in Pacific Northwest Laboratory

The following nuclear safety specifications were reviewed:

- 1) Specification B-19 for Reactor Physics and HTR Physics covering 3.9 wt% and 4.3 wt% U^{235} enriched UO_2 pellets and rods.
- 2) Specifications Q1-Q10 for Ceramics Research and Development covering plutonium handling in 308 Building.
- 3) Temporary Specification 14 for Metallurgy Development covering the special casting of plutonium metal.

A series of DTF calculations were made to confirm the nuclear safety of the plutonium solution concentrator newly installed in the Mixing Room of the Plutonium Critical Mass Laboratory. This concentrator is of a special annular design which may be of interest. The O.D. of the concentrator is 14 in. and its height is 29.5 in.; the vessel is clad on the outside with 0.030 in. Cd sheet. The solution annulus is 1.57 in. thick (13.75 in. O.D., 10.62 in. I.D.); the total capacity is 29 liters. Steam coils of 1/2 in. diameter are located within the inner section for heating the solution. In normal operations plutonium nitrate solutions will be concentrated to 300-400 g Pu/l. The maximum expected concentration is 600 g/l. The nitric acid content will range from 1-10 molar.

In the present calculations, the solution was assumed to consist of a homogeneous PuO_2-H_2O mixture (no nitrate); other conservative assumptions were also made in evaluating the conditions for criticality. The results are as follows:

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<u>Assumptions</u>	<u>Computed k_{eff} of Concentrator</u>
1. No nitrate; 1000 g Pu/l; Cd sheet on outer surface. Outside water reflector of 8 in.; and inside water reflector of 1/4 in.	0.901
2. Same as (1), but inside water reflector of 0.59 in.	0.974
3. Same as (1), but inside water reflector of 0.79 in.	1.002
4. Same as (1), but inside water reflector of 1.0 in.	0.952
5. Same as (1), but Cd removed and outside water reflector of 1.0 in.	0.918

To obtain criticality, it would be necessary to 1) exclude nitrate; 2) obtain a concentration of 1000 g Pu/l (tank full); 3) increase the neutron reflection on the outside from nominal to full, and on the inside from 1/2 in. thick steam to a uniform layer of water 0.8 in. thick. It is concluded that the operating contingencies afford adequate nuclear safety.

b. Nuclear Safety in N-Reactor Department

Proposed revisions in the nuclear safety specifications for processing slightly enriched uranium (up to 1.95 wt% ²³⁵U) in the 306 Building Pilot Plant were reviewed for the N-Reactor Fuels Section.

NEUTRON FLUX MONITOR PROGRAM

Regenerating Detectors

Further operating data, obtained from the two regenerating neutron flux detectors installed in KW Reactor, showed good sensitivity after a year of in-core exposure. The resistance degradation problem appears to be lessening in rate. Liaison is being conducted with manufacturers of ceramic seals and cables to determine if fabrication of two more assemblies might provide improved performance regarding insulation degradation.

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Beta Current Detectors

Operational characteristics of a commercial beta-current neutron flux detector employing rhodium rather than boron as the emitting material are being determined on the basis of data obtained from tests at KW Reactor. The detector has operated correctly for several months and the dynamic range appears to be three decades of neutron flux in the specific environment.

Another detector, identical to the reactor-installed unit, was irradiated in a Co^{60} gamma pit. The observed current was about 10^{-17} A per R/hr with the readings strongly dependent on the material and thickness of several surrounding test foils.

Fabrication was initiated on an improved quartz capsule to contain the B^{11} for the experimental beta-current neutron flux detector. The fabrication process is difficult in that a small aperture is required to permit out-gassing, yet not permit loss of the powdered boron, and also allow a sealed penetration for the signal wire. The advantages of the boron approach versus the rhodium are a considerably longer in-core life and a faster response.

Microwave Detectors

The Invar material, which is being tested for use as a resonant cavity material, was annealed following machining to relieve stresses. Following this, tests established the coefficient of expansion as about 1×10^{-6} inches/inch, which is reasonably compatible with the handbook values. Thus, the material demonstrates promise for a resonant cavity with a minor, if not negligible, temperature effect problem. Reactor tests will be conducted to verify the properties as soon as a suitable gas-sealing procedure is established and an assembly is fabricated.

DETECTION OF COOLANT IMPURITIES AND HIGH TEMPERATURE PROGRAMMicrowave Techniques

Consideration of radiometric methods, both X and K bands, for measuring reactor high temperatures indicates that the optimum operating frequency may well be in the X band due to instrumentation limitations regarding noise figures. Theoretically, the higher band (K) appears better since maximum power is radiated at the higher frequencies; however, typical K band radiometers have a noise figure of about 19dB and the X band units show a 5dB noise figure.

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Infrared Techniques

Assembly of equipment to prepare helium gas with known moisture contents moved a step nearer completion with acquisition of necessary components and partial fabrication. Work on this will be accelerated as much as possible.

CRACK PROPAGATION DETECTION PROGRAM

This effort is to satisfy a need for a system to continuously monitor pressure piping and vessels, particularly in nuclear reactors, for the onset of a stress condition which represents impending failure of the component. At present, a system is being investigated which consists of a group of specialized sensors feeding their signals into conditioning electronics, a pulse counter, and finally, to a magnetic tape in order to preserve significant data for subsequent analysis. The data analysis would be for the purpose of locating the area emitting the signals.

Five notched tensile specimens of type 212-B and 1040 steel have been pulled while monitoring the specimen surface with charge accelerometers. In all cases, sharp pulses of energy were detected slightly before the onset of noticeable metal yielding and continuing through the early stage of gross yielding. The frequency of the pulses decreased significantly as the process was continued on to specimen fracture. Output from some of the accelerometers has been recorded on magnetic tape to facilitate analysis of the characteristics of these pulses. This analysis is in process.

Tensile specimens of aluminum, stainless steel and zirconium are being fabricated for similar tests to investigate pulse characteristics which may be unique to a given material.

The Containment Systems Experiment (CSE) program has been reviewed with the thought that pressure testing of CSE piping and vessels provides an opportunity to test an acoustic emission monitoring system under field conditions.

NONDESTRUCTIVE TESTING RESEARCH PROGRAMElectromagnetic Testing

In the application of the multiparameter eddy current test method to the testing of tubing, the test coil assembly is excited (driven) by a multi-dimensional signal and the test coil output signal is expanded in terms of its Fourier components. In one form of the tester, the test coil multi-dimensional output signal is separated into its component carrier signals

by use of tuned amplifiers. The Fourier components of the signal are then obtained by phase detectors which demodulate the signal. Because the Fourier components of the signal are recombined in the instrument to form the independent parameter readouts, it is important that the transmission times of the modulation components of the different carriers through the amplifiers, detectors, and filters be equal. Some differences in transmission time have been previously observed which interfered with the proper combination of the Fourier components under the dynamic conditions of the test. This difficulty was overcome by changes made on low pass filters in the demodulator output circuits. A study of the factors contributing to these effects shows that there are two places in the tester which are particularly important. These are the bandpass filters in the receiver amplifier and the low pass filters in the output circuits of the demodulators.

The filters should be designed to give equal delays of the signal through the different channels. To obtain equal carrier envelope delay through the receiver amplifiers, the rate of change of phase angle with respect to frequency should be the same for each receiver amplifier. The requirement for the low pass filters is that the ratio of phase angle to frequency for each frequency to be transmitted is the same for each filter. Practical application of these conclusions in the design of multiparameter testers will next be explored. Tests made with the multiparameter tube tester in its present stage of development show that filter performance is not a limiting factor.

Fundamental Ultrasonic Studies

Previous work comparing experimental pulse distortions due to attenuation with those calculated from the wave propagation model revealed unexplainable differences in pulse shapes. Similarity of the difference error revealed by subtracting calculated from measured values lead to the belief that a systematic error was present in the wave model program. The program is currently being checked in detail; no significant corrections have been made.

The computer program which was written to calculate the general attenuating wave reflection and transmission factors, refraction angles, and phase velocities as functions of incident angle has been checked with a water stainless steel interface. The data have not been completely analyzed; however, the calculations appear to be in agreement with results which have been observed experimentally.

Infrared and Thermal Testing Research

Practical application experiments can best be carried out with a thermal transducer system having high sensitivity to changes in thermal properties of the test objects. Although fair sensitivity has been obtained with the original system, modifications to obtain higher sensitivity and accuracy appear both feasible and desirable. A peak temperature readout and phase detector triggering circuit has been developed for the thermal transducer system. Performance of a breadboard model was better than the original goal of ± 0.1 percent accuracy at frequencies between 0.0001 and 10 Hz.

Construction of a rack mountable prototype is nearing completion. When incorporated in the system, this circuit should make it possible to detect smaller differences in samples. In addition, troublesome effects from transients due to small differences in initial sample temperature will be eliminated.

INSTRUMENTATION SUPPORT FOR LABORATORY 04 PROGRAMSPlutonium Properties Measurement

As previously described, a mechanical system was fabricated for use in plutonium properties measurements. This equipment was installed and checked out in the 231-Z Building prior to installation in a hood. All equipment functioned as anticipated. The equipment has now been installed in a plutonium metal working hood and is ready for use in measuring plutonium properties.

A system using 5 Mc eddy currents for measuring plutonium properties has been constructed and debugged. The design and construction of probes have been started in preparation for in-hood installation. This system will be used simultaneously with the ultrasonic equipment.

The possibility of using the sound wave generated from a spark discharge milling machine for measuring specimen thickness and possibly controlling the machining limits is being investigated. It has been established that a spark discharged on a specimen surface generates a sound wave pulse which propagates through the specimen at longitudinal wave velocity. The specimen thickness is then determined by the pulse-echo time interval. Further investigation of this measuring technique awaits the construction of special, optimized receiving transducers and time interval measuring equipment.

The construction of multiunit transducers was continued. These transducers will be used for measuring equipment alignment and simultaneously conducting multiple tests on a given specimen. Work was continued on the development of techniques for depositing multiunit electrodes on ground focused ceramic ferroelectric elements.

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Eddy Current Testing of Crack Propagation Velocity

The development phase of the prototype eddy current instrument for measuring crack propagation velocities has been completed. The next phase will be to test the instrument in cooperation with Reactor Metals Research, Metallurgy Research Section.

A new eddy current test coil probe having a nylon form and a longer connecting cable was built to better satisfy the final test conditions. The instrument bridge drive transformer and bridge circuit components were modified to match the new probe assembly.

New calibration curves were obtained showing instrument output versus probe to crack tip position for aluminum and zircaloy specimens under static conditions. These show a linear relationship between output signal and crack tip position over a 0.25 inch change in position. The bandwidth of the instrument is about 3 MHz and was determined by applying a modulated carrier to the input of the first amplifier and observing the demodulated instrument output.

Thickness Measurement

A new ultrasonic technique for measuring plating thickness has been developed which has potential for solving the problem of measuring the thickness of one material plated on a different metallic substrate. The technique, which is based on the difference in the velocity of sound in the two materials, is useful for making thickness measurements when the plating material has an acoustic impedance similar to the substrate. Tests were performed using a sample with the acoustic impedance of the plating and the substrate almost identical. A good signal indicating plating thickness was received where no thickness indication would be obtained using conventional pulse-echo ultrasonic techniques.

Solidified Waste Storage-Pot Inspection Facility

Attempts to measure storage-pot wall thickness while the pot is rotating have met with only limited success. Difficulty with transducer alignment will probably necessitate redesigning the mechanical pot scanning mechanism. A traveling V-block will be tried but this will be ineffective in some areas because the thermocouples fastened to the outside of the waste pot will interfere with the V-block travel. In addition to this the pot is made up of welded pipe consisting of a flattened area near the weld seam which will be difficult to measure. At present it appears that only about 75% of the vertical wall can be tested. Removal of the thermocouples and the use of seamless pipe would probably solve this problem.

Problems with the ultrasonic crystals may require alteration of the crystal design. Pulse reflections in the present type make it very difficult to measure wall thicknesses below about 3/16 inch. A different type of crystal may be used but will probably require an additional amplifier.

Solidified Waste Storage-Pot Level Detector

The time domain reflectometer was successfully used to measure glass level in the top section of the glass furnace-waste pot combination. A one-inch diameter platinum probe was used in the all-platinum furnace to measure level changes of about six inches.

Sodium Level in EBR-II Fuel Capsules

Development has been completed on a sodium level monitor for use in fabricating experimental fuels. The device operates at 27 kc and uses mismatched coils to register passage of the sodium meniscus as well as the presence or absence of the sodium column. The device was tested successfully at 550°C using a simulated sodium level. The system is now ready for incorporation into the sodium furnace.

Laser Hole Drilling

The giant pulse laser was used on two occasions for customer hole drilling. In the first instance, holes were drilled in a number of aluminum fuel casing end caps for use as controlled defects. In the second instance, a number of holes were drilled in a 0.005 inch thick tantalum sheet in an effort to produce holes of less than 0.001 inch in diameter for use in collimating neutrons. The holes actually produced varied from less than 0.001 inch to approximately 0.007 inch in diameter with the possibility that some (not visible) were much less than 0.001 inch. This interest in laser hole drilling has prompted the fabrication of a jig which will permit a higher degree of control over the beam focal point and therefore over the hole diameter.

Laser Induced Ultrasound

When light from a giant pulse laser is reflected from the surface of a metal block, considerable energy and momentum are transferred to the block. The extent to which this energy is converted to ultrasonic pulses is being investigated. Tests were made with a laser, focused and unfocused, and Q-switched and unQ-switched. These tests indicated that the amplitude of the ultrasonic pulse was most closely correlated with the total energy of the laser beam, independent of whether or not the beam was focused.

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Li-Ion Drifted Germanium Diodes

The first Li diode has been prepared and is presently being drifted. The present diode had to be acid etched a second time because a bath of trichloroethylene removed the Li exposed during the first etch. This bath has been used by other investigators and will be re-evaluated in future fabrications. The drifting apparatus was found to have sufficient adjustment in both the heating and cooling systems to maintain the diode at the desired temperature and current.

