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AEC
RESEARCH
and
DEVELOPMENT
REPORT

**PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT
MARCH 1966**

on
AEC DIVISION
of
REACTOR DEVELOPMENT AND TECHNOLOGY PROGRAMS

APRIL, 1966

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

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Nuclear Technology
(~~Special Distribution~~)

PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT
MARCH 1966

on
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of
Reactor Development and Technology

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April, 1966

NOTICE: PRELIMINARY REPORT

This report contains information of a preliminary nature prepared in the course of work under Atomic Energy Commission Contract AT(45-1)-1830. This information is subject to correction or modification upon the collection and evaluation of additional data.

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SUMMARY OF ACTIVITIES

CIVILIAN POWER REACTORS

SPECIFIC FUEL CYCLE ANALYSIS

The fuel cycle costs were calculated for the Allis-Chalmers 1000 MW_e reference design based on burnup data from the ZODIAC code. A comparison between U-235 and plutonium as core enrichment showed plutonium to be worth up to \$20/g fissile.

Fuel utilization studies are now being carried out using the Allis-Chalmers burnup data with a U-235 enriched core as a fast converter.

CONCEPTUAL REACTOR DESIGN STUDIES

In the study of siting effects on capital cost, variations in local labor rates are estimated to affect total direct construction costs by 3% at most. Further investigation has shown that plant construction costs will probably not be significantly affected by seismic considerations. Coastal property prices in southern California were estimated and additional problem areas regarding appropriate isolation distances in densely populated or recreation regions were identified.

In the study comparing U²³⁵ and plutonium fueling of fast breeders, burnup calculations for a carbide-fuel reactor were begun. Two plutonium-fueled batch cycle cases were completed and a U²³⁵ fueled case was started.

USAEC-AECL COOPERATIVE PROGRAM

Analytical calculations and test section fabrication were continued in the study to determine the values of cross-channel mixing during two-phase flow in rod bundle fuel elements. Fabrication of the parts for the test section, which represents the square-triangular type of flow channel, is nearly complete and ready for welding.

The entire set of nuclear and thermohydraulic equations describing the BLW-250 have been programmed on the analog computer. The system is currently being debugged in preparation for a test run to compare with an earlier calculation made with the digital computer code, MIDAS.

Additional crack propagation tests have been run. The data obtained continue to show the beneficial effects of low hydrogen impurity concentrations. The creep-rupture test loop has been started. The first test run has operated for about 250 hours without difficulty.

APPLIED AND REACTOR PHYSICS

PLUTONIUM CRITICALITY STUDIES

Installation of a critical assembly vessel for studying the criticality of plutonium nitrate solutions in slab geometry was completed, and a checkout of the equipment is underway preparatory to beginning the experiments. A unique feature of the slab tank is that its thickness may be adjusted (due to the use of stainless steel bellows in its construction) over the range from three to nine inches.

Further criticality calculations have been made on plutonium compounds for which experimental criticality data are lacking. The calculations apply to homogeneous mixtures of common plutonium compounds in water for H/Pu atomic ratios ~ 20 .

HIGH TEMPERATURE REACTOR PHYSICS STUDIES

A contract has been awarded for the fabrication of the enriched uranium driver fuel for the High Temperature Lattice Test Reactor.

High temperature environmental tests on rare earth oxides dispersed in nickel show that these materials are acceptable for use as poisons in the safety rods.

Acceptance tests on the Programmed Measurement and Control System (PMACS) were conducted in the HTLTR control room. Some specifications were not met and some problem areas were found. These details are being resolved.

Programming of the control loops within PMACS is proceeding, with the assistance of the HTLTR operations staff. Characteristics of the control and sensing elements in the control loops external to PMACS are being obtained.

The Final Safeguards Analysis and a rough draft of the Operating Limits were completed for transmittal to the Richland Operations Office of the AEC.

Discussions on the experimental program for the HTLTR were held with the HTGR Technical Advisory Committee of the AEC and with personnel from General Atomics.

The determination of a 3-D power map for the MTR experiment continues to be an important design problem. New approaches to splicing 2-D power distributions are being considered. Improvements are also being made in the MTR core thermal hydraulics computer code.

Preparations for the experiments in the CAF in support of the MTR-Phoenix experiment are complete.

Fabrication of fuel for the PRCF critical experiments is progressing well. The second 2-kg batch of PuO_2 was reduced to Al - 20 wt% Pu alloy which was cast into billets, extruded to sheetbars and is now in the process of being rolled to core plates.

REACTOR FUELS AND MATERIALS

FAST FUELS OXIDES AND NITRIDES

Installation of the high-pressure, high-temperature filament furnace in a plutonium-contaminated glove box was completed. This furnace will be used to determine melting points of various UN - PuN mixtures.

Successful modifications to improve the accuracy of the high-temperature thermal expansion dilatometer were completed.

The laser-pulse thermal diffusivity apparatus was installed in a hood. Calibration is in progress. A modification of the bias voltage supply for the indium antimonide infrared detectors increased the usefulness of this detection method to above 1500°C.

Samples as low in stoichiometry as $\text{PuO}_{1.72}$ did not show any significant changes on heating to 700°C in the reflection electron microscope. Observation of a phase change was expected.

BASIC SWELLING STUDIES

Three capsules continue to operate successfully. One is controlled at 1000 psi and 450°C; the other two are at low pressure (approximately 30 psi) and are controlled at 525°C and 700°C (beta phase), respectively.

Density measurements made on uranium specimens recovered from capsules controlled at 500 psi and at 550°C and 625°C, respectively, support the belief that mechanical processes are responsible for swelling in alpha uranium. Annealing studies of irradiated thorium-uranium alloys show the formation of small pores in the alloy after a 140 hour anneal at 825°C.

NONDESTRUCTIVE TESTING

Laboratory comparisons with the multiparameter eddy current tester indicated a favorable ratio between response due to simulated flaws and deviations due to detector nonlinearity. Basic studies on the self focusing properties of intense ultrasonic beams have been initiated by investigation of analogous effects from intense laser light.