The deposition of the Li on the crystal proved to be difficult, but this was overcome by the use of a specially fabricated stainless steel boat such that the boat could be placed above the crystal for the deposition. A Vicor tube was used around the boat and crystal to catch the excess Li to keep it from contaminating the vacuum system.

Laser Extensometer

The laser extensometer, which measures dimensional changes of heated specimens, has been tested in the customer's furnace and data are presently being analyzed. At the present time the system is drifting and as yet the cause of drift is unknown. It appears that it may be due to a shift in the DC level of the output pulse. Tests are presently being performed to determine the source of drift.

Creep Capsule Facility

A 10 KVA isolation transformer was installed in the number 13 unit of the creep capsule facility. The GEMAC-silicon controlled rectifier temperature control system has been tested and is ready for use. The swelling research program now has four complete temperature control systems.

Work continued on the fabrication and installation of additional micro-positioner control circuits for the creep data logger. The completed motor control chassis was installed and tested. Operation is satisfactory in all eight channels. The MP control chassis fabrication was completed and is ready for installation and testing.

The four Allen-Bradley relays in the swelling heater power circuits were relocated to allow room for future expansion of the creep control instrumentation. Wiring instructions and circuit diagrams were prepared and sent to Irradiation Testing personnel for their use. Two unused saturable core reactor units were removed to facilitate a new and improved wiring layout.

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CANDU-BLW-250 Boiling Stability Study

Study of the relative effects of diagonal leakage and inverse node length calculations on the two-dimensional, four-node kinetics model of the CANDU-BLW-250 Reactor showed that they were small when compared with the effect of the geometric buckling calculation. Elimination of these calculations frees up some nonlinear equipment for use in the hydraulic model.

The complete hydraulics model has been received from the Thermohydraulics Unit and development of the analog simulation was started. Simplification to the model is required because of equipment limitations (insufficient number of multipliers).

Contacts with the customer have suggested that the neutronics model be changed from an in-out to a side-to-side model, i.e., one flow channel (two nodes) should be on either side of the center line instead of both flow channels (all four nodes) on one side of the center line of the reactor.

Containment Systems Experiments

The Nuclear Data analyzer core locations were transferred from 4000-5777 to 7000-10777 (octal) at the request of the system programmer. Also the Micro Tape interrupts, previously ignored when the EXTEND switch was on, have been returned to their normal mode; this means that the EXTEND switch has no effect on these interrupts.

Plutonium Metallurgy

Estimates were prepared for two jobs in the 308 Building. One job would provide computer control of inventory for fissionable material in the building and also control of potential criticality problems; the second involves reading analog recordings and calculating the statistical properties of the signals recorded.

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CHEMISTRY DEPARTMENTPLUTONIUM RECYCLE PROGRAMPRTR Study

The behavior of some 15 radionuclides in the aqueous and gaseous systems of the PRTR containment vessel following the fuel rod failure of September 29 was summarized for reporting. One of the interesting observations was that the 5000 cfm air coolers removed about 99% of the radionuclides (except for the noble gases) released to the air in the containment vessel. Had this air purifying action not been achieved, the deposition on the containment vessel surfaces rather than in the condensing water would have been about 100-fold greater and the decontamination would have been more difficult. The rate of deposition from the containment vessel air was sufficiently low that the coolers could provide effective removal.

FFTF PROGRAMDriver Fuel Plutonium Composition

During an evaluation of the use of mixed oxide rather than cermet fuel in the FFTF design, a concern developed that more Pu^{238} might grow into the mixed oxide fuel than the cermet fuel due to an $n,2n$ reaction on U^{238} . This would form Np^{237} and an n,γ reaction could then produce Pu^{238} . The concern was that appreciable Pu^{238} formation by this process could increase the radiation hazards and thus the costs of reprocessing and recycling the Pu.

Assistance from Engineering Physics was obtained in calculating plutonium isotopic compositions from the two fuel types. The results clearly demonstrated that there is no cause for concern. The small amount of Pu^{238} (7×10^{-3} wt %) that is present grows in mostly from the $n,2n$ reaction on Pu^{238} . In the mixed oxide fuel, only about 10% of the Pu^{238} grows in from the U^{238} source. The Pu composition in both fuels is well within the limits of compositions that have been processed in the Plutonium Fuels Laboratory and there is no reason to expect difficult handling problems to develop.

RADIOACTIVE RESIDUE PROCESSING DEVELOPMENTPhosphate Glass Studies

The coefficients of expansion of the phosphate glasses which will be used for startup of the WSEP facility were measured; they ranged from 7.1×10^{-6} per °C for a BNL glass (PW-1 composition) to 20×10^{-6} per °C for an ORNL glass (PW-2 composition). No serious problems are expected from differential contraction of solidified wastes and the stainless steel containers.

WSEP Design Verification Tests

Three runs (CSCM-4, -5 and -6) were made with the prototype spray calciner-melter system. After Run 4 the temporary three-zone induction-heated furnace was replaced with the prototype three-zone resistance furnace. The feed in Run 4 was PW-1 modified to melt by addition of H_3PO_4 to a metal equivalent/phosphate ratio (M/P) of 2.0; the feed in Run 5 was PW-2 modified to M/P = 2.0. In Run 6 the feed was modified with $NaNO_3$ and H_3PO_4 to give three feeds 0.83M Na and M/P of 2.4, 2.4M Na and M/P of 0.9, and 2.6M Na and M/P of 2.3. All final feed compositions were approximately 120 gal/ton of U equivalent. In all runs calciner wall temperatures were 700 C, melter furnace temperatures were 1150 to 1200 C, and air atomization was used.

Good operation was demonstrated during the two 66-hour runs with PW-1. Filter performance remained good, with pressure drop varying from 4 to 8 in. water depending on feed rates. Feed rates up to at least 27 l/hr were satisfactorily demonstrated. Periodic heating of a stainless steel receiver up to the melting point of the product was demonstrated as a reasonable technique for completely filling the receivers. Continuous melt discharge rates from the external weir were demonstrated at rates between 1.0 and 1.8 l/hr. By pressure-balance control, semicontinuous melt discharge rates up to 28 l/hr in batches up to 3.5 l were demonstrated. Discharge of the melt through the external weir was accompanied by significant spattering on the weir heater liner, particularly when discharging melts which were gassing and viscous. In the M/P range tested, relatively minor variations of PO_4 content had significant effects on melt viscosity. Good feed make-up control will be required for all runs.

The selected operating conditions of the spray calciner run with PW-2 produced volatilization of SO_3 in the melter and subsequent condensation in the cooler portions of the calciner.

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Spray-Calciner-Melter Development Studies

In run CSCM-4, stalagmites of solid product were formed indicating poor slumping of the product. Two samples of the product, one glassy and one devitrified, melted at 910 C; glassy product devitrified in the laboratory melted at 900 C. Apparently the temperature in the receiver was below 900 C during the run. Laboratory tests showed sodium to be less effective in lowering the calcine melting point at M/P = 2.0 than at higher M/P ratios.

Preliminary scanning of all 46 melts prepared for a statistical study of the effects of additives to PW-2 on melt and product properties was completed. None of the specific compositions prepared has the desired properties of retaining sulfate and having a stickpoint above 700 C. A statistical analysis of the data will be made to see if a composition having these qualities can be predicted.

An Othmer still is being used to obtain acid vapor-liquid data for simulated waste solutions. Data have been collected for PW-1 composition at 1, 2, 3, 4 and 5M H⁺ and at 75, 100 and 150 gal/ton U. Densities of the corresponding solutions at 24, 50, 75 and 100 C were measured.

Nickel, Corronel 230 and Hastelloy G were tested as possible materials of construction for the melter. All were exposed for 24 hours at 1100 C to PW-1 PO₄ and PW-2 PO₄ melts (both at M/P = 2.7). Nickel and Corronel 230 were completely dissolved by the PW-1 melt; Hastelloy G was severely attacked. Nickel and Hastelloy G were catastrophically attacked by the PW-2 melt; Corronel 230 suffered severe localized attack.

Scavenging Mechanisms

Sorption experiments were performed using the mineral pyrolusite as representative of MnO₂ having well defined crystallinity and acid condensate waste containing radoruthenium as the sorbate. Distribution coefficients were determined at various pH values. From pH 2.8 to 9.8 the K_d values exceeded 3000 ml/g, the maximum was 4000 at pH 4.9. The studies showed that radio-ruthenium in adjusted acid condensate waste (pH 5.1) is sorbed to <10% breakthrough at 400 column volumes and to <20% at 1250. Similarity exists between pyrolusite and precipitated MnO₂ in the sorption of radoruthenium from acid condensate waste. Work was planned to determine if this similarity can be ascribed to crystalline properties.

CONTAINMENT SYSTEMS EXPERIMENTMajor Facilities and Equipment

Closure of the containment vessel was completed early in the month by construction forces and the vessel was turned over to operations for leakage rate tests.

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The five part stress report on the reactor simulator vessel was received and was started. Struther-Wells Corp. was requested to provide a firm schedule for the balance of the work on the vessel when the delivery date of November 10 was passed.

Additional FY-1966 program funds will be used for features essential for limited aerosol studies following initial leakage rate tests. Major priority items include the aerosol injection line, aerosol sampling system, vessel ventilation loop and purge systems, and liquid level gages.

Leakage Rate Testing

The containment vessel was instrumented with 11 RTD sensors, a nickel resistance wire, a reference vessel, a static manometer, a differential inclined manometer and an absolute pressure gage. The RTD sensors were placed in positions appropriate to volume weighting methods for the leakage rate tests. The locations were determined by running horizontal and vertical temperature transients throughout the vessel.

The vessel was pressurized to 10 psig on November 17. Since that time, intermittent instrument readings have been taken to estimate the leakage rate. The observed leakage rate is 0.6% per day, with reasonable agreement between results of the absolute and reference vessel technique. Systematic testing of leakage at individual leak points was still in progress but the total leakage as derived from the integrated leakage rate tests is not yet accounted for by the sum of the individual leaks.

Instrumentation

Instrumentation for use in the leak testing generally performed satisfactorily. The resistance temperature detector (RTD) system provided measurements within its ± 0.5 F accuracy band. The single nickel wire, also used for large volume temperature measurements, apparently is subject to some drifting probably due to stresses set up in the wire at the insulating supports. Humidity data were provided by moisture detectors, following completion of laboratory calibration tests. An RTD and humidity cable terminal box and Conax fitting penetrations were added to the penetration panel.

A two-probe incremental water level detector was installed on the SM-2 vessel. Preparations were completed for autoclave performance testing of the liquid level conductivity probe and its ceramic seal at 2500 psi and 600 F. Information was developed concerning jet impact measurement and for installation of weldable strain gages. A prototype signal terminal box was made up with a special design to provide moisture-free routing of SM-2 vessel strain gage signal cables.

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Mathematical Modeling and Computer Programming

The manual portion of the code providing for the digital computer to run as a four input multichannel analyzer was made essentially operable.

Fission Product Aerosol Development

Tests in the Aerosol Development Facility emphasized demonstrations of CSE prototypic sampling equipment and a study of generation methods for Cs, Te and Ba aerosols. A twelve Maypack cluster assembly was successfully used in a steam-air containment test containing I_2 and UO_2 fumes. One deficiency was noted in that the check valves at the Maypack inlets (provided for stopping stray diffusion during non-sampling periods) retained excessive amounts of the I_2 . The cluster was next tested in a containment tank at pressures up to 50 psig and temperatures to 260 F for a period of 16 hours. The CSE sample flow control station was mocked up for this test. Both the cluster and the flow control station operated satisfactorily.

One aerosol containment test was made (Run S-48) using I_2 and stainless steel- UO_2 fumes. In this test, steam from demineralized water was generated by a small electrical boiler. Results were compared with previous tests which used boiler house steam. No significant differences were noted, which is reassuring in view of CSE plans to use boiler house steam until the reactor simulator is installed.

Generation of a Cs_2O aerosol by decomposition of Cs_2CO_3 was demonstrated for both milligram and gram quantities; generation of a TeO_2 aerosol by volatilization appeared to be practicable for both micro and macro releases. Deposition of both aerosols was dependent on carrier gas flow rate, with lower line and furnace losses at higher viscous flows.

Generation of a Ba aerosol by high temperature volatilization from carbonate sources was unsuccessful with releases $<0.2\%$ even at temperatures up to ca. 2500 C.

A stainless steel canned synthetic UO_2 fuel element, doped with appropriate stable and radioactive fission-product elements as used in the Nuclear Safety Pilot Plant at Oak Ridge was received and heated in the ADF furnace. At 1000 C the pressed end cap blew off and broke the quartz furnace tube. Another NSPP synthetic was requested from ORNL. The test will permit comparison of the CSE aerosol with that used in the NSPP.

Pilot Blowdown Experiments

Three experimental blowdown tests on the SM-2 vessel were made with 4 in. nominal, double rupture discs to check the performance of the rupture disc

triggering method prior to installing instrument sensors on the vessel. In the first attempt the vessel was heated to about 250 F by steam injection and pressurized to 180 psig, when the blowdown occurred prematurely. The cause was traced to a lack of disc back-pressure due to a leaky valve. Post examination of the discs showed a good rupture profile. The inner rupture disc failed shortly after steam heating was started in the next two attempts. The probable cause was that steam was released into the vessel in too short a time. Replacement of the steam throttling valves will be done before the blowdown attempt.

Program Planning

A comprehensive, detailed program outline was prepared and distributed for comment by AEC personnel, AIF consultants, and other interested parties.

Removal of Iodine from Containment Systems

Plans were made for the investigation of water spray systems with chemical additives and solid adsorbers. A start was made in the design of equipment to provide a stable methyl iodide aerosol of suitable concentrations and atmospheres to measure removal efficiencies for both filter media and chemical sprays.

SHIPPING HAZARDS

Experimental work was carried out with Purex-type waste simulated for 20,000 Mwd/ton irradiation and with M/P ratio of 1.5 for PW-1 type and M/P ratio of 2.5 for PW-2 type.