NUCLEAR CERAMICS

Irradiation of UO_{2+x} fueled capsules designed to study oxygen relocation and bulk composition effects on fuel-temperature distribution was completed. Five capsules contained $\text{UO}_{1.90}$, $\text{UO}_{1.99}$, $\text{UO}_{2.02}$, $\text{UO}_{2.05}$, and $\text{UO}_{2.15}$ were irradiated at 30 kw/ft for 16 hrs. Each capsule also contained a $\text{UO}_{2.00}$ internal standard. Mean thermal conductivities of $\text{UO}_{1.90}$ and $\text{UO}_{2.02}$, deduced from metallographic examinations of the irradiated capsules, are significantly different; k for $\text{UO}_{1.90}$ is about 25% greater than for $\text{UO}_{2.00}$, and k for $\text{UO}_{2.02}$ is about 15% less than for $\text{UO}_{2.00}$.

Preliminary data for polycrystalline UO_2 indicate that the electrical conductivity is a function of the frequency of the alternating current. The conductivity increases with increasing frequency at temperatures below 750°K , and decreases with increasing frequency at temperatures above 1650°K . The conductivity is independent of frequency at intermediate temperatures.

A recently developed mechanical thinning technique was successfully applied to a single crystal of UO_2 . Further work is required to eliminate sample preparation artifacts.

Densities of all materials used in this study (UO_2 powder, graphite crucibles, and Al_2O_3 powder matrices) were increased during pneumatic impaction at 1140°C and 285,000 psi. Good bonding was observed between UO_2 powders and UO_2 single crystals; no bonding occurred between UO_2 single crystals.

Experiments to investigate the growth of PuO_2 from a solution of PuO_2 in $\text{Na}_2\text{O-SiO}_2$ glass were initiated.

UO_2 crystal specimens were shipped to three off-site laboratories in conjunction with the materials and information exchange program.

Two PuO_2 simulation pellets were fabricated to establish pneumatic impaction parameters for subsequent fabrication of ultra-high purity PuO_2 . This material will be used for electrical conductivity research studies.

Initial experiments are in progress in impacting at temperatures above 2000°C. These temperatures should be high enough to allow sufficient plastic deformation in the ceramic during impaction to produce a structurally sound ceramic component.

Two alumel reinforced magnesia composites were prepared. Properties achieved were not satisfactory.

An unsymmetrical titanium shape was prepared by pneumatic impaction to demonstrate the fact that piece symmetry is not required.

Coining studies were continued on 30 vol% UO_2 -304L cermet. Strain gage and microformer type extensometers have been calibrated. Initial results on pins for cermet elastic modulus, yield stress, and tensile strength measurements have been obtained.

UO_2 spheres were formed by agglomerating UO_2 , coating it with aluminum powder, impacting the coated spheres, then dissolving away the aluminum.

NUCLEAR GRAPHITE

The binary gas system for controlling temperatures in the irradiation capsule in the GETR is working well. Maximum graphite sample temperatures have been maintained at $815^\circ \pm 10^\circ C$ for the entire reactor cycle. Two modifications to obtain heat-transfer data needed to modify capsule design when the reactor power level increases this fall were partially successful.

Tests are in progress to determine the optimum design for graphite irradiations in EBR-II. Carbon felt pads to protect sample ends failed when subjected to vibration tests simulating reactor conditions. The felt disintegrated and therefore cannot be used where mechanical wear may be encountered. Physics calculations have been started to determine the effect on fission rates and gamma heating rates caused by placing a capsule containing a moderator material (graphite) in the EBR-II fast-flux system.

At temperatures above 800°C the rate of air oxidation of graphite affects the rate of modulus change (and presumably also the strength). This indicates that the reaction occurs primarily at the external surface. It had been determined previously that below 780°C, the modulus change for a given burnoff is unaffected by the rate of oxidation.

A small amount of B_2O_3 appears to be present in boronated (B_4C) graphite. Removal of the B_2O_3 by leaching with water has only a small effect on the initial rate of reaction with oxygen, but does affect the rate upon continued oxidation.

Studies have begun on the mechanism of fracture in graphite. Microscopy is being employed to observe crack propagation. The first results indicate that cracks produced under tensile stress tend to be parallel to the layer planes of adjacent graphite crystallites. Once the tensile stress is relieved by fracture of the sample, not all cracks close again. Permanent set in graphite may be related to the non-closing of cracks when stress is removed.

IRRADIATION DAMAGE TO REACTOR METALS

Superalloys exposed to high-velocity helium at 2050°F in the Dynamic Materials Testing Apparatus show a marked increase in evaporation rate at gas flows >330 fps. A transition from laminar to turbulent flow in the test section, suspected of causing the increased rates, must be ruled out on the basis of gas flow-pressure drop studies.

In-flux hydrogen pickups for Zr-4, Zr-3 Nb-1 Sn and low-nickel Zr-2 exposed for 36 days in the ETR G-7 loop, high oxygen content water (1 ppm), were 10% or less, expressed as a fraction of the corrosion hydrogen. Hydrogen pickups were up to 30% for Zr-2 specimens.

Design was completed for the ATRC specimen container. Working drawings of the container assembly and a proposed typical loading pattern of nickel alloys for the loop during initial operation has been sent to NRTS for approval. cursory evaluation of nickel-base alloys for the alloy selection program continues with thirty-two assorted alloy tensile specimens tested the past month.

The 300°C in-reactor creep test of heat treated Zr-2.5 wt.% Nb was raised to a stress of 55,000 psi. The specimen had been stressed at 36,500 psi and 45,000 psi for which creep rates were less than 1×10^{-7} and about 3×10^{-7} in/in/hr, respectively. Comparable rates for the unirradiated material were 1×10^{-7} and 4.6×10^{-7} in/in/hr. The Zr-2.5 Nb in-reactor creep test conducted at 350°C, 36,500 psi yielded a creep rate of about 3×10^{-7} , while ex-reactor creep rate was 8×10^{-7} in/in/hr. Recovery of the specimen for postirradiation examination is planned.

A liquid-metal, temperature-controlled, irradiation capsule has been discharged from the ETR after obtaining an exposure of approximately 2.5×10^{20} fast fluence at an average irradiation temperature of 650°C. The AISI 304 and 348 stainless steel specimens from this capsule are being examined and tested. Several tensile tests between room temperature and 750°C were performed on AISI 304, 348, and 406 stainless steels irradiated in a gaseous environment at about 650°C to an exposure of 1.1×10^{20} fast fluence. The results of all tests performed to date on AISI 304, 348, 406, 410, and AM-350 stainless steels have been summarized in a paper to be presented at the ASTM Symposium on the Effects of Radiation on Structural Materials in June 1966.

Transmission microscopy has identified the precipitates induced by irradiation and/or tensile strain at 1350°F (730°C) in preconditioned Hastelloy X-280 to be $M_{23}C_6$ (10.6Å) and M_6C' (10.85Å).

Postirradiation stress-rupture data for preconditioned Hastelloy X-280 when compared with standard treated material has shown up to 106% increase in rupture life, 100% increase in duration of secondary creep, and 37% increase in duration of tertiary creep. These improvements are found to be due to a greater tolerance for plastic strain after irradiation by the microstructure developed by the preirradiation treatments.

Fracture toughness data from annealed Zircaloy-2 irradiated to 6×10^{20} n/cm² ($E > 1$ Mev) indicate that the decrease in toughness due to irradiation essentially saturates beyond 3×10^{20} n/cm².

Tests on 20% and 40% cold-worked Zircaloy-2 revealed that cold-working lowered fracture toughness slightly, promoted plane strain fracture at temperatures as high as 100°C, and caused considerable anisotropy in fracture toughness.

The fracture toughness of annealed Zircaloy-2 charged with 100 ppm hydrogen was found to decrease with increasing temperature in direct contrast with what is normally observed for Zircaloy-2.

The introduction of three common organic solvents into a crack in a high strength tool steel specimen very greatly increased crack growth rate.

An impurity effect which influences activation volume measurements of iron for carbon levels of 1 to 10 ppm has been observed. It is not apparent at this time whether the effect is due to dislocation motion or multiplication or is an indirect effect due to grain boundary segregation. Bend tests of iron single crystals have been used successfully to confirm the operation of (110)[111] and (211)[111] slip systems in unirradiated samples. Additional samples are being prepared for irradiation.

Calculations have been completed which show that the present heater design will meet the operating specifications while maintaining acceptable tube temperatures. These calculations indicated that the power distribution between the tube sections should shift depending on the inlet temperatures of the gas. This power distribution has been calculated for flow conditions covered in the operating specifications.

The prototype gas heater has been installed in the model loop. Welding of the primary piping has been completed. The loop has been leak checked. The heater was turned on while helium was being circulated and the heating elements operated satisfactorily at low temperature. Work on the water cooling system and electrical instruments is continuing.

METALLIC FUELS DEVELOPMENT

Irradiation of tubular thorium-uranium fuel elements continued successfully in the ETR P7 loop. Maximum fuel exposure is currently 13,200 MWD/T with a corresponding fuel volume increase of 2.1%. Examination of one test element with an exposure of 11,800 MWD/T is in progress.

Laboratory experiments have shown that Th-2.5% U-1% Zr alloy oxides are dissolved by 85% H_3PO_4 in about 1 hour.

ENGINEERING DEVELOPMENT

NEUTRON FLUX MONITORS

Performance of the experimental regenerating neutron flux detectors under extended irradiation in the KW Reactor continued to be successful. Preliminary microwave experiments indicate that alternate coupler configurations may be successful in minimizing reflection effects at the cavity-waveguide interface.

CRITICAL FLOW AT HIGH TEMPERATURES AND PRESSURES

The study of critical flow of steam-water mixtures discharging from pipes was continued with six experimental runs. It was found that a 7 in. long test section of 0.5 in. internal diameter was long enough to eliminate the effects of metastable flow conditions which are sometimes observed in short flow passages.

MICROWAVE DETECTION OF COOLANT IMPURITIES AND MEASUREMENT OF IN-REACTOR TEMPERATURES

Laboratory comparisons of dry and partially moist air indicated a shift in microwave frequency in good agreement with theory. Further stabilization of the signal generator will be required, however, for achievement of the desired sensitivity.

UPSTREAM BOILING BURNOUT

Laboratory experiments were continued to investigate the occurrence of boiling burnout at locations upstream of the exit of uniformly heated test sections. Data with a 12 ft long test section showed burnout occurring 3 ft from the upstream end when the coolant at the entrance was 32% quality by weight.

PLUTONIUM RECYCLE PROGRAM

FUELS DEVELOPMENT

The design of the PRTR flux monitor for operation with the HPD fuel elements, Mark I-R₂, was approved for fabrication and assembly.

The first plutonium recovered from Shippingport samples will be available for beta-gamma measurements by April 1, 1966.

A paper entitled "Plutonium Handling and Control Practices at Pacific Northwest Laboratory" (BNWL-SA-525) was presented at the March 1-2, 1966, Plutonium Symposium at Washington, D. C.

Fabrication of twelve 2 wt% PRTR fuel elements is on schedule for completion on April 29, 1966. Fabrication of fifteen UO₂ fuel rods is scheduled for completion prior to April 15, 1966.

Initial postirradiation examination results of a defected (1/16-inch diameter hole) vibrationally-compacted UO₂ fuel rod from test element GEH-12-39 revealed no distortion and some discoloration of the cladding downstream from the defect. The test element successfully operated 29.7 effective-full-power-days in ETR at a designed power generation of 26 Kw/ft.

Design and fabrication of a prototype FERTF test fuel element is nearing completion. The element will be available for flow testing in April, 1966.

Four non-failed fuel rods removed from the outer ring of FE-6504 show evidence of localized overheating of the cladding in the top end region. Analyses show that the O:U ratio of the fuel in the ruptured, previously intentionally defected rod increased from near stoichiometric to nominally 2.1 as a result of irradiation in the FERTF.

MATERIALS DEVELOPMENT

Dissolution experiments showed that 85% H₃PO₄ at 155°C will dissolve with over 90% efficiency arc fused and Sol-Gel thorium in 24 hours.

Corrosion tests with AISI 304 stainless steel in concentrated solutions of H₂SO₄-HNO₃-H₃PO₄ produced data showing corrosion losses at 100°C and 170°C were satisfactorily low.