For M/P ratios of 2.6 or lower, the release rate of Cs^{137} from a 15 gram sample with 5 cm² exposed surface is <0.5% per hour at 1100 C. This apparent release rate applies to the PW-1 and PW-2 type materials with 20,000 Mwd/ton irradiation and with the PW-1 type up to 75,000 Mwd/ton. In comparison, previous results with PW-1 type material at M/P ratios of 2.8 and greater gave release rates of >1% per hour under the same release conditions. Attempts to establish a particle entrainment rate by using a Sr^{85} spike were not successful because of insufficient release. Release was $<10^{-5}$ with a flow rate of 3000 cc of argon per min.

OFF-GAS DISPOSAL TO GROUND

For many situations, the gravity component of the compressible potential was found to be as much as three orders of magnitude smaller than the pressure component. If the former is neglected, the problem may be formulated in terms of pressure rather than potential. The pressure formula gives rise to equations which have closed-form solutions; hence, a wide range of problems can be studied economically. The accuracy of this approach is being tested.

COLUMBIA RIVER SEDIMENT STUDIES

Contract scuba divers were used for five days of work in sampling Columbia River bottom sediments. This is the only system that has been found to work in rocky bottom areas. Samples of bottom sediment were obtained on nine cross-river traverses of the river between Ringold and McNary Dam. At least three bottom sites per traverse and three sediment depths per site were sampled. Results from these samples will help complete a radionuclide inventory of this reach of the river and will help explain the deposition phenomena.

APPLIED MATHEMATICS

General Reactor Technology

Applied and Reactor Physics

High Temperature Reactor Lattice Physics Studies

Work continued on the keyboard executive and the scan programs for the PDP-7 control computer for the HTLTR. Outlining of the display program for the colored alpha-numeric display tube is well underway. It will be required to display continuously the position of the horizontal control rods, the status of the vertical control rods, current time, maximum temperature, and flux, as well as alarm signals and operator demands.

Reactor Fuel and Materials Development

Nondestructive Testing

The EDPM program for the investigation of the generalized Snell's law for visco-elastic media, and the tabulation of reflection factors as a function of incidence angle has been extensively modified to provide data more compatible with laboratory needs. Theoretical investigations are in progress on the nature of power propagation at a plane boundary joining two dissimilar media.

Irradiation Damages to Reactor Metals

A study was initiated relative to an experiment design for an inter-laboratory calibration of tensile testing machines.

Quantitative Metallography

Physical Metallurgy and Applied Mathematics personnel have formed a team effort to document the several statistical techniques developed in the last three years by Applied Mathematics to quantitatively evaluate electron microscopy photographs of irradiated uranium. The effort will include, in addition to the documentation, the writing of individual computer programs to summarize and statistically evaluate Ziess particle-size data from photographs. The end result is to be a magnetic tape of quantitative metallographic programs which will be available upon request to Battelle-Northwest personnel.

Initial efforts are directed toward the documentation of the mathematical method for unfolding a particle size distribution and providing variance estimates of the individual cells of the resulting histogram. A Univac 1107 program SALTY is currently being debugged which performs this unfolding operation.

[REDACTED]

Nuclear SafetyReactor ContainmentContainment Systems Research

The main effort of PDP-7 programming was devoted to chain linking programs; that is, preparing a total system which will bring in segments of programs from micro-tape at the proper time, executing and restoring the original core area. Five such links now exist, but they have not been completely debugged.

A survey of available instrumentation systems for a gamma spectroscopy study of various isotopic mixtures of plutonium was conducted. It was concluded that the best system for the proposed gamma spectroscopy work consists of a lithium-drifted germanium solid state detector to take advantage of the unprecedented resolution obtainable in this way for the range of gamma ray energies above 100 KeV. It is necessary to supplement this system in the range of energies below 100 KeV with a lithium-drifted silicon solid state detector possessing a thin, dead layer and mounted in a cryostat with a thin beryllium window. Both systems will be operated at liquid nitrogen temperatures; the Ge(Li) to prevent deterioration by diffusion, the Si(Li) to reduce noise.

An extrapolation chamber measurement was made on two samples of Am^{241} to determine the surface dose from (1) a rod 1/8 inch long and 0.5 mm in diameter, and (2) a plane rectangular source of dimensions 3/16 inch by 3/8 inch. Ultimately, it was decided to calculate, based on the extrapolation chamber measurements, the mean dose to a disc in the geometry of the thermoluminescent dosimeters also used in the study. This permitted a comparison of the measurement and it represents a good approximation of the geometry of the thumb or index finger of a person handling the sources. The doses calculated from the extrapolation chamber data were:

$$D_{\text{rod}} = 4.7 \text{ rads/hr}$$

$$D_{\text{plane}} = 4.2 \text{ rads/hr}$$

These values are a factor of 3 lower than those obtained with LiF thermoluminescent dosimeters and a factor of 2.3 lower than those obtained with "Low-Z" glass rod dosimeters. Because of the large geometry corrections in the use of the extrapolation chamber on such very small sources, the measurement obtained with the other dosimetry systems are believed to be more accurate. The dose rate from the samples studied was about 12 rads/hr. The calculated surface dose rate for an infinitely thick slab of Am^{241} is 1500 rads/hr.

Copies of the Power Reactor Monitoring Guide were distributed to various AEC Regulatory Divisions, Public Health Service personnel and eight power reactor plants by the sponsor. Comments were requested on or before December 1, 1965.

TEST REACTOR AND ENGINEERING SERVICES DEPARTMENTPLUTONIUM RECYCLE PROGRAMTest Reactor OperationOperating Experience

There were no operating periods during the month. All activities were directed toward recovery efforts from the FERTF fuel failure of September 29, 1965, and toward preparations for chemical decontamination of the primary system. Progress during the month included the following:

1. The process tube was removed from channel 1550. It was necessary to remove the top shield plugs to permit the passage of the defected portion of the process tube.
2. Borescopic examination of the inside of the shroud tube reveals there is no hole in the shroud. There is an area, approximately one inch in diameter, which is directly opposite the hole in the process tube that has a sand-blasted appearance. A plastic mold of this area was obtained. This mold indicates that there are small quantities of material adhering to this area and there are several small pits of approximately 5 mils depth in this area, but there is no shroud tube distortion.
3. The process tube bellows was removed from channel 1550. Efforts are in progress to evaluate how much core material is present in the bellows. Procedures for removal of core material from this and other bellows are currently being prepared.
4. Radiation probing inside the calandria indicated that the major accumulation of core material "fines" are on the top of the calandria top tube sheet and that smaller amounts of this material are present in the bottom of the calandria and the shim cooling recesses.
5. Significant progress has been made in the cleanup of the containment vessel floors, walls, and equipment. The two remaining surface areas requiring extensive effort are the upper and lower access spaces, and these efforts are in progress.
6. All of the dry gas lines in the hot shop have been cut to permit flushing from the inlet bellows to "C" Cell sump and from the -21' rotameter rack to "C" Cell sump.
7. With the exception of the gasometer, the helium pressurization system, including compressors, etc., has been made ready for service.
8. The primary system D₂O has been drained and stored in tanks and drums in the north tunnel of the service building basement. The system is currently filled with demineralized H₂O.

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Maintenance Experience

Significant maintenance items accomplished were the removal of 1550 bellows and top shield pieces, removal of the dew cell refrigerator and moisture detection panel, cutting of the 90 dry gas system sample lines, removal of six shim rods and installation of dummy rods, shielding fabrication of several types, and fabrication of many cleaning and remote handling tools. A total of 2,872 craft mahours were utilized during the November outage as follows:

FERTF Fuel Failure	939
Decontamination	579
Repair	532
Modification & Improvement (mainly capital work)	257
Operations Assistance	290
Preventive Maintenance	225
	<u>2 872</u> manhours

Improvement Work Status

Work Completed

Gas Storage Facility Addition
Rupture Loop Makeup Seal Pressurization
Primary System Purge Equipment

Work Partially Completed

Hydrogen Addition to Bulk Storage
Hydraulic Thermal Barrier Lift
B Cell Lighting Improvements
New Instrument Power Supply
Rupture Loop Tunnel Air Monitor
Traveling Flux Wire Probes
Reactor Hall Crane Work Platform
Automatic Air Vents, Recirculation Units
Moisture Detection System
Change Reflector System from D₂O to H₂O
PRTR Storage Basin Addition

Design Work Completed

Differential Pressure Measurement - Ventilation Bypass Line
Hose Connection Survey
Shortened Outlet Nozzle

Design Work Partially Completed

PRTR Experimental and Building Facilities Addition
Power Increase
Liquid Waste Handling Project
Fuel Transfer System Improvements
Demineralized Light Water Cooling System for Reactor Equipment
Outlet Nozzle Flow Restrictors
Instrumented Fuel Element Installation
Analysis of HX-1 and Primary Pump Supports
Lower Access Space Shielding
Remote Tool for Lower Face
PRTR Power Calculator
PRTR Conversion to Single Speed Reverse Face Drive

Process Engineering and Reactor Physics

A study of the effect of an H₂O reflector on the adequacy of existing flux monitoring equipment has been completed. The existing equipment will be adequate for power operation, although the range of adjustment available to operations will be rather small. It will be difficult to get the instruments on-scale for startup tests.

Preparation of the formal report of the PRTR FERTF Incident of September 29, 1965, was begun. The detailed outline of the report and the general descriptive sections were completed. The investigation of fission-product-gases released to the containment vessel as a result of the incident has been completed for inclusion in the report. A fission product material balance is being prepared for the AEC Investigating Committee.

A calculation has been made of the air activity that will result when fuel particles are irradiated in the dry gas space. Based on the past dry gas leakage and an assumed particle size of <12 microns, the reactor hall noble gas air activity limit will be reached if 24 grams of fuel is irradiated in the dry gas space above the calandria or 0.2 grams of fuel is irradiated in the shroud tubes.

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FAST FLUX TEST FACILITY PROJECTConceptual DesignOverall Conceptual DesignBattelle-Northwest Studies

Conceptual design studies are being actively pursued in reactor vessel arrangement, heat removal systems optimization, instrumentation, and building arrangement. Various engineering drawings and graphics work were redrawn or modified to be consistent with the latest overall design concept. The text of Volume 2, BNWL-CC-175, was also revised to conform to the most recent concept. Both text and drawings will be incorporated in a new release of the conceptual design description during January 1966. The new release will constitute a working reference concept for the alternate study contractors and for BNW in its continuing effort.

Special conceptual studies are active at the present time. A rather intensive survey was begun to ascertain the advantages (and disadvantages) of decreasing the core lattice size to achieve a more subdivided core assembly of more numerous fuel positions. Alternate reactor building arrangements are being studied to improve the functional relationship of various process and maintenance activities. For the present concept, a study of the lower containment structure of the reactor enclosure will be directed toward improvement of the structural efficiency of mass concrete and structural steel.

Industrial Design Reviews and Studies

Preliminary contacts were made with several industrial companies to explore their possible participation in the FFTF project conceptual design studies.

Cost StudiesCapital Cost Estimates

The FFTF design and construction project, as defined by Volume 2 of the Progress Report, Conceptual Design, BNWL-CC-175, was cost-estimated by Bechtel Corporation with the assistance of Atomics International. The Bechtel-AI estimate was quite similar to the budget cost estimate for the same work scope prepared at Hanford by BNW-FFTF and Vitro personnel. In effect, a verified cost estimate (budget level) was obtained.

Immediately upon completion of the estimate for the "Volume 2 Concept", Bechtel prepared a cost estimate for the latest conceptual design developed by BNW. The new estimate shows a cost reduction of about 11% for the new design as compared to the "Volume 2 Concept".

Other Cost Analyses

Design Optimization

A general method has been devised for optimizing the design of the FFTF. Relative values are estimated for the useful range of each irradiation testing variable (flux, temperature, test dimensions, etc.). Trade-off factors are established, values are normalized, and a composite value function is derived. Similarly, costs are estimated for the same ranges of each major test variable. All costs--including fuel cycle costs, operating and maintenance costs, capital costs, and incremental costs to achieve improved plant availability--are added to yield a total cost function. The performance index is the ratio of these two functions:

$$\left[\text{FFTF} \right] \text{Optimum} = \left[\frac{f(\text{value})}{f(\text{cost})} \right] \text{Maximum}$$

Parametric Cost Studies

Analyses were conducted to explore the costs of various power levels and fuel configurations for the FTR cermet-fueled system. Parameters examined included:

1. Power levels: 200-600 MW
2. Core volume: 200-1900 liters
3. Number of subassemblies: 37-91
4. Pins per subassembly: 169-397
5. H/D ratios: 1.0 - 0.75

Optimum fuel cycle costs were found to lie in the range of 250 to 400 MW, with 61 subassemblies and 271 pins per subassembly.

A similar analysis will be conducted for the mixed-oxide core.

Nuclear Design

Fast Reactor Cross Section

The two-dimensional DDK transport theory code was used to calculate the reactivity for the Fermi Core B critical assembly using the RUSS II cross section set. Results of similar calculations using the ANL-224, Hansen-Roach, and the original Russian set were reported last month, but are included for comparison. All computations were performed with 4 energy groups derived from the above sets using the CRUNCH group collapsing code.

Calculated k_{eff} - Fermi Core B

<u>Cross Section Set</u>	<u>k_{eff}</u>
ANL-244 Set	0.985
Hansen-Roach Set	1.018
Russian Set	1.032
RUSS II Set	0.988

FTR-MCA Analysis

The methods and models to be employed in calculating the energy release during a maximum credible accident (MCA) in the FTR were discussed with Nuclear Technology Corporation (NTC) personnel. NTC has been engaged as a consulting firm for the FFTF Project. A program was outlined to evaluate the presently available computer codes and analytical models. The following topics will receive major attention:

1. Mechanism which initiates the accident and the means by which the core becomes supercritical.
2. Effect of distribution of fuel in the core.
3. Geometry of the supercritical core.
4. Doppler temperature coefficient.
5. Type of fuel in the core.
6. Material parameters (i.e., equation of state) of the reactor constituents.