A simplified H₃BO₃ analysis procedure based on measurement of the electrical conductance of H₃BO₃ before and after addition of d-mannitol does not appear to have sufficient sensitivity for use with trace concentrations of H₃BO₃.

Evaluation of a new resin material for H_3BO_3 removal from water indicates it is more selective for boron than for carbonate, hydroxide, and sulfate. This resin may be useful for recovering R^{10} from PRTR wastes generated during chemical shim operation.

It has been demonstrated that a magnetic field drastically effects the corrosion of carbon steel in citrate-oxalic acid mixtures. The attack is nonuniform. Additional tests are being conducted to further define this phenomenon.

Laboratory evaluations demonstrated that propargyl alcohol, phenylthiourea, diethylthiourea and the pertechnetate ion were inhibitors at low temperature ($25^\circ C$). At higher temperatures, 60 and $80^\circ C$, the pertechnetate ion was not effective. At higher temperature phenylthiourea was the most effective inhibitor followed by diethylthiourea and propargyl alcohol.

Operation of a PRTR dump valve has continued to be normal after three months testing over a 160 ppm boron solution.

Corrosion probes of aluminum and carbon steel located in the PRTR have indicated uniform corrosion rates of less than .01 mils/mo during the past month.

Two of three tests necessary to determine the degree of anisotropy displayed by PRTR Zircaloy-2 tube specimens have been performed.

REACTOR PHYSICS

The plutonium atom ratio data from 2 wt% PuAl fuel containing initially 16% Pu^{240} and the Pu^{239} atom density data, N^{49} , have been analyzed by fitting analytic functions to these data.

Correlation of analytical and experimental data for PuAl containing initially 1.8 wt% Pu^{240} has been completed using Program S.

A number of calculations have been made in support of the fuels testing program in the high power density experiment in the Plutonium Recycle Test Reactor.

Subcritical measurements using 19 rod clusters of PuO_2 - UO_2 fuel in a 13.86 in. hexagonal lattice were completed, and the core was returned to an 8 in. lattice spacing using 13 clusters.

The computational methods used in analyzing the effect of bowing of fuel rods in the Fuel Element Rupture Test Facility (FERTF) have been evaluated using experiments conducted in the Plutonium Recycle Critical Facility (PRCF).

Two papers, "Sensitivity of Reactor Multiplication Values to Cross Section Uncertainties of Thermal Systems" and "Sensitivity of Thermal Reactor Parameters to Scattering Model" were submitted for publication in the proceedings of the Conference on Neutron Cross Section Technology in Washington, D.C. The first was presented orally at the Conference.

Theory Development

To study the effect of particle size on reactivity for PuO_2 grains, existing resonance collision theories are being extended to include the moderation effect of oxygen.

Work on improvement of least squares fitting methods has led to discovery of a computer loop capable of learning by sifting linear correlations from a sequential flow of observations. A potential application is real-time processing of reactor measurements into evaluated reactor theory.

Code Development O4, O2

An improved treatment of resonance self-shielding was developed for the slowing down code HRG.

New data were added to the B-NW master cross section library for F-19, U^{232} , U^{234} , U^{238} , Na-23, and the isotopes of nickel.

Conversion of the SRL-BNW computer lattice analysis system HAMMER to the Univac 1107 was started.

O4

The TRUNK code to calculate the buckling of truncated spheres was completed and is being documented.

The isotropic scattering portion of the RBU Monte Carlo was tested. All the results indicate that isotropic scattering collisions are handled correctly.

The Rotating-Crystal Spectrometer was put back into operation following modifications to the crystal and chopper systems. A survey measurement was made of the inelastic scattering of 0.6 eV neutrons from H_2O .

Triple-axis spectrometer measurements were made on the inelastic scattering of neutrons from H_2O in order to reduce systematic errors in previous measurements.

The slow-neutron scattering law and cross-section programs LEAP, ADDELT, and SCS were assembled for independent or chained operation on the Univac 1107.

Design and construction was started on an interface circuit to incorporate the PDP-8 computer into the fast-neutron time-of-flight spectrometer system.

Work continued on the analysis and interpretation of variant-covariant error analysis of MeV total cross-section measurements.

CYCLE ANALYSIS

The four group version of the ALTHEAE burnup code is in operation.

Reducing the density of plutonium enriched thorium fuels in water reactor lattice may reduce fuel cycle cost approximately 2.0 mill/kWhr, which will make the fuel more competitive with slightly enriched uranium.

FUELS REPROCESSING DEVELOPMENT

Three of four samples of Shippingport Reactor Fuels have been dissolved. Plutonium has been recovered from two samples and partially decontaminated by batch contact solvent extraction procedures. Beta-gamma emission measurements will be made on the recovered plutonium as part of the program to characterize high exposure plutonium.

REACTOR ENGINEERING DEVELOPMENT

Heat transfer calculations were made to determine the temperatures which might be expected in a pressure tube or a basket following the rupture of a molten core fuel element in the Fuel Element Rupture Test Facility (FERTF). For the model assumed it was calculated that the temperatures would be high enough that the internal pressure could burst a Zircaloy pressure tube very rapidly, and a Hastelloy "C" tube in several hours.

Equations have been established for programming core thermal and hydraulic design and operating limits for the water moderated core studies.

Modifications were made to the computer programs to facilitate design of the particle separator for the FERTF.

TEST REACTOR OPERATION

The work in March continued to be directed toward those activities required for reactor startup with batch core loading. Replacement of the process tube-to-nozzle gaskets was completed and good progress made on the fabrication of the chemical shim system piping. Work continued on cleanup of the reactor with vacuuming and water flushes. Containment vessel decontamination continued to be a problem of holding contamination levels in various work areas to acceptable values. Current lower access space radiation levels range from 500 mr/hr at the angle valves to 1,800 mr/hr at the center of the reactor face. Fifty-one process tube inspections were made during the month. Preliminary evaluation does not disclose any significant damage. The general plans for the batch core critical tests have been prepared and the detailed test procedures completed.

EBWR DEMONSTRATION PROGRAM

Evaluation of available irradiation test data indicates that the production-run EBWR rods operated initially at predicted linear power ratings and that the change in linear power with burnup agrees with the calculational model within $\pm 6\%$.