The results of this investigation should indicate the sensitivity of the calculations to various assumptions.

 Pu^{238} Buildup in the FTR Driver Fuel

Processing of FTR driver fuel could be affected by the presence of Pu^{238} . To assess the severity of this problem, some very preliminary estimates have been made of the buildup of Pu^{238} in cermet driver fuel (PuO_2 -SS) and in ceramic driver fuel (PuO_2 - $U^{238}O_2$) in a 400 MW, 600 liter FTR reference core. Since the available cross section data is quite uncertain, some rather crude assumptions had to be made.

The results of the calculations are tabulated below for several irradiation intervals.

 Pu^{238} Buildup in FTR Driver Fuel

<u>Time, days</u>	<u>PuO_2-SS Driver ppm Pu^{238}/Pu</u>	<u>PuO_2-$U^{238}O_2$ Driver ppm Pu^{238}/Pu</u>
0	0	0
40	18	20
80	35	41
120	53	64

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Facilities Systems Design

Reactor Heat Removal System

For optimization purposes, the reactor heat removal system has been divided into two segments: (1) heat removal, and (2) heat dissipation. The heat removal system consists of the primary circuit, the intermediate heat exchanger and secondary pumps, while the heat dissipation system consists of secondary piping and the final heat dump systems. These systems will be considered separately (to the extent feasible) in optimization studies. Four leading concepts for the heat removal system are under detailed study for a reactor power level in the range of 400 MW. Studies are also under way on heat dump optimization.

Fuel Handling, Examination, and FFTF Equipment Maintenance

Atomics International has accelerated work on alternate fuel handling machine configurations in an attempt to reduce the height of the machine by devising new fuel transfer techniques that will require a shorter reactor tube. AI now has one or more personnel on each of three concepts.

Sodium Support Facility

Preliminary engineering studies were continued to define the requirements for the prototypical sodium facilities. A suitable location for this complex appears to be the basement of the 324 Building. Several alternate locations are under investigation.

Drafts of technical specifications were prepared for heat transfer and fuel washout-decontamination prototype facilities.

Design Analysis

Core Stress Analysis

The SEAL-SHELL 2 code for thermal and mechanical loadings of axisymmetric vessels is now operating on the 1107 computer. The difficulties in converting this "working" Fortran IV program were due to the Univac Fortran IV compiler losing variables between subroutines and the main program. This problem can be eliminated by inserting redundant statements in the main program at return points from subroutines.

First steps were taken toward acquiring a finite-element code for axisymmetric structures from either Bettis or General Atomic. Such a code would provide very detailed stress pictures for fatigue analysis.

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Containment Analysis

After a review of experimental work by the Naval Ordnance Laboratory for Enrico Fermi reactor, the present FFTF design is considered to have more than adequate protection against any initial shock wave (microsecond phenomena). This protection is provided by the multiple layers of steel cylinders and piping around the core. The venting of explosion vapors from the core will be a major way to (1) prevent extra high pressure from developing on the cover plug and (2) keep the water hammer phenomena minimized. Both of these problems result from the quasi-static pressure (millisecond phenomena). Concept drawings of the reactor core now incorporate reduced shock wave absorption material and have a vent path which would come into effectiveness only on a maximum credible accident.

Fuel Subassembly Analysis

TIGER 5, a transient thermal digital computer program, has been modified to calculate temperature profiles in an FFTF fuel bundle. In addition, a secondary program has been added to TIGER 5 which calculates nodal heat generation rates based on any radial and axial flux profile in the core. To date, preliminary steady state results have been obtained for the fuel subassembly. These first results are being checked by hand calculation to insure the correctness of the program. In the next report period, the analysis of normal transients (e.g., scrams, startup, etc.) will begin.

Hydraulic Analysis of the FTR Driver Fuel Element

The hydraulic characteristics of the fuel element (conceptual design) were analyzed. Also considered were the variation of most flow parameters, the possibility of leakage flow between annulus channels, and the strength of the strips forming the grid spacers to withstand the fluid forces.

For a sodium flow of 57 lbm/sec, the fuel element pressure loss will be between 100 and 130 psi. The leakage flow between channels for an annulus pressure loss of 70 psi amounts to 6.4×10^{-3} percent of the annulus flow. The grid spacer strips will withstand the forces due to a sodium velocity of 50 fps, with a large factor of safety.

Since the grid spacers are the major contributors to the fuel element pressure loss, they may require redesign in order to reduce this loss. One possible design makes use of parallel zigzag strips that are arranged so that adjacent strips are on different planes and every other strip is on the same plane.

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FFTF Fuel Pin Analysis

Work has begun on the detailed analysis of the temperature profiles of single FFTF fuel pins under steady state and transient conditions. This study will include a parametric operating condition of all candidate fuels.

Due to the tricuspid shape of the coolant passages in a rod bundle, the surface heat transfer coefficient and the coolant sink capacity vary circumferentially around a fuel pin. The analysis of this effect is based on a method developed by O. E. Dwyer.*

FFTF Dynamics Model

Development of the FFTF dynamics model continued in the areas of reactor and primary loop components. The responses of logarithmic and linear, step-wise reactor startup models to selected input disturbances were compared, with good results. The effects of primary coolant inertia were included in the primary loop hydraulics model. A heat exchanger model using a log-mean ΔT calculation is being prepared for analog computer test runs during the next reporting period.

The four segments of the analog controller simulation program were debugged and combined to form the complete program. All the necessary cabling and computer connections were made for controlling a process and recording the input to and output from the G.E. 412. A portable analog computer was connected to simulate a second order process with a variable transport lag. This simulated process, with lag set to zero, was controlled automatically by using the simulation program. Preliminary data show that good control is possible.

Program Control ManagementPERT Computer Applications

Several PERT computer programs were located and a test run was made on each of the following:

1. NASA PERT "Charlie"
2. NASA PERT Time I
3. USAF PERT/Cost

*O. E. Dwyer. "Analytical Study of Heat Transfer to Liquid Metals Flowing In-Line Through Closely Packed Rod Bundles," paper submitted to Nuclear Science and Engineering.

The NASA programs ran satisfactorily but the USAF PERT/Cost program failed to run due to memory restrictions in the available computer systems. Attempts to adjust the USAF program for a test run will be continued.

Other methods of applying computers to the PERT program are being sought. A plotting routine which plots the PERT Chart from computed time data is on order and will be tested when received.

FFTF Technical Interchange

BNWL-CC-331, Neutronic Characteristics of Various Driver Fuels for the Fast Test Reactor (FTR), W. W. Little, October, 1965.

BNWL-CC-341, Neutronics Calculations for a Modified EOCR for Fast Reactor Fuel Testing, P. L. Hofmann, L. L. Maas, R. H. Holeman, November 4, 1965.

BNWL-SA-244, Physics Aspects of the Fast Flux Test Reactor, P. L. Hofmann and W. W. Little. Paper presented by W. W. Little at the Winter Meeting of American Nuclear Society, Washington, D. C., November 15-18, 1965.

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D. DIVISION OF PHYSICAL RESEARCH - 05 PROGRAM

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SUMMARY OF ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTRadiation Effects on Metals

Irradiated nickel foils and quenched nickel foils have been found to show no small angle x-ray scattering effects. The results are interpreted as indicating that the defect clusters are planar in nature.

The majority of the defects observed by electron microscopy in as-irradiated high purity nickel have now been established as being vacancy in nature.

Irradiation-induced length changes in molybdenum crystals are partially recovered during annealing at 200°C for crystals irradiated to fairly low exposures. After higher exposures ($>4 \times 10^{19}$ nvt) no length-change recovery has been observed at 200°C.

Plutonium Physical Metallurgy

The dependence of beta plutonium physical properties on the parent phase from which it was formed--alpha or gamma--has led to an investigation of the basic differences introduced in this phase by its mode of formation. Subtle differences have been noted.

PHYSICS AND INSTRUMENTS DEPARTMENT

See Page D-7.

APPLIED MATHEMATICS DEPARTMENT

Several modifications were made to the GEM program to improve the handling of background spectra. When presented with a set of background spectra the GEM program now determines whether or not the variability among these spectra is in agreement with counting statistics. If so, the spectra are averaged and a single background spectrum is used to correct sample data. When variability among the background spectra is excessive, that spectrum which is closest in time to each individual sample spectrum is used for background correction. Two Univac 1107 versions of GEM were put into production this month.

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ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTRADIATION 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. Polycrystalline specimens of molybdenum and nickel were studied during the month.

Nickel foils irradiated to 2.5×10^{18} and 5×10^{18} nvt ($E > 1$ Mev) have been examined by the small-angle X-ray scattering technique. No measurable scattering has been observed for any of these samples, despite the presence of the high density of clusters which appear in transmission electron micrographs. The most likely explanation of the absence of scattering is that the visible defects are unresolved dislocation loops enclosing a stacking fault. Such a configuration would not lead to local inhomogeneities in electron density, and so would not produce small-angle scattering. The dark field electron microscopy has shown the presence of many very small dislocation loops which correlates with the small-angle X-ray scattering interpretation.

Scattering is also not observed for nickel foils quenched from near the melting temperature. These samples are known to contain vacancy defect clusters, but in this case the clusters must also be in the form of planar type defects with no great effect on electron density. These samples will be aged at successively higher temperatures in an attempt to alter the defect configuration.

The previous monthly report stated that the majority of the defects in neutron irradiated nickel were interstitial in nature, as determined by transmission electron microscopy. It has since been determined that an error was made in the analysis of the dark field micrographs and that the majority of the defects are vacancy in nature. Previous interpretation required the presence of predominately interstitial type defects because vacancies were too immobile at the irradiation temperature to form a cluster or dislocation loop. This recent analysis requires a new interpretation of the radiation damage state. It must now be presumed that the vacancy defects originate directly from the displacement spike. This would require little vacancy migration.

As mentioned previously, the overall density of the cluster defects decreases during annealing. This can be interpreted as being due to the annihilation of the interstitial cluster defects by free vacancies which originate from the break-up of the vacancy defect clusters or loops.

Similar specimens of high purity nickel which have been irradiated to a higher dose level, $\sim 5 \times 10^{18}$ nvt, are currently being examined. At the higher exposure level, it is hoped that sufficiently large dislocation loops will have formed so that independent crystallographic analysis can be made and that results can be compared with those obtained from dark field analysis.

Small-angle X-ray scattering from foils of molybdenum irradiated to 1×10^{20} nvt ($E > 1$ Mev) have been confirmed in numerous specimens. Secondary maxima, indicating interparticle interference, have been observed. Rough analysis of these secondary peaks indicates an interparticle distance of approximately 550A and a particle density of about $6 \times 10^{15}/\text{cc}$. No scattering has been observed from molybdenum foils irradiated to only 1×10^{19} nvt, even though these foils are known to contain a 1×10^{16} interstitial cluster per cc, as estimated by electron microscopy. It is therefore logical to conclude that at higher neutron exposures, something other than the clustered defects visible by electron microscopy is responsible for the scattering. The scattering at higher exposures must arise as a result of clustering of vacancies or alteration in the configuration of interstitial clusters.

Low-temperature annealing experiments on irradiated molybdenum single crystals are continuing. A molten-salt bath has been substituted for the silicone oil bath previously employed. The results of these experiments to date show that irradiation-induced length changes are partially recovered at 200°C only in specimens receiving a comparatively low exposure, 1×10^{19} nvt or less. A sample irradiated to 4×10^{19} nvt has shown no recovery of length increase after a 260°C anneal, and another sample exposed to 1×10^{20} nvt shows no recovery after a 400°C anneal. Apparently vacancies form very stable clusters after a critical concentration is reached. These annealing experiments will be extended to still higher temperatures.

Polycrystalline molybdenum tensile samples irradiated to 5×10^{18} nvt ($E > 1$ Mev) have been tested in the as-irradiated condition at temperatures of 350, 323, and 295°K. Strain rate sensitivity data were obtained on samples at 350 and 323°K, but the low ductility of the samples prevented tests at 295°K. The data are currently being analyzed.

Further testing of the ZrH₂ purification system was carried out during this period. Molybdenum tensile samples with a 0.080 inch diameter reduced section were treated for 24 hrs, 120 hrs, and 240 hrs at 1300°C. There was no significant change in the mechanical properties of the treated samples. Some operational problems were experienced during the 240 hr run that may have had a detrimental effect on the purification process. Further tests are in progress to ascertain the proper conditions for molybdenum purification in this system.

A paper, entitled "Neutron Irradiation Damage in Rhenium," was presented at the 1965 Fall Meeting of the Metallurgical Society of AIME. A second paper, entitled "The Combined Effect of Carbon and Neutron Irradiation on Molybdenum," has been published in the November 1965 issue of Acta Metallurgica.

PLUTONIUM PHYSICAL METALLURGY

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

Phase Transformations in Plutonium

The significant differences in the strength as well as other physical properties of beta phase plutonium formed from the alpha phase as contrasted to that formed from gamma has warranted a detailed study of the basic differences responsible for the property variations in the two types of beta material.

The X-ray diffraction pattern difference in the two beta materials has previously been reported. Slight shifts in atom positions within the basic unit cell of the beta structure may account for the change in relative line intensities noted. This is particularly true since there is no apparent difference in the density of the two beta phase materials. The effect of atom position changes on line intensity is under investigation. No conclusions can as yet be drawn from this area of study.

Electrical resistance measurements have been employed in the study of the reaction kinetics of the transformations of the two types of beta to gamma, $\beta_1 \rightarrow \gamma$ and $\beta_\gamma \rightarrow \gamma$, as well as the gamma to beta, $\gamma \rightarrow \beta_\gamma$. In all cases the resistance passes through a definite minimum during the transformation process. This value is several percent of that of the parent and daughter phases. In the case of the gamma to beta transformation the resistance passes through a maximum as well as a minimum value during the transition. An attempt to correlate these effects with a possible transformation process is underway.

Diffraction studies directed toward establishing, if possible, the crystallographic relationships between alpha and beta plutonium are continuing. Alpha plutonium which had been textured by transformation from beta under a compressive load was converted into the beta phase by heating in the diffractometer furnace. From an initial cursory survey, the beta pattern appears to be similar to that of a random

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structure. The alpha phase, which is obtained from a subsequent reverse transformation, apparently has a texture which is essentially 90 degrees to the original. A paradox therefore exists: how can an $\alpha \rightarrow \beta \rightarrow \alpha$ transformation series show no textural relationship between the first part of the series, $\alpha \rightarrow \beta$, and yet show some after the final part of the series. An explanation, of course, would be that some texture is present in the beta. A more detailed study of the complex beta pattern may reveal this texture.

Some additional creep data for both β_1 and β_2 are being obtained to confirm and extend current information on these two types of beta material. Data necessary to verify the effects of tensile stress on the $\beta_2 \rightarrow \alpha$ transformation kinetics are also being obtained.

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PHYSICS AND INSTRUMENTS DEPARTMENTISOTOPIC ANALYSIS PROGRAM

Isotopic analyses were provided on program samples at a greatly increased rate during the month. Intensive operation of the program mass spectrometer for a two-week period effectively demonstrated that the alumina vacuum lock and scintillation-type ion detector have more than doubled the analytical capacity of the instrument.

A paper tape punch and coupler have been modified to permit data output from the program mass spectrometer via punched paper tape. A program to read the tape and perform statistical calculations on the data is being written for the small digital computer. This will eliminate the present method of data reduction which requires as much time as the analysis itself.

Design of the permanent magnets for the new three-stage mass spectrometer was completed during the month. The magnets are being constructed on site. On a cost basis, design or procurement remains to be initiated on only four percent of the components of the spectrometer.

APPLIED MATHEMATICS DEPARTMENTChemistry ResearchActinide Element ChemistryIndexing of Crystals

A supplement to BNWL-22 was issued which contained an errata list, corrected program listings, and suggested improvements or variations in method. A joint paper on the topic has been accepted for presentation at the Conference on Computational Methods in Crystallography scheduled in London.

Special ProjectsStatistical Evaluation and Development

Several modifications were made to the GEM program to improve the handling of background spectra. When presented with a set of background spectra the GEM program now determines whether or not the variability among these spectra is in agreement with counting statistics. If so, the spectra are averaged and a single background spectrum is used to correct sample data. When variability among the background spectra is excessive, that spectrum which is closest in time to each individual sample spectrum is used for background correction. Two Univac 1107 versions of GEM were put into production this month.

A debugged version of Fortran language program PEAK is now complete. PEAK locates photopeaks in gamma pulse height spectra using only the mathematical relationship between the resolution of the crystal involved and energy. In its present form the program graphs the raw spectra which is not linear up to counting statistics in which a single peak model is fitted to the spectrum and writes out a summary of all regions of the spectrum so investigated. Thought is now being given to the statistical question of how to best fit a multiple peak model to a portion of the spectrum.

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Section E

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SUMMARY OF ACTIVITIESPHYSICS AND INSTRUMENTS DEPARTMENTAtmospheric Physics

November was an active month for diffusion and scavenging field experiments. A total of eleven experiments was successfully completed using a variety of tracer materials that included zinc sulfide, fluorescein, xenon-133, hydrogen chloride, zinc chloride, bromine and radioiodine. Five diffusion experiments were completed at Hanford in programs designed to study deposition, effect of source height and effects of turbulence in building wakes. Five additional tests were conducted in natural rain to determine washout coefficients for several chemicals. The first test in the Air Force Mountain Iron Program was successfully conducted during November.

Radiological Physics

A theory was developed that relates target theory, local-energy-density theory, and absorbed dose theory. The details have not all been worked out yet, but the new theory promises to suggest some interesting new tests for radiobiological data.

New Fe⁵⁵ measurements show that the amounts in people are increasing and that the amounts in people (Eskimos) who eat a lot of fish are higher than those in people who do not.

Instrumentation

Physical well being of personnel working in adverse environments is being assured through use of recently developed and improved physiological monitoring equipment which telemeters body temperature and electrocardiographs to an outside station.

CHEMISTRY DEPARTMENT

All of the major hydrology digital computer programs have now been converted from the 7090 to the 1107 computer in order to benefit from a resultant increase in problem-solving time efficiency.

The general gravity survey of the Hanford area is progressing well.

Geologic mapping just north of the Horse Heaven Hills between Kiona and Ice Harbor Dam gives further evidence that the confined aquifer containing ground waters from the 200 Areas may emerge near the Snake River-Columbia River junction.

Study of the removal of trace material from the atmosphere by rain was continued with four experiments using ZnCl₂ and Br₂ as a tracer. Analysis was by neutron activation. Atmospheric dispersion was measured to distances of one mile using radioxenon.

Programs have been devised which permit computer analysis of electron spin resonance data.

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BIOLOGY DEPARTMENT

Both in vitro and in vivo studies in rats have shown a severe depression of sodium transport in the irradiated intestine. In vivo studies indicate that the movement of sodium and water into the intestine is not increased by radiation. Thus the diarrhea characteristic of intestinal radiation damage must be attributed primarily to impaired reabsorption.

In normal rats there is a 10-15 fold increase in blood triglycerides following injection of the non-ionic detergent, Triton. There is no such elevation in female rats pretreated for 24 hours with ^{237}Np , suggesting that the fatty livers characteristic of ^{237}Np poisoning may be due to impaired secretion of fat by the liver.

Studies of ^{65}Zn uptake by marine amphipods from sediments indicate that such uptake is dependent upon leaching of the ^{65}Zn into the sea water and does not occur by direct uptake from the sediments.

Rats subjected to 800 R thoracic x-irradiation showed a 100% increase in lung elastin, and a 60% increase in lung lipids and lung lipid phosphorus.

Some rather involved microbiological experiments have suggested an interesting model for ribonucleic acid (RNA) synthesis in which ribonucleoprotein particles play a key role as feedback inhibitors of RNA synthesis from precursors.

Moss from the forest floor, collected near Packwood, Washington contained 10 times the ^{137}Cs concentration of seed plants from the same area - values higher than those observed in the Alaskan lichens. Transfer to vertebrate herbivores is unlikely since moss is not palatable to these species.

Peak salmon spawning occurred during the first week in November. The 1778 nests counted between 300 Area and Priest Rapids was the highest count on record.

One PuO_2 particle fed to a pig was held up in the intestine in an encapsulated inflamed area for 14 days postingestion.

APPLIED MATHEMATICS DEPARTMENT

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RADIATION PROTECTION DEPARTMENT

Teflon impregnated lithium fluoride dosimeter discs were studied. The discs are 1/2 inch in diameter, about 0.015 inch thick, and have about 20 mg of Li^7F mixed with the teflon. Sets of five discs were exposed to 50, 100, 200, 400 and 500 mR of radium gamma radiation and readout with special planchettes. The background reading is approximately 15 mR equivalent, and the system is adequate to detect doses of 100 mR or more. These dosimeters are being studied as possible replacements for the currently used finger ring film. They offer the advantages of a low detection limit, ease in handling and no darkroom requirements.

The third and final phase on an inter-laboratory calibration study under the auspices of Division of Biology and Medicine, U.S. AEC, was successfully completed during November. The purpose of this program is to compare calibrations for cesium-137 between several of the major whole body counter laboratories in the United States. The third phase of the program consisted of measuring two visiting human standards in the Battelle-Northwest Whole Body Counter, one of whom contains a "known" elevated cesium-137 burden, and the other who contains the "normal" quantity of cesium-137.

Battelle's portion of the research program, "Correlation of Lifetime Health Experience of AEC and AEC Contractor Employees with Occupational Exposure", is being reviewed to determine the cost of providing data for a continuing program after the initial phase is completed.

ACTIVITIES

PHYSICS AND INSTRUMENTS DEPARTMENT

ATMOSPHERIC PHYSICS PROGRAM

Many diffusion and precipitation scavenging experiments were successfully completed in November. The fifth successful combined release of Xenon-133, zinc sulfide and fluorescein was completed and the zinc sulfide assay was made. Two additional releases of zinc sulfide were made from the 85- and 185-foot levels of the Hanford Tower and air samples were obtained near the ground out to two miles downwind. In this program, eight tests have been completed this year. A new test series designed to experimentally determine the diffusion of particulates in the wake of buildings got under way. A grid was constructed that consists of three arcs, each more than 300 degrees in extent. The arcs are at distances of 30, 50 and 100 meters from the center of an abandoned cold storage building at White Bluffs. Two tests were completed in which zinc sulfide was released upwind from the 10-meter high building and samples were obtained in the building wake. Assay of the samples is continuing. These data will be correlated with wind and temperature difference measurements obtained during the tests and compared with those results obtained for undisturbed flow in similar atmospheric conditions. In off-site programs, the first diffusion test for the Air Force in the Mountain Iron series, at South Vandenberg, California, was successful on November 10. Sample assay was completed at Hanford. Significant counts were found at the outer sampling road, five miles downwind from the source. Heavy rain that began in the middle of the month has temporarily halted these tests.

At Hanford, the winter rains provided an opportunity to experimentally determine washout coefficients for a number of tracer materials. In four tests conducted in the middle of the month, zinc chloride, hydrogen chloride and bromine were released from the Hanford Meteorology Tower and samples of natural rain were obtained downwind. The fraction of the tracers washed out per unit time will be determined from the sample assay. On November 24, an additional natural rain scavenging test was successful using the Purex Plant stack gases as the tracer. These samples were submitted to Radiological Chemistry for determination of organic and inorganic radioiodines and will yield additional washout coefficients for these components.

Organization and summarization of turbulence data continued during the month. Spectra for the lateral wind component were compared in a manner similar to that of the other two components described in the two previous monthly reports. $n S_v(n)/\sigma_v^2$ was plotted against $n z/\bar{u}$ for unstable, neutral and stable groups. The data showed considerably more scatter than was noted for the other two component spectra. This scatter results from the large variations in the energy introduced at low wave numbers, particularly in the neutral and unstable cases. The total normalizing variance, σ_v^2 , also varied greatly in these cases because of the variation in the contribution of energy. Large amounts of turbulent energy and large eddy sizes are most noticeable in unstable atmospheres where no specific peak in the spectral distribution was discerned. In the neutral cases, a peak was always found near $n z/\bar{u} = 0.1$ cycles with other large peaks found at lower wave numbers. In the stable cases, a peak was always found at about $n z/\bar{u} = 0.1$ cycles with turbulent energy decreasing for lower wave numbers. For all cases, the spectra drop off with increasing wave number beyond $n z/\bar{u} = 0.1$.

Weather Summary

<u>Type of Forecast</u>	<u>Number Made</u>	<u>% Reliability</u>
8-Hour Production	90	80.7
24-Hour General	60	82.9
Special	97	78.4

Number of calls received on automatic pre-recorded forecast machine: 3,697.

November was considerably warmer than usual and became the first month of the year to record above normal precipitation.

RADIATION PHYSICS PROGRAM

The optical spectrometer for the time-resolved optical spectrometry system just arrived. The fast-pulse averaging part of the system was completed with addition of a circuit for counting the number of averages taken.

The differential pumping system was added to the counter for the measurement of proton energy losses. The resulting pressure differentials were about as expected. However, the proton pulse-height distribution observed in a solid-state monitor counter indicates that scattering in the edges of the collimator may be taking place.

Part of the measurements of optical-discrimination phenomena in NE-102 scintillating plastics were completed.

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Mitsumi Kinoshita of the Japan Atomic Fuel Corporation completed a training assignment in the Van de Graaff laboratory.

RADIATION DOSIMETRY PROGRAM

The counter for local-energy-density measurements without use of a grid was assembled and tested. The counter had good resolution. Electrostatic shields around the counter strongly influenced the pulse-height distributions obtained; therefore, final testing of the counter will not be possible until the tank for the "wall-less" counter is received.

With the help of Nucleonic Instrumentation a circuit is being prepared that will permit artificial generation of local-energy-density spectra from given spectra of individual events in local-energy-density counters. This will reduce the number of experimental spectra required. It will also permit us to make corrections to experimental spectra and to study the effects of changes in spectra.

A theory was developed that relates target theory, local-energy-density theory, and ordinary absorbed dose theory. Not all the details have been worked out, but the new theory promises to suggest some new experimental tests for radiobiological data. The new theory employs local-energy-density spectra in ways not envisioned before, but they are ways that should be easier to handle computationally.

Tech Shops completed fabrication of an absorbed dose calorimeter.

Improvements in commercially available instruments for measuring capacitance led us to re-examine the possibilities of using gas thermometry, with pressure changes translated into capacitance changes, in radiation calorimetry. Tests of a transducer we prepared indicated sensitivity much less than a microdegree should be possible.

A cover was installed over the shielding of the neutron facility for the University of Washington spermatogenesis experiment to improve its appearance. Operation of the neutron generator at approximately the rated output current was achieved.

WHOLE BODY COUNTING PROGRAM

New results for Fe^{55} are as follows:

<u>Subjects</u>	Total Body Burden - nCi (Number of Subjects)	
	<u>Jan. 1965</u>	<u>Nov. 1965</u>
Richland adult males	8 (4)	13 (4)
Anaktuvuk adult males	59 (6)	81 (6)
Tanana adult males	--	252 (6)
Tri-City cattle	440 (8)	228 (4)
Anaktuvuk caribou	728 (11)	953 (6)

Generally, the body burdens have increased; those of cattle did not, probably because their food now contains less fallout than previously. The data for Tanana Eskimos was collected when our previous studies showed high activity densities in some fish and suggested that such fish-eating people should have relatively high burdens. They have three times as much Fe^{55} as the caribou-eaters at Anaktuvuk. None of these data have been corrected for a 19% decrease due to radioactive decay.

A stepping motor was installed to drive the bed of the shadow shield counter. The stepping rate is controlled by the live-time of the multichannel analyzer used. This should minimize errors due to analyzer deadtime for strong sources. In a test with a source that gave 25% deadtime when directly under the scintillator, the use of the new drive resulted in a 14% higher total count; the improvement in a differential scan would have been even more significant.