A two-dimensional, two-group diffusion theory burnup code, EQUIPOISE BURNOUT, which recently has been made available has been evaluated using codes now in use.

ADVANCED SYSTEMS

U²³³ FUELING OF A FAST COMPACT REACTOR

Core sizes for internally-cooled fuel configurations were recalculated after discovery of a programming error. Pin core modifications to the program were completed. The TIGER V transient heat transfer code is being evaluated for use in the kinetics study. Reactor core size relationships are being recalculated using the Russian 26-group cross section set in place of the 16-group Hanson-Roach set previously used.

NUCLEAR SAFETY

CONTAINMENT SYSTEMS EXPERIMENT

Installation of the ventilation system was completed except for acceptance tests and the cooling coils. Instrumentation work was continued in support of CSE and included preparation of specifications for load cells and strain gages, modification of the impact transducer and cone assembly, and development of a readout circuit for the prototype liquid level probe.

Aerosol tests were performed to compare the behavior of individual components of an aerosol with that of a mixture of all five components. There is apparently no significant interaction of components in a steam-air atmosphere. Initial tests of different methods of injecting an aerosol into a pressurized system were performed.

PRESSURE VESSEL CRACK MONITORING

Burst-type acoustic emission signals were observed well before yield in laboratory tests of flaw propagation in stainless steel plates.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENT

Installation of the WSEP is continuing on schedule.

Investigation of the melt system $\text{PW-1;Na}_2\text{O:P}_2\text{O}_5$ was continued.

Data analysis was initiated to assess the effects of various Purex components on the phosphate melt system.

Good performance of the wiped film evaporator was achieved while producing a highly viscous product.

Waste containers were successfully cored for samples with a diamond tip drill.

A capacitance probe liquid level system was installed in the waste shipping cans.

Compacted densities of aluminum oxide showed significant variation with preheat temperature.

REACTOR SAFETY ANALYSIS AND EVALUATION

During this period several adjustments were made in the subsized-vessel specimen compliance testing procedure and the specimen dimensions. These changes will significantly improve the accuracy and reproducibility of the data, as well as reduce the technician and engineer time. Arrangements were made for the irradiation of additional subsized-vessel specimens.

Ultrasonic monitoring has been performed on five A106B pressure tubes at temperatures ranging from -60 to -30°C . The signals received have been modified and recorded and are now being analyzed by replaying them through an oscilloscope at various speeds.

COLUMBIA RIVER SEDIMENTATION STUDIES

Results of about 5000 analyses for radionuclides in 622 samples of Columbia River bottom sediment were turned over to U. S. Geological Survey during March.

FISSION PRODUCT AEROSOL CONTAINMENT

The stability of vapor mixtures of 35% hydrazine hydrate and methyl iodide was demonstrated as a safety measure preliminary to starting a planned series of experiments.

Methods were studied to determine those which would be most applicable to measuring the kinetics of the hydrazine-methyl iodide reaction. Reaction models were suggested and rate constants evaluated from nuclear magnetic resonance measurements obtained earlier.

LOW AND INTERMEDIATE LEVEL WASTE

The concurrent mixed oxide-ferricyanide precipitation process appears to be effective in substantially reducing fission product concentrations in a simulated waste solidification condensate. Separation of sulfate ion from glass melter condensate by sodium sulfate crystallization was investigated.

CUSTOMERASSISTANCE TO PHILLIPS PETROLEUM COMPANY

Four fuel element pins, loaded with 8 wt% enriched UO_2 pellets, were fabricated.

WESTINGHOUSE 25-ROD DNB STUDY

All fabrication necessary for Phase II of the test section to study DNB with a 25-rod bundle has been completed. The test section assembly should be finished during the first part of April, and installation in the vertical test section area will begin following the completion of the current experiments in the facility.

DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY04 PROGRAM - F. W. ALBAUGHCIVILIAN POWER REACTORSSPECIFIC FUEL CYCLE ANALYSIS - E. A. EschbachFast Reactor Fuel Cycle Costs

Recently, some effort has been expended in order that the economics section of the Fuel Cycle Analysis chain could be coupled to the ZODIAC code. Recent monthly reports have described the code, FAST, which takes ZODIAC data (punched on cards) and generates a burnup tape which is compatible with the Fuel Cycle Analysis chain. This recent work has been on a 1000 MW_e fast reactor design by Allis-Chalmers (A-C). Both UO₂ and PuO₂ cores have been used in the ZODIAC computations.

The costs in the table represent the overall fuel cycle costs of the reactor for several fueling and economic situations. The core burnup situation designated as "First Core" represents the burnup generated by ZODIAC for the initial core of the reactor. The situation designated "near equilibrium" was generated by taking the ZODIAC output and putting together a "best estimate" of an equilibrium burnup situation. These two situations are generated for the PuO₂ and UO₂ cores.

The economic situations represented in the table are similar to AEC fuel ownership in one case (4.75% fuel inventory charges) and to private ownership in the other (10% fuel inventory charges). All other charges are similar to those given by Allis-Chalmers for the Reference Design.

A comparison of the costs shown for ZODIAC results, and A-C results (in parentheses) for the plutonium indicates that the cost results from ZODIAC data are higher than the A-C numbers. A more detailed analysis not included here shows that this difference results from a difference in calculating interest charges on the fuel cycle by the QUICK cost code. The A-C numbers appear to represent only fabrication costs, and the results presented here include interest on the whole fuel cycle, which is believed to be more realistic.

The curves in the figure show the fuel cycle costs of the reactor as a function of the assigned plutonium price. The intersection of these two curves represents the indifference value of plutonium. The value of \$20/g fissile must be considered approximate, since the costs are based on the approximation of near-equilibrium. It would be fair, however, to say that the value will be very near \$20/g fissile, and roughly twice the value in a

FAST REACTOR FUEL CYCLE COSTS

Core Fuel	Core Burnup Situation	Approximate Core Enrichment, %	Fuel Cycle Costs, Mills/kWhe	
			1*	2**
Plutonium	First Core	20	1.323	1.653
Uranium	First Core	33	2.166	2.811
Plutonium	Near Equilibrium Core	20	0.968(0.78)***	1.205(1.11)
Uranium	Near Equilibrium Core	33	1.841	2.347

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* Fuel inventory working capital interest = 4.75%
 Fuel cycle working capital interest = 10%
 ** Fuel inventory working capital interest = 10%
 Fuel cycle working capital interest = 10%
 *** Allis-Chalmers numbers shown in parentheses

INDIFFERENCE VALUE OF PLUTONIUM FOR THE ALLIS-CHALMERS FAST REACTOR
UNDER PRIVATE OWNERSHIP ECONOMICS

thermal reactor. It should be pointed out however, that this is the indifference value of plutonium to this reactor, assuming the only source of plutonium to be operating this reactor as a fast converter. This represents the highest price a fast reactor of the A-C design would pay for plutonium. However, thermal reactors will be willing to supply plutonium to fast reactors at a price slightly exceeding \$10/g fissile. Thus the "probable market" price for plutonium as fast reactors are introduced will be between \$10/g and \$20/g. Although the fast reactor is a breeder (approximately 20-year doubling time), the fuel cycle costs increase as plutonium price is increased because the inventory costs are increasing faster than the plutonium credit. The percent plutonium added to the system by breeding must be greater per year than the economic interest rate on the reactor inventory before the costs will decrease as plutonium price increases.

Cross Section Studies

In an attempt to evaluate the range of applicability of one of the approximations made by ALTHAEA, the Bell⁽¹⁾ correction for lattice contributions was generalized for the case of an array of hollow cylinders. The procedure is straightforward, and the result is easily obtained. The Wigner or canonical approximation for the escape probabilities was made, but it is not a necessity nor a restriction. A problem, however, arises because of the final form of the result. For a solid rod, Bell found the result to be independent of the rod cross section or energy. Hence his generalization immediately carried over to the complete resonance integral via a substitution of his $\tau_0^{(1)}$ for the usual s_0 . This is not the case for a hollow cylinder, and one would either have to obtain this independence by approximations for some range of variables, or attempt to incorporate this energy dependence into the energy integral done in obtaining the resonance-integral. The evaluation of the present formula used for ALTHAEA will continue utilizing data from the RBU code for hollow cylinders.

Fuel Utilization Studies

The Allis-Chalmers fast reactor design has been simulated on the ZODIAC physics chain by the Reactor Physics group. The data from this has been translated into input data for the VESTA code using both plutonium and uranium enriched cores. The VESTA calculations are now being run.

The CURVES routine which has been added to VESTA is being used to draw plots of thermal converter characteristics. Some difficulty is being experienced in making the code label the curves properly. It is not anticipated that this difficulty will be persistent. The library routines are available for adding the CURVES subroutines to the 1107 version of VESTA.

(1) Bell, G. I., Nucl. Sci. & Eng. 5, p 138 (1959)

The final report on the VESTA code has been retyped but is waiting for inclusion of the fast converter data mentioned above. The CURVES and EPIMETHEUS work will not be incorporated into the final report, but will be the subjects of separate documents.

The cross recycle of plutonium from each of two PWR's, one clad with stainless steel and the other with zirconium, in the other is being calculated as a first step in preparing a two-reactor model with interaction between them. Analyses of the value of each type of plutonium in the other reactor are being performed.

Compound Interest Fuel Costs Calculations

Work is proceeding on the compound interest version of QUICK. The compound interest calculations in the working capital and fuel inventory cost sections of the QUICK code are currently being reviewed in detail for correctness, both of the equations and the programming.

CONCEPTUAL REACTOR DESIGN STUDIES - J. C. Fox

Nuclear Electric Plant Capital Costs vs. Site Location

Incremental plant construction costs associated with earthquake protection were initially considered to be a potentially significant cost which should vary with the seismic zone at each site. Further consideration of local geological conditions at the reactor sites of interest as well as the type of reference reactor indicates that plant construction costs will probably not be significantly affected by seismic considerations.

Variations in local labor rates at the sites under consideration are estimated to affect total direct construction costs by three percent at most. The hypothetical AEC site was selected as a reference site, with labor rates typical of western Massachusetts. A comparison of estimated direct construction costs at each site, based on local wage rates from the Bureau of Labor Standards, showed direct costs at sites in California to be the largest and about three percent greater than at the reference site.

A number of factors influencing site and plant capital costs in California were discussed with Bechtel Corporation personnel in San Francisco on March 16, 1966. It is apparent that coastal property costs can be as high as \$100,000 per acre and that \$20,000 per acre is not an unreasonably high value to assign to southern California coastal sites. The capital costs of large reactor plants located in high population density regions can be influenced by assumptions regarding isolation distance requirements and land costs. In highly populated regions

such as southern California problems arise, not only in determining appropriate isolation distances and exclusion areas, but in identifying population centers. Coastal sites located in relatively low population density areas may have transient population densities on weekends which significantly alter their compliance with any set criteria on distance from a population center.

Fast Reactor Fuel Cycle Calculations

Neutronics (statics and burnup) calculations for the large, fast-spectrum CE-type⁽¹⁾ carbide breeder reactor have been resumed. Using a one-dimensional model, burnup calculations for two plutonium fueled batch cycles have been completed. The first batch cycle calculation for the U²³⁵ fueled core has been started.

USAEC-AECL COOPERATIVE PROGRAM - F. W. Albaugh

Cross-Flow Mixing Between Parallel Flow Channels During Boiling

Analytical work and test section fabrication are continuing 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 subchannel of the fuel bundle. This information will provide a more accurate method to calculate coolant conditions at boiling burnout.

Fabrication of parts for the test section which represents the square-triangular type of flow channel is nearly complete. Welding jigs and fixtures have also been completed, and it is expected that the precision welding of this test section should be completed next month.

Fabrication of additional parts for the test section which will simulate the square-square type of flow channel has been started. The support structure which is common to both test sections is about 60% complete.

Calculations were continued this month to check out the computer program which is being developed to compute the local flow and enthalpy in multirod fuel elements. An unstable behavior of the numerical calculations has been found in some calculations. Several calculations have shown that unstable behavior can be eliminated by choosing a sufficiently large value for the cross-flow resistance and small step size. The calculations have also shown that the cross-flow resistance is a weak parameter, and increasing it over a couple orders of magnitude causes only small variations in the computed values. Further study will be given to this instability problem so that true values of cross-flow resistance can be used.