The experiment with Biology Department to study short term retention of inhaled plutonium in dogs is continuing. Movement of the dogs during counting is turning out to limit the accuracy obtainable.

In the study of semi-absolute ways of calibrating whole body counters, a method was found for correctly allowing for differences in depth and in lateral position of sources within a homogeneous phantom.

RADIOELEMENTS IN ALASKAN ESKIMOS PROGRAM

A trip was made to Alaska to gather samples of blood, salmon, and caribou. Results of analysis of some of these samples for Fe^{55} are given in the first paragraph under "Whole Body Counting Program." They show that Fe^{55} levels are still increasing in Alaska and that fish-eating people contain much more than caribou-eating people.

Analysis of some caribou bone samples showed they contained more than enough Pb^{210} to support the Po^{210} present. Many of the above samples will be analyzed for Pb-Po^{210} .

INSTRUMENTATION PROGRAM

Further progress was made to reduce circuitry interference difficulties with the experimental animal physiological function and thyroid radio-nuclide uptake telemetry system. This effort has not as yet been completely successful.

For the lithium-drifted, germanium solid state detector studies, the previously used 0.001-inch-thick tantalum window was replaced by an aluminum window of the same thickness. This eliminated interference from the 56 keV X-rays from tantalum. Calculations demonstrated that the window thickness plus the detector dead layer thickness prevented observation of the low energy X-rays from Pu²³⁹. Due to the small amount of activity, tests to measure the spectrum of a U²³⁵ fission foil proved inconclusive. Attempts will be made to obtain a "thin window" detector to permit low energy X-ray measurements. In addition, work was outlined to reduce the present background interference.

Modifications incorporated in the experimental algae radionuclide uptake measuring instrumentation achieved a better balance between the two channels of background-only and background plus signal and improved operation of the most sensitive count-rate-meter. Calibration tests were successfully completed.

Development work, initiated on a solid state dose rate monitor using field effect transistors, will lead to replacement of the present fish dose-rate monitor used at the Biology Department fish facility. The new unit is being designed to provide improved sensitivity and measurements from 0.1 mR/hr to 100 mR/hr. Present scintillation detectors and the recorder will be used.

A review of the Biology Department counting room instrumentation system has been completed to aid in establishing an upgraded system. The effort, for the most part, is being directed toward acquisition of appropriate commercial equipment; however, some specific development of interconnecting circuitry will be required.

Experiments were conducted with several dielectric materials to determine the best approach for use in the experimental animal activity monitoring system. The requirements are quite stringent regarding both resilience and compressibility. Of the many materials examined to date, foam rubber demonstrates the greatest promise.

The neutron dose-rate measurement channel, including the detection head which incorporates five lithium-foil covered, surface barrier diode detectors,

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was nearly completed for the experimental mixed-field, neutron-gamma, dose-rate monitor. The fabricated ionization chamber, which will be used as the gamma detector, is being assembled and the gamma channel circuitry was partially designed. All rechargeable batteries will be used.

Development of instrumentation to measure the electrical charge and rate of fall of rain drops, being done in support of Atmospheric Physics studies, progressed with laboratory checkout of circuit refinements which improve stability of the charge-measuring portion of the instrument. Experiments were also conducted to better define the more difficult problem of measuring rain drop falling velocity.

A system of cascaded, RC time delay circuits was designed for use in specific dosimetry studies. A single control for each delay circuit permits adjustment of the delay period from 1 to 100 microseconds. Three circuits were breadboarded for testing of short-term stability. The measured stability was $\pm 0.2\%$ at the maximum delay time. In final form, approximately 100 of the circuits will be used in cascade as a simulator in microdosimetry studies.

INSTRUMENTATION EVALUATION AND ENVIRONMENTAL MONITORING SERVICES

Evaluation and minor modification effort was accelerated on the remaining few scintillation, transistorized, combined alpha-beta-gamma hand and shoe contamination monitors. The units delivered to the field to date have performed in a fully satisfactory manner as have the similar, but less complex, alpha-only hand contamination monitors. In addition, full training classes are being conducted to acquaint the plant maintenance technicians with the instruments.

Evaluation of a commercial, portable GM monitor, which might be obtained to replace the present GM monitors, was partly completed. General operation of the commercial unit shows promise but further tests will be necessary.

Engineering advice, rendered to radiation monitoring maintenance technicians, served to outline steps to convert the Model I 614 Monitors to the Model II configuration.

Fabrication of a solid state portable survey instrument employing an ionization chamber was 50% completed. Field effect transistors are used with a rechargeable battery. The unit may replace the present Hanford CP instruments which employ electrometer tube circuits.

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Section E

CHEMISTRY DEPARTMENT

TERRESTRIAL ECOLOGY - EARTH SCIENCES

Hydrology and Geology

All of the major hydrology digital computer programs have now been converted from the 7090 to the 1107 computer. The conversion effort has reduced the turn-around time for most of the programs from 1 - 2 days down to 1 - 2 hours thus increasing the problem-solving time efficiency. The Big Steady program has been proved to yield correct results in the saturated flow system both with and without the unequal grid spacing and the addition of inflow quantities at various nodes.

The general gravity survey of the Hanford area progressed. Gravity readings have indicated that the assumption upon which the success of the survey was predicated, a constant density contrast of about 1.0 between the basalt flows and the overlying alluvium, is essentially satisfied. A gravity survey of what are believed to be buried basalt boulders south of the Purex plant indicated no anomalous gravity highs which would be the case if the basalt were a massive basalt hill or ridge rather than boulders.

Geologic mapping just north of the Horse Heaven Hills between Kiona and Ice Harbor Dam showed that the upper Beverly Member interbed continues beyond Pasco and Kennewick about to the junction of the Snake and Columbia Rivers. The overlying Elephant Mountain basalt flow is increasingly thin and patchy toward Wallula Gap, particularly southwest of the Columbia River. It accordingly has been extensively scoured by the river. The Ringold Formation silts and clays, in turn overlying the Elephant Mountain basalt, maintain confinement of the water in the Beverly Member aquifer to the point where they terminate. Such a site is near the Snake River-Columbia River junction as well as to the northeast as earlier reported. If the possible site of emergence of confined waters is substantiated, the distance of travel of ground waters from the 200 Areas is nearly double the distance to which they are indicated to have travelled to date.

ATMOSPHERIC RADIOACTIVITY AND FALLOUT

Atmospheric Tracer Studies

Washout and rainout studies were continued on the removal of natural and artificial tracer materials from the air by rain. During the Fall rains four runs using $ZnCl_2$ and Br_2 were made where rain which had passed through the tracer cloud was collected; it is being analyzed by neutron activation.

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One release of HCl gas was made to determine if it could be followed by pH measurements; the test was marginal and will be tried at a higher tracer concentration.

Atmospheric Dispersion Studies

Another successful radioxenon release experiment was completed. Samples were obtained at distances up to 1 mile.

RADIOLOGICAL AND HEALTH CHEMISTRY

Radiation Chemistry

Computer manipulation of electron spin resonance (ESR) spectra became possible with the completion of a signal generator designed for use with the instrument and a paper tape data recorder. Using this arrangement and specially derived programs the data can be handled by the 7090 computer. With some additional programming, second moments can be calculated and simple averaging of successive signals can be performed for signal to noise enhancement.

RADIOISOTOPES AS PARTICLES AND VOLATILES

Particle Deposition and Re-entrainment

The deposition of 1 μ particles generated with the spinning disc generator was determined in the 0.62 in. ID tubes. These particles represent the near lower limit of sizes which can be generated with the present equipment. For Reynolds numbers from 800 to 30,000 the deposition within the 48-foot tubing was less than 1%. As the flow rate was increased to a Reynolds number of 42,000, the total deposition increased to 7%. In comparison with the deposition of 1 μ particles for the same flow range, the deposition of 10 μ particles increased from 1% to a maximum of 97% and then decreased to 4%. Maximum deposition as a function of flow rate has been measured for particle sizes greater than 6 μ .

Deposition in 0.62 in. ID tubes was measured for particle sizes from 1 to 10 μ and Reynolds numbers from 800 to 42,000. For this range of variables, the differential deposition as a function of length still needs additional study but the cumulative deposition or penetration through the 48-foot tubing is partially understood in terms of maximal deposition or onset of significant particle re-entrainment. The deposition data are interpreted to suggest that a dimensionless particle diameter y^+ , exists above which re-entrainment becomes important. If y^+ , based on flow rate and particle size, is > 0.07 , particle re-entrainment is important. The value of $y^+ = 0.07$ can also be considered as a dimensionless distance from the tube wall and is < 5 for the commonly accepted thickness of the laminar sublayer. A comparison of the values for y^+ of 0.07 and 5 leads to the tentative conclusion that the laminar sublayer is penetrated by fluctuating eddies to within a distance of $y^+ = 0.07$ of the tube wall.

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BIOLOGY DEPARTMENT

SOMATIC EFFECTS OF RADIATION

Radiation Effects General

Mechanism and Treatment of Gastrointestinal Radiation Effects

An in vitro process was used to determine the bioelectric characteristics and measure transport in the irradiated small intestine of the rat. In obtaining material for study, lead shields about 5 cm in length were used to protect segments of the intestine from exposure. These shielded segments were used to determine normal transport while adjacent, unshielded segments were used for irradiated gut measurements. This method has supported previous in vivo data, showing that sodium transport is severely depressed three days after 1500 r. In addition, there are alterations in the bioelectric potential and chloride transport that were not established previously. Further experiments are in progress to support these observations.

In vivo data from over 100 rats show that water and sodium absorption is decreased about 50 percent in both the large and small bowel three days after 1500 r. The movement of sodium into the small intestine is also decreased about 50 percent and water about 25 percent. In the large bowel, neither sodium or water excretion was affected by exposure. These results suggest that changes in absorption are paralleled by compensatory changes in water and electrolyte excretion that function to conserve essential fluid and sodium, and implicate the bile as the major source for losses in the intestinal radiation syndrome.

Response of Insects to Radiation

Proportions of observed lethality in "Chicago standard" (T. confusum) and "Brazil cI" (T. castaneum) strains of flour beetles agreed with values expected on the assumption that lethal effects due to x-radiation and insecticide, DDT, in combination simulated genetic dominants. Observed and expected lethal proportions differed markedly for "sooty" (T. castaneum).

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Expected lethal values were consistently greater than those observed, indicating that productivity was less adversely affected by the combination of radiation and DDT than was predicted on the basis of each stress given singly. Perhaps radiation-induced polygenes made their bearers better fitted to DDT-stressed environments. Productivity of "Chicago standard" in x-rayed populations was increased by moderate levels of DDT suggesting that in this species also polygenic overdominance induced by x-radiation may be of even greater fitness value to the bearer under conditions of stress than under normal conditions.

Toxicity of Radioelements

Factors Affecting the Toxicity and Metabolism of Radionuclides

A non-ionic detergent, Triton, causes an accumulation of triglycerides in the blood of rats two hours after injection, and may be used to test fat secretion by the liver. In normal rats there was a 10-15 fold increase in triglycerides after Triton injection, but in female rats pretreated for 24 hours with ^{237}Np there was no elevation of that fat fraction, indicating that ^{237}Np may cause fat deposition in the liver by interfering with the secretory process. Further study using male rats is in progress.

A study relating age, iodine content in the diet, and ^{131}I retention has been completed. The data are not complete but early indications are that there is an age-dependent change in the ratio of ^{131}I retention in rats on a low iodine diet as compared to those maintained on a normal diet. Retention/mg thyroid gland was inversely related to age.

The results of studies on a marketable sized lamb sacrificed in October showed stewing to be the best method of meat preparation to eliminate ^{137}Cs from retail meat cuts. Roasting eliminated about 40 percent of the ^{137}Cs burden in a "leg of lamb". Frying was the least successful method for elimination of ^{137}Cs .

Body burdens of the adult sheep on chronic ^{137}Cs feeding are increasing as the seasonal temperatures decrease.

Four modes of thermoluminescent dosimeter readout operation (integrated light output vs. glow peak height, with or without nitrogen purge) were evaluated. Integrated light output with nitrogen purge was selected for routine dose evaluation because of consistent low standard error of the mean. Evaluation of the effect of irradiation temperature on the thermoluminescence of LiF was inconclusive. Refined temperature control is now available, and this experiment will be repeated.

Factors Affecting the Toxicity and Metabolism of Radionuclides in Aquatic Organisms

Marine amphipods maintained in an environment with ^{65}Zn labeled sediments showed a very low concentration factor from the labeled sediments as compared with uptake from labeled sea water.

The sediments, collected at 160 meters, were either left untreated, sterilized, or oxidized with peroxide prior to labeling with ^{65}Zn . In all cases the results would indicate that uptake is dependent upon leaching of ^{65}Zn into the sea water rather than uptake directly from the sediments.

Inhalation Studies

Two dogs died about 40 months after inhaling $^{239}\text{PuO}_2$. Death was due to cardiac arrest during anesthesia for minor surgery in one dog and possibly due to a middle ear infection in the other. Since histopathology studies are not yet complete, the possible implication of the plutonium exposure has not been determined. The body burdens at death were 0.97 and 0.56 μCi . The lungs contained about 40 percent, bronchial and mediastinal lymph nodes 30 percent, liver 20 percent, and bone 4 percent of the body burden. The fractions found in the liver were the highest observed with the exception of one dog in which metastatic pulmonary tumors were observed in the bronchial lymph nodes.

In two dogs sacrificed 7 months after inhalation of $^{106}\text{RuO}_2$, 97 and 98 percent of the total body burden was in the lungs and the remainder was predominantly in the bronchial and mediastinal lymph nodes. These results are consistent with our earlier work in mice.

The lung collagen content of rats exposed to 800 r thoracic x-irradiation increased 100 percent above the control values two days after exposure, remained high for two weeks and then dropped to control levels. Lung elastin of the irradiated rats increased 50 percent above the control values after eight days. Although, the ratio of lung lipid phosphorus to total lung lipids remained constant, the total lung lipid content of the irradiated rats increased 60 percent above the control values after 8 days. If these results are repeated in a subsequent experiment which has been started, attention will be directed towards determining their significance.