(1) Liquid Metal Fast Breeder Design Study, CEED-200, Volumes I and II, January 1964.

Hydraulic and Nuclear Stability in Parallel Flow Channel Systems

The entire set of nuclear and thermohydraulic equations describing the BLW-250 have been programmed on the analog computer. The system is currently being debugged in preparation for a test run to compare with an earlier calculation made with the digital computer code, MIDAS.

Activity during the month was principally devoted to the wiring and checkout of the analog computer circuits. A written description of the development of the present thermal and hydraulic equations is being prepared and will be reviewed with AECL personnel during their scheduled April visit.

Evaluation of Zr-2.5 wt% Nb Pressure Tubing

The general objective of this program is to evaluate Zr-2.5 wt% Nb alloy tubing as a pressure tube material with reference to the tests and reactor experience that has been obtained on Zr-2 pressure tubes. Continuation of the crack propagation tests have produced the following results.

The tests at -30°C continue to show that the extremely low hydrogen content (estimated at less than 10 ppm by metallography) of the 60% CW material strongly influence the crack propagation characteristics of the Zr-Nb alloy. The 30% CW material which contains about 30-40 ppm (estimated by metallography) exhibits brittle behavior, while the 60% CW exhibits ductile fracture.

A creep test has been started in the pressure tube creep apparatus. This test is scheduled to run for 400 hours uninterrupted at a temperature of 280°C and a hoop stress of 44,000 psi. The 400 hour cycle is determined by corrosion tests in parallel test legs of the 350°C loop. This test has run for about 250 hours without difficulties. Based on room temperature pressure versus strain calibration data, the pressure differentials indicate a total creep strain of about 0.3% to 0.4%. These values appear to be high but are uncorrected for temperature. Additional calibration data at elevated temperatures will be obtained.

APPLIED AND REACTOR PHYSICSPLUTONIUM CRITICALITY STUDIES - E. D. ClaytonBasic Criticality Experiments With Plutonium Nitrate Solutions in Slab Geometry

A series of criticality experiments will be initiated during April for determining critical thicknesses of bare and water reflected "infinite" slabs of plutonium nitrate solution. Installation of the criticality vessel for the experiments was completed in March and a checkout of equipment in preparation for the experiments is proceeding. The criticality vessel is of unique design in that its thickness can be varied throughout the range from three to nine inches. This is made possible due to the use of welded stainless steel bellows in construction of the vessel. Structural rigidity over the slab's thin stainless steel sides (42 in. x 42 in.) is accomplished by means of a low density lattice reinforcement. Accurate adjustment of thickness is obtained by use of a single crank geared to adjustment screws on the edges of the vessel. The vessel is mounted in the upright position to provide the greatest measure of safety in fuel addition and in entry of the control and safety blades into the solution. It is also equipped with solution dump, and reflector dump mechanisms.

The neutron decay constant was determined from pulsed neutron source experiments for the vessel filled with water. Later, pulsed neutron source data will be obtained with the vessel containing a known concentration of plutonium. These data will be useful in the study of proposed methods for criticality assessment of process equipment using pulsed neutron source techniques.

Criticality of Plutonium Compounds

Further criticality calculations were made for plutonium compounds on which experimental criticality data are lacking. The calculations apply to homogeneous mixtures of plutonium compound molecules in water for H/Pu atomic ratios in the undermoderated range with $H/Pu \leq 20$. Critical masses for bare spheres have now been computed for the following compounds with the H/Pu ratio extending up to 20: PuN , PuO_2 , $Pu(NO_3)_4$, Pu_2O_3 , PuC , Pu_2C_3 , PuH_2 , PuH_3 . Critical masses for reflected spheres were previously computed for these same compounds, and also for $PuCl_3$. Correction factors are being derived that will permit the determination of critical masses for these compounds over a range of densities for any given H/Pu ratio in the range $H/Pu \leq 20$. The DTF transport theory code, using multigroup constants (18 energy groups) obtained from the GAMTEC-II computer code, is being used in the calculations.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS STUDIES - R. E. HeinemanMockup and Materials Studies

A contract for fabricating the driver fuel pellets was awarded to the low bidder (Kerr-McGee Industries, Inc.) on March 8, 1966. The pellets are scheduled to be shipped 10 to 12 weeks after receipt of the enriched UF₆ from Oak Ridge National Laboratory. Shipment is expected about June 6, 1966.

Bids received for graphite cladding for the driver fuel and flux adjusting fuel rods are being evaluated.

The major effort on the development and fabrication of the oscillators during the month has been directed toward evaluation and improvement of the present design. Significant accomplishments during the month are: fabrication of the modified key and the modified carriages were initiated; design layouts of the modified oscillator are complete and drafting of the assembly and details was started; a pretensioned cable has been selected as the rotary to linear motion transition system; and evaluation of alternate drive concepts was continued during the month.

A purchase order for the temperature coefficient shim rods has been submitted. These will be graphite rods containing 0.65 mg/cc of gadolinium oxide clad in graphite.

The specimens of gadolinia-europia-dispersed nickel which were described last month have been tested in an environment simulating the conditions to which they will be subjected in the HTLTR safety rods. For the tests, a 1" x 3" sample of each of the four types of material was riveted between two 0.030" TD-nickel sheets, leaving the edges exposed. The four specimens were held in contact with graphite at 1000°C for 213 hours in a flowing nitrogen atmosphere containing 300 ppm water. The exposure did not result in any identifiable corrosion effect in the specimens. All specimens underwent a weight gain of about 0.1%, which is somewhat smaller than has been observed for thorium-dispersed (TD) nickel specimens similarly exposed, but in pure (no water) N₂. The gain in weight is attributed to pickup of carbon by the material.

Reactor Instruments and Equipment

The Programmed Measurement and Control System (PMACS) for the HTLTR was observed at the Anaheim plant of the vendor, Astrodata Corporation. It has been shipped and installed in the 318 building control room. Acceptance testing was performed on the system to determine if the system meets the PMACS specifications.