Additional data were obtained showing increased levels of taurine in urine of dogs having lymphoma as a result of plutonium dioxide inhalation.

COMBATING DETRIMENTAL EFFECTS OF RADIATION

Therapeutic Agents

Removal of Internal Emitters

Tests of EDTA, DTPA, TTA, and Triton for their effectiveness in removing $^{106}\text{RuO}_2$ from rat lungs were negative.

MOLECULAR AND CELLULAR LEVEL STUDIES

Fundamentals of Radiation Damage

Cellular Regulatory Mechanism

Methionine, ethionine, tyrosine, cysteine and phenylalanine were identified as inhibitors of tryptophan transport in *Neurospora*. These amino acids

[REDACTED]

showed inhibitions greater than 70 percent at molar ratios of 40 to 1 (inhibitor/tryptophan). Histidine and serine inhibited uptake by 40 percent, alanine by about 20 percent. Other compounds such as 3-methylindole, indole, and 3-indolepyruvate were without effect. We infer from these results that families of amino acids are transported by the same reactive site.

We reported last month the conditions for optimal labeling of ribonucleoprotein (RNP) particles in *E. coli* treated with the antibiotic, chloramphenicol (CAP). When cells containing labeled RNP particles are removed from CAP, recovery occurs and growth resumes. The radioactively labeled particles undergo a gradual increase in sedimentation velocity which has been interpreted by others as a conversion of these particles to 70 S ribosomes. We have studied the factors affecting the rate of this conversion. The addition of 0.5% casamino acids to a minimal recovery medium doubles the rate of conversion of the particles to 70 S ribosomes. These results suggest a model for regulation of RNA synthesis in which RNP particles play a key role as feedback inhibitors of RNA synthesis from precursors.

ENVIRONMENTAL RADIATION STUDIES

Terrestrial Ecology

Analyses of moss (mostly *Rhytidiopsis robusta*) collected in September from the forest floor of hemlock and douglas fir stands near Packwood, Washington, showed values for ^{137}Cs ranging from 57-68 pCi/gram dry weight. This was 9-10 times greater than the values obtained for seed plants harvested from the same places at the same time. These values for forest moss are generally higher than those obtained for lichens collected in Alaskan tundra. Direct transfer of ^{137}Cs from moss to vertebrate herbivores appears unlikely because the moss is apparently not palatable to these kinds of animals.

Plant Uptake of Radionuclides

The effect of disodium EDTA on the movement of ^{60}Co in acidified solution in cinebar soil was studied. Both the one dimensional ion migration front movement and the ion uptake by soil follow the previously proposed equation, $Z = AT^B$, where Z is either the quantity of ion carried away by the moving water or the movement of the ion migration front from the source in time T ; both A and B are parameters. There is no obviously significant difference in the amount of ^{60}Co uptake by soil from the disodium EDTA solution or the control. However, contrary to expectation, the ^{60}Co -EDTA migration front moved more slowly than the control.

Dynamic Balance of Radionuclides in the Columbia River

Waterfowl utilizing the Hanford section of the Columbia River as a wintering ground increased from about 42,000 to about 165,000 birds during the first two weeks of November. This is about twice as many birds as were present during this time of the past few years, and is indicative of an early influx noted at other wintering sites in this region. The population now centers in the Hanford-Ringold area, having shifted from the 300 Area-Ringold area in response to hunting pressure in the area opened to public access.

Discussions were held with Mr. Douglas Gilson, Bureau of Sport Fisheries and Wildlife, concerning pre- and post-impoundment studies of the proposed Ben Franklin Dam reservoir. We are uniquely prepared to conduct large-scale studies of the effects of the proposed dam on wildlife. Liaison will be maintained and further discussions are planned.

Tests on periphyton, carried out with artificial substrates in the Columbia River, indicate a stabilization of chlorophyll content within 20-40 days while the dry weight is still rising in an essentially linear fashion. No data are yet available to evaluate the contribution of suspended solids to the measured dry mass.

Levels of ^{65}Zn in Columbia River fish increased slightly over last month at Priest Rapids and Hanford. Whitefish were more numerous in the gill net catches at Priest Rapids and most of these fish showed prior recent exposure to reactor effluent.

Three more surveys were made of the locally spawning chinook salmon. Peak of spawning occurred during the first week in November. The total number of salmon redds between 300 Area and Priest Rapids was 1778. This compares with 1443 observed in 1964, and exceeds the previous record count in the year 1958 by approximately 300 nests. About 37 percent of the total were observed near Midway, the nearest spawning area below Priest Rapids Dam.

Temperature Effects on Metabolism of Aquatic Organisms

In vitro studies of tissues from temperature-acclimated fish have shown that the turnover of acetate varies as shown below (numbers are $\text{pCi } ^{14}\text{CO}_2/\text{mg}_\text{N}/\text{hr}$):

	<u>Red Muscle</u>	<u>White Muscle</u>	<u>Liver</u>	<u>Cardiac Muscle</u>
5° acclimation	54	25	18	230
20° acclimation	42	12	18	290

These results indicate that muscle tissues have an enhanced ability to turn over acetate at lower temperatures; liver does not. This could be due to: (1) a quantitative increase in the oxidative enzymes at low temperatures, (2) a qualitative change in metabolic pathways with less incorporation of acetate into structural units at lower temperatures, or (3) a change in the Q_{10} for the metabolic system. Incorporation studies and CO_2 evolution under reciprocal conditions, i.e., the 5° animals at 20° and the 20° animals at 5°; will be undertaken to answer these questions.

SNPO AND SNAP BIOLOGICAL STUDIES

Two female pigs were each given 1 Ci $^{238}\text{PuO}_2$ plasma-jet fired microspheres orally. At necropsy, 14 days postfeeding, one particle was found at the ileo-cecal junction. This particle was in an encapsulated and inflamed area. The inflammation was about 2 cm wide and 4-6 cm long. This particle was first detected by wound counter. No particles were detected in the digestive tract of the second pig.

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The particle passage time differed in both of these pigs from that measured in the first animal given the plasma-jet fired particles. The first gamma-detectable count in the feces was on day 3, with a rapid increase to day 8 or 9, then a rapid falling off. Whole body counts and tissue radioanalyses are not completed.

SERVICE ASSESSMENT STUDIES

The release of columnaris organisms from the spawning channel and fish ladder of Rocky Reach Dam decreased by a factor of ten during the month of November. Lower water temperature, a shift in the migration pattern of various resident river fish, and smaller numbers of salmon passing through the ladder, are probable causes for the decrease in discharge of columnaris organisms.

Approximately 4,000 newly spawned chinook salmon eggs of Green River stock were obtained from the University of Washington and placed in reactor effluent concentrations of 0, 2, 4 and 6 percent at the 1706-KE Experimental Hatchery. Adult chinook salmon from the local Priest Rapids stock continue to ripen at 146-FR. When a spawn is obtained from them, the eggs will be placed in similar concentrations of reactor effluent at 1706-KE. Since most of the earlier monitoring studies were done with Green River stock, this study will provide data that will enable a more meaningful interpretation of earlier results as they apply to local races of salmon.

PERSONNEL

L. C. Brazle, AEC Health Physics Intern Completed his 6 week assignment on 11-5-65.

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APPLIED MATHEMATICS DEPARTMENTSomatic Effects of RadiationRadiation Effects - GeneralResponse of Insects to Radiation

A computer program was written to facilitate the study of population cycles exhibited by flour beetles irradiated with Co^{60} .

Toxicity of RadioelementsToxicity and Metabolism of RadionuclidesFactors Affecting

A study was completed relative to rodent patterns of retention and excretion of Sr^{85} and Ca^{45} as functions of time.

A second data analysis study was completed on the effects of Zn^{65} and P^{32} on the body burdens of fish raised in reactor effluents.

Aquatic Organisms

An analysis was carried out on data from a biochemical experiment involving labeled CO_2 in fish.

Radiostrontium in Miniature Swine

The computer program for scheduling the periodic collection of growth data was modified and expanded.

Radioiodine in Sheep and Swine

Additional analyses were carried out on data obtained via whole body counting of sheep injected with selenium.

Environmental Radiation StudiesTerrestrial and Fresh Water EcologyDynamic Balance of Radionuclides in Columbia River

The Canonical Form method has been satisfactorily adapted to the EDPM program GENORO. A test case comparison of this new fitting method with that used previously is currently in progress. New input-output features are being contemplated for this new program.

Terrestrial Ecology

Statistical models were formulated for the relationship between conductivity and osmotic pressure in sap for certain plants.

Models for the Prediction of the Transport of Radionuclides

Work continued on defining compartments to be included in a model of the transport through a river ecosystem to man.

Radiological and Health Physics and InstrumentationRadiological and Health PhysicsRadiation Dosimetry

A statistical analysis was carried out on audit data from the U. S. Testing Company, Richland.

Work continued on specifying an optimal experiment procedure for a quantitative study of the equipment and techniques currently used in calibrating the gamma dosage recorded on the personnel film badge dosimeters.

Hanford Occupational Health Foundation

An obesity study based on Hanford personnel medical data was completed.

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RADIATION PROTECTION DEPARTMENTDosimetry Studies

The third and final phase on an inter-laboratory calibration study under the auspices of Dr. S. Allan Lough, Division of Biology and Medicine, U.S. AEC, was successfully completed during November. The purpose of this program is to compare calibrations for cesium-137 between several of the major whole body counter laboratories in the United States. The third phase of the program consisted of measuring two visiting human standards in the Battelle-Northwest Whole Body Counter, one of whom contains a "known" elevated cesium-137 burden, and the other who contains the "normal" quantity of cesium-137. Results from all three phases of this program will be reviewed and compared by an ad hoc committee, who will make recommendations to each of the participating laboratories on how to improve their cesium-137 whole body counting techniques.

Battelle's portion of the research program, "Correlation of Lifetime Health Experience of AEC and AEC Contractor Employees with Occupational Exposure", is being reviewed to determine the cost of providing data for a continuing program after the initial phase is completed. It is anticipated that this will become a regular program at Hanford. This review was initiated as the result of a meeting with personnel from Hanford Occupational Health Foundation, Inc.

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F. DIVISION OF ISOTOPE DEVELOPMENT: 08 PROGRAM

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Section F

SUMMARY OF ACTIVITIES

REACTOR AND MATERIALS TECHNOLOGY DEPARTMENT

The pneumatic impaction of five isotope capsules in a single cycle was successfully completed. TIG welding development on isotope capsule end closures has produced penetrations exceeding the amount specified. Design of an isotope capsule impact tester was started.

CHEMISTRY DEPARTMENT

Analysis of the nuclear reactions involved in Pm^{146} formation has suggested that the $\text{Pm}^{146}/\text{Pm}^{147}$ ratio may be lowered by a factor of 10 for 35 days' irradiation in a Hanford reactor test facility. An experiment to test this conclusion has been planned.

Oxysulfides and sulfides of promethium have been prepared as possible heat source materials.

Compatibility data on Haynes-25 samarium oxide shows an average attack of 0.0003 in. after 14 days at 1100 °C. Iridium is being evaluated as a liner material and the concept of using an oxygen getter (to lower oxygen activity) is being tested.

The promethium purification run showed that the resin could be washed very free of radioactivity; however, little promethium was recovered in the process. Resin damage will be measured.

A series of ten simulated radioisotopic heat sources was produced with overall diameter variations of 5 mils and a height variation of 20 mils. Core density varied 0.4% of the mean value.

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ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTISOTOPIC ENCAPSULATION

Initial TIG welding development on the end closure of the Ames capsule produced weld penetration equal to 75% of the joint thickness. This depth of penetration exceeds the amount specified.

Impact Tester

An investigation of the requirements and components necessary to construct an impact tester was started. The machine must be capable of simulating impact of capsules on reentry from space.

Pneumatic Impaction Techniques

The continued development of isotope encapsulation using pneumatic-impaction techniques was continued this month. A total of 24 capsules was impacted. The first multiple impaction of five capsules at a time was successfully completed with excellent results. The length to diameter ratio was 4:1 before impaction and 1:2.75 after impaction. Excellent right cylinder characteristics were maintained without clad folding.

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CHEMISTRY DEPARTMENT

PROMETHIUM ISOTOPIC FUELS

Promethium Purification and Fabrication

The A-Cell ion-exchange columns were exhaustively stripped with the objectives of improving the Pm mass balance over that of the October purification run and decontaminating the resin so that samples can be removed to the laboratory for measurement of the extent of radiation damage. Only small amounts of Pm were found (less than 1% of that in the feed). The recovered samples of resin are radioactively cold. Laboratory column experiments were started to study DTPA as an elutriant in attempt to decrease further the time cycle for promethium purification.

A start was made in the decontamination of C-Cell so that it may be used on an interim basis for converting the purified Pm to oxide. Plans were made to prepare samples at an early date for measurements of physical properties, to load several small compatibility capsules with Pm_2O_3 (vice Sm_2O_3), to prepare 10 kilocuries for shipment to Oak Ridge (as oxide), and 3 kilocuries for shipment to Arco.

Pm¹⁴⁶ Burnout Calculation

The possibility of considerably reducing the concentration of undesirable Pm¹⁴⁶ in Pm¹⁴⁷ by irradiation in a well-thermalized neutron flux was further considered. This is possible because the measured cross section of Pm¹⁴⁶ (8400 barns) is so much larger than that of Pm¹⁴⁷ (235 barns). Thus, irradiation at a flux of 10^{14} neutrons/cm²(sec) of purely thermal neutrons should reduce the Pm¹⁴⁶/Pm¹⁴⁷ ratio by a factor of 10 for every 35 days of irradiation, and a 70 day irradiation would be expected to reduce the Pm¹⁴⁶ content of Hanford Pm by a factor of 100 to 0.0025 ppm. In a real reactor, however, a limiting or equilibrium Pm¹⁴⁶ content will be reached due to the fact that the flux is not purely thermal. Computer calculations were made for several actual reactors. These show that the well thermalized Hanford reactors are probably the best available for the purpose. With special reactor loadings, the fast-to-thermal flux ratio would be about 0.03, yielding a limiting Pm¹⁴⁶ value between about 0.01 and 0.02. It is unlikely that the Pm¹⁴⁶/Pm¹⁴⁷ ratio can be reduced to less than 0.01 ppm. There is sufficient uncertainty in these calculations, due to lack of information on the fine structure of the Pm¹⁴⁶ cross section, to make an experiment highly desirable. If the shape of the neutron absorption spectra of Pm¹⁴⁶ and Pm¹⁴⁷ differ sufficiently, it may be possible to improve the efficiency of the Pm¹⁴⁶ removal process and to minimize Pm¹⁴⁷ burnout by proper design of the geometry of the target element. An experiment to test actual burnout performance was planned, using "green" Pm.

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Promethium Heat-Source Compounds and Compatibility Measurements

Three new potential Pm heat source materials were prepared using Sm as a stand-in and evaluation was started to determine if their compatibility with cladding materials at high temperatures is better than that of the oxide. These include Sm_2S_3 , $\text{Sm}_2\text{O}_2\text{S}$, and $3\text{Sm}_2\text{O}_3\text{WO}_3$. In addition, Sm_2O_3 was treated with H_2 prior to encapsulating in the hope that this treatment will affect the defect structure of Sm_2O_3 in a favorable manner - by the formation of reducing holes (Sm^{++}) in the crystal lattice rather than oxidizing holes (Sm^{+4}).

Compatibility tests at 1100 C of a variety of cladding materials were continued. Haynes-25 samples pneumatically impacted into contact with the oxide showed an average attack of about 0.0003 in. in 14 days at 1100 C; there was some pitting, the maximum depth of which appeared to be about 0.001 inch. 304-L stainless steel showed about the same depth of attack. Such rates of attack are probably tolerable, provided that they do not increase with more prolonged heating. In other experiments, Ir was under evaluation as a liner material and the use of powdered Ta as an oxygen getter (mixed with the Sm_2O_3) was under study. Capsules of a variety of materials were fabricated and are being loaded and compacted for heating for long periods of time.

Isotope Packaging Program

A series of ten simulated radioisotopic heat sources was produced on the 1220 F Dynapak Machine for the purpose of testing the reproducibility of a particular product form. Can design, powder loading, heat cycle and impaction parameters were held constant. The height and diameter of each can were measured, including variation on individual cans. Wall thickness variations will be subsequently measured. The resulting data will be analyzed statistically to determine the reproducibility of the operations. The overall diameter variation was 5 mils and the mean diameter variation was 2 mils. The spread in the heights of the individual cans was 20 mils. Final core density varied from 93 - 93.7% theoretical density. The core material was commercial grade SrTiO_3 which had been sintered at 1400 C and ground.

During this series of runs the actual energy input into the capsule during the impaction step was obtained from the double integration of the punch strain curves through the use of a computer program. To date only the first peak of several that occur during the hit can be integrated. Analysis of this first peak indicated that 33,000 ft-lb of energy were absorbed by the punch of which 26,000 ft-lb were transferred to the piece. The manufacturer's theoretical energy input data indicates that under these conditions some 35,000 ft-lb of energy should be available from the machine.

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In preparation for a study of impact resistance of heat sources produced by high energy impactation, a fillet-type weld joint was being developed as final closure for the containers; the best of three geometries under study will be used. In preparation for the same studies, a test was run to determine whether He introduced at one end of a capsule would pass through the compacted core so that He leak detection methods could be used to determine the integrity of a container after it had undergone impact testing. The test showed that He passes through SrTiO_3 impacted to 94% theoretical density.

Several runs were made with the new thermoconductivity apparatus using SrTiO_3 powder. Reproducibility was good. The equipment will allow thermoconductivities to be obtained at high temperatures and under relatively high vacuum.

A 90 cm long by 1 in. diameter heat pipe, consisting of a boiler and a condenser with a wick to return condensate to the boiler, was tested with water over the pressure range 2 - 20 psia. With a horizontal wick a maximum heat flux of 150 watts was attained at atmospheric pressure. Although X-ray measurement showed that the wick was not against the pipewall in a portion of the boiler, the heat flux was apparently limited by the ability of the 200 mesh stainless steel screen to move liquid toward the boiler, since the maximum heat flux agreed with approximate theoretical calculations. There was no detectable temperature drop along the insulated section of the pipe between the boiler and the condenser when it was operated as a closed system or with an argon reservoir.

Brief consideration was given to the possibility of irradiation in a recirculating solution loop. This did not appear attractive because of a 6-fold increase in reactor target space required.

A high degree of uncertainty in the cost estimates is indicated by the spread between the low and high estimates. Nevertheless, comparing the estimated costs with the current price for Pm^{147} of \$560/watt, it was concluded that the prospects are at present marginal for economic production of Pm^{147} by Nd^{146} irradiation. Further consideration of this process is questionable unless a large volume of reactor target space can be provided with an effective flux of at least 1×10^{14} n/cm²/sec.

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G. USAEC-AECL COOPERATIVE PROGRAM

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AIM

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SUMMARY OF ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENT

A tubular specimen made up of Zr-2 butt-welded to Zr-Nb alloy and tested at room temperature showed brittle behavior in the Zr-2 but not in the Zr-Nb alloy.

Property measurements made ultrasonically on Zr-Nb pressure tubing show orthogonal anisotropy for tubing in the as-received condition.

The hardness flow surface motion as a function of strain has been determined for Zr-Nb pressure tubing and shows an anisotropic work-hardening effect.

Analytical studies and test section design were continued in the study to determine the values of cross-channel mixing during two-phase flow in rod bundle fuel elements. A welding method was developed which looks promising for fabricating the electrically heated test sections.

A four-node nuclear kinetic model was programmed for the analog computer and runs made to demonstrate simplifications in the model. Preliminary simplification was made of the hydrodynamic equations to fit on available computer equipment.

ACTIVITIESREACTOR AND MATERIALS TECHNOLOGY DEPARTMENTIN-REACTOR MEASUREMENTS OF CREEP IN Zr-2.5 Nb ALLOY

The in-reactor creep tests on heat treated Zr-2.5 wt.% Nb at 350°C and 36,500 psi, and 300°C and 45,000 psi are continuing. An in-reactor creep rate of 3×10^{-7} /hr was measured for the 350°C - 36,500 psi test (Capsule IV-7) after 1700 hours of test time. The ex-reactor creep test at the same conditions exhibited a rate of 9×10^{-7} /hr after 1000 hours of test time. The stress on Capsule IV-7 is scheduled to be raised to 45,000 psi. The in-reactor creep test at 300°C - 45,000 psi has accumulated 3400 hours of test time. A rate of 3×10^{-7} /hr was measured and appears to be the minimum. Additional time will be required to determine if this is the case. The Capsule IV-14 test was previously run for 1600 hours at 300°C - 36,500 psi creep stress. No significant rate was observed for these conditions, but the ex-reactor creep test exhibited a creep rate of approximately 1×10^{-7} /hr.

A third capsule will be charged in January 1966 and will be used to run a creep test at 400°C - 36,500 psi stress. An ex-reactor creep test at these conditions exhibited a creep rate of 8×10^{-6} /hr after 500 hours of test time. This may not be the minimum rate.

Zr-Nb PRESSURE TUBING

The general objective of this program is to evaluate Zr-2.5 Nb as a pressure tube material with reference to the tests and reactor experience that have been obtained on Zircaloy-2 pressure tubes. Continuation of the crack propagation tests on tubes in the cold worked and hydrided, to 250-300 ppm, conditions has produced the results tabulated below.

Crack Propagation Data for Zr-2.5 wt.% Nb Alloy

<u>Specimen Identifi- cation</u>	<u>Specimen Condition</u>	<u>Test Temp. °C</u>	<u>Slot Length (in.)</u>	<u>Hoop Stress at Failure (psi)</u>	<u>Crack Length at Failure</u>
30-9	30% CW & Hydrided	25	1/2	76,500	Brittle crack propagated full length of test piece
30-7	30% CW & Hydrided	25	1-1/2	31,400	"
30-10	30% CW & Hydrided	25	2-1/2	28,000	"
30-9	30% CW & Hydrided	150	1-1/2	27,300	"
60-9	60% CW & Hydrided	25	2-1/2	8,300	"
30-17	30% CW & Non-hydrided	25	4	33,300	"
60-17	60% CW & Non-hydrided	25	4	26,600	No crack propagation

Photomicrographs of samples from the tubes reveal uniform hydriding except for a thin hydride case on the exterior. This case is machined off before testing. Orientation of the hydrides is similar to that observed in Zircaloy-2. Near the inside surface the poles of the plates are radial and in the interior of the tube wall the plate orientation is more random when viewed on a transverse section.

Also tested was a specimen made up of equal lengths of Zr-2 and Zr-2.5 wt.% Nb alloy tubing which AECL had joined by butt welding. The Zr-2 piece was 3.25" ID x 0.160" wall and the Zr-Nb piece was 3.25" ID x 0.095" wall. A 1/8 inch deep x 1 inch long slot located about two

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inches from the welded joint was milled in the outside surface of the Zr-2 section. Pressurizing the test piece initiated brittle failure in the Zr-2 section at a hoop stress of 52,000 psi; the brittle crack propagated only a short distance through the weld and was arrested in the Zr-Nb alloy section.

A two-day room temperature test run was made in the creep loop. All instrumentation appeared to operate satisfactorily. A high temperature run will get underway within about a week.

The as-received material properties of 30% cold worked Zr-2.5 wt.% Nb pressure tubing were measured ultrasonically, using a bi-element transducer. Measurements in each of the principal directions of the shear and longitudinal wave velocities were made. The calculated Poisson ratio values vary from 0.365 in the axial direction to 0.350 in the radial direction. The calculations were made assuming the materials to be isotropic. Work is being done to develop the correct anisotropic equations to generate the properties of an orthogonal tube from six velocity measurements.

The strain hardening as determined by the Knoop hardness number flow surface technique has been determined for solution annealed Zr-2.5 wt.% Nb pressure tubing pulled in the axial direction. The work hardening appeared as if the material had been strained by alternate compression in the radial, then in the tangential directions rather than uniaxial tension in the longitudinal direction. At a maximum strain, 70% reduction of area, the flow surface maximum hardness was 270 KHN with a minimum of 227 KHN and an average of 248 KHN. The 0% reduction of area flow surface had a maximum hardness of 248 KHN with a minimum of 227 KHN and an average of 236 KHN.

CROSS-FLOW MIXING BETWEEN PARALLEL FLOW CHANNELS DURING BOILING

Analytical work and test section design were continued in the study to determine the values of cross-channel mixing during two-phase flow in rod bundle fuel elements. It is the purpose of this study to obtain a method of predicting local flow and enthalpy conditions in each sub-channel of rod bundle fuel elements. This information will provide a more accurate means to calculate coolant conditions at boiling burnout.

The analytical study has been directed toward numerical methods to solve the equations which describe the interchannel mixing between two adjacent flow channels.

The numerical approach is necessary since the equations are very complex and can only be solved in closed form for a few special cases. A digital computer program is being written to do the numerical work. The calculations will include both the effects of fluctuating cross flow and flow diversion which is caused by pressure drop differences between adjacent boiling channels. An experimental program which is included in this study will provide key information to support the mixing model assumptions.

Several possible methods of fabricating welded test sections for the experiments have been investigated. One method of fabrication which gives an excellent simulation of two adjacent rod bundle flow channels shows promise. This is accomplished by positioning the work on a lathe bed and using the carriage to hold and move the welding torch. Sample test section pieces have been fabricated and welded with promising results. It is presently anticipated that this welding technique will be used as a basis for completing final test section design.

PARALLEL CHANNEL REACTOR BOILING STABILITY STUDY

The "desired" four-node model of the reactor neutronics contains calculations of diagonal leakage, inverse node length and geometric buckling which are functions of boiling length. Analog computer studies this month have shown that the relative effects on the power level with changes in boiling length due to the diagonal leakage and the inverse node length calculations are small when compared with the effects caused by the geometric buckling. Consequently, a large number of multipliers can be made available for use in the other portions of the simulation without much of a sacrifice in the accuracy of the power level calculations.

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Analog diagrams have been made of the entire "desired" model including the neutronics, flow, temperature, and steam drum calculations. The "desired" model required 18 more multipliers than the analog laboratory has. The initial simplification of the calculations in the two phase regions reduced the non-linear equipment requirements by 16. Further simplification is required. Although the simplifications reduce the rigor of the analysis, they are not believed to cause significant errors. Further study of these simplifications is planned.

A transient fuel temperature model has been developed using a four-node representation for the fuel and one node for the cladding. A multi-node representation is required for the oxide fuel simulation since the heat flux from the fuel is mainly dependent upon temperature gradients within the fuel and is not very dependent upon changes in coolant temperature. An equivalent electrical circuit is being used for this fuel temperature simulation since it can be built external to the computer and will not use computer equipment.

A steam drum pressure transient model has also been developed for the study. For analysis purposes, the drum is assumed to consist of three regions: (1) a mixing region for the feedwater and reactor two-phase flow, (2) a steam separator, and (3) a central drum region. Equations describing the first two regions are based on steady state energy and mass balances. The third region uses methods of transient thermodynamics to describe the pressure transient.

Pressure changes within the drum are very complex and depend upon how much of the vapor condenses during a pressure increase and upon how much water flashes during a pressure decrease. To rigorously include these effects is extremely difficult; therefore, a simple approximation is used. During a pressure increase, it is assumed that some constant fraction of the total drum water is heated to saturation by condensing vapor. During a pressure decrease, this same fraction is allowed to flash; however, if the pressure decreases sufficiently to reach the saturation pressure of subcooled water at the bottom of the drum, all the drum water is allowed to flash. This method of calculation would provide a reasonable method to determine if the condensation of vapor in the drum is a significant parameter affecting the stability of the system.

Results of work to date were discussed in a meeting with AECL personnel during the month.

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