Regulated and isolated ac power for PMACS was not available during acceptance testing. A substitute source of power, which was unregulated and unisolated, was used. The regulating transformers originally purchased for supplying PMACS appear to be inadequate, and an electrically driven alternator has been recommended.

As part of the Acceptance Test Procedure, the safety system logic and power amplifier were tested under normal and off normal conditions. A test computer program was used to provide the PMACS originated normal signals needed to activate the safety logic circuit. Toggle switches were used to simulate the contacts of the trip relays of the two nuclear safety channels.

The safety logic operates to de-energize the power amplifier, removing power to the rod magnets if an off-normal condition were to occur on any of the contact closure outputs from the nuclear safety channels or on any of the outputs from PMACS. A bypass switch permits bypassing only three of the contact closures at a time. The manual scram pushbutton also may be operated to de-energize the power amplifier.

The final design of the safety system logic and power amplifier has undergone close scrutiny, a detailed reliability analysis having been included in the Final Safeguards Analysis. The safety system was tested to verify the results of the failure mode studies completed for the reliability analysis for those failure modes that could be simulated without disconnecting soldered joints or removing components from the circuit cards.

The performance of the Programmed Measurement and Control System was found to meet the specifications for the system in all but the following areas:

- a. High Precision Resistance Temperature Detector (RTD) - Required resolution is ± 0.01 °C and system capability appears to be only 0.014 °C resolution. In addition, the precision RTD system noise produces a 5-count scatter in the analog-to-digital converter for a fixed resistance input. The specifications allow ± 1 count of scatter. It is not clear at this time if the scatter is due to noise originating in the RTD system or if it is due to ac power noise. Further measurements will have to be made when regulated power is provided.

- b. Alpha-Numeric Display - The three-color IIT alpha-numeric display failed shortly after the system was installed and turned on. Replacement of a power transistor in the low voltage power supply and replacement of several logic transistors including the transistors in the three unblanking circuits. The circuits supplied appear to have been built around selected transistors with specially selected compensating components for each transistor used. Replacing the transistors required replacing the associated passive components. To eliminate this shortcoming, a new circuit was designed and built for the unblanking circuits.
- c. Analog Display - The analog-to-digital converters supplying X and Y deflection to the analog display are too noisy to meet the specification for that system. Noise of approximately 60 to 80 millivolts appears on both deflection signals.

In addition to the three areas listed above that do not meet the system specifications, the following problems were observed and resolved as listed:

- a. Analog Input System - The analog-to-digital converter has an intermittent failure of one bit. The failure appeared too seldom to troubleshoot. Astrodata has agreed to send replacement logic boards upon request (if the failure appears often enough to locate the fault), or they will check out the converter at their plant in Anaheim upon request. The common mode configuration required by the PMACS specifications for analog inputs was changed to allow the use of the input amplifiers which were supplied. All analog inputs must now be wired and installed to allow the guard wire to be ungrounded. Vitro Engineering has given their assurance that this new requirement for analog inputs is easily achieved with the present signal transducers.
- b. Bypass Switch - The bypass switch associated with the safety circuit occasionally caused the safety circuit to trip when moved to the bypass position. An alternate type of switch will need to be purchased and installed.
- c. Teletype - The teletype occasionally failed to advance the typewheel following the first character after a carriage return. The fault, which causes character strikeover to occur, is probably the result of a minor misadjustment and Battelle offered to locate the problem.

A listing was prepared of all inputs and outputs (such as thermocouples, stepping motors, etc) associated with PMACS and their positions on the input/output connectors has been assigned. This I/O assignment is now being reviewed by the PMACS programmers and will be forwarded to Vitro for their use during installation of signal and control wiring.

PMACS control of the electrical heater disconnects within the HTLTR system required further consideration during this report period. The present design of the electrical heater disconnects would not allow control directly from PMACS. Auxiliary relays or other intermediate equipment will be needed, and Vitro was made aware of the problem.

Information was requested on the saturable reactors and drivers for the heater rods and on the temperature sensing elements in the reactor pile for control purposes.

Operating instructions for a Univac 1107 program were obtained which solve for transient heat transfer problems in a reactor. A three-dimensional thermal model of the reactor was completed, and 90 per cent of the initial data cards for the 1107 program were listed for keypunching.

Before the heat transfer program can be run on the 1107, the heat loss mechanisms and the associated constants must be determined.

An approximate nodal model of the HTLTR was devised and the description put on punched cards for processing on the 1107 using the TIGER V code. A simple example was run initially to test the program, and then a more extensive case was run. Although there seemed to be little trouble in learning TIGER V, no useable results were gotten. The running time seems to be much too long for the data used. The cause of this trouble will be determined.

A literature search was conducted on the synthesis of sampled-data multipole controllers. This search yielded a few promising references that might be used as a basis for the actual controller design.

The HTLTR operations staff provided the requirements for the display format for the 3-color, alpha-numeric TV tube. Several alternate display command structures were discussed. When a definite choice has been made, actual writing of the display program will begin.

An orientation talk covering the HTLTR and PMACS was presented to interested staff members within the Laboratory. In addition, reactor operations personnel are being trained to prepare, edit, and assemble programs.

Tests of a Westinghouse ion chamber indicate that it is adequate for initial installation in the HTLTR. Testing methods to the extremely tight tolerances of Specification HWS-7552 cannot be made economically in a manner to show (beyond doubt) that all specifications were met. However, saturation curves to the upper limit of the specified flux range and linearity tests in this range both indicate that the Westinghouse chamber will be satisfactory

for the specified conditions, whereas the other chamber tested would not. Radiographs show that interelectrode spacing of the Westinghouse chamber is considerably less than for the Reuter-Stokes chamber; this tends to improve the saturation characteristics.

Reactor and Building Construction

The Phase I contractor successfully completed testing the fire protection system and it is now tied into the 300 Area system.

Acceptance of the building for beneficial use will follow the Acceptance Tests on the heating and ventilating system, adjustment of the system for proper air balance and satisfactory completion of the items on the final punch list. Approximately 30 days is estimated to complete these items.

Installation of the insulation inside the reactor shell was begun on March 30.

Experimental Program and Reactor Operation

The Final Safeguards Analysis for the HTL/TR, BNWL-CC-225, has been approved for transmittal to the Richland Operations Office of the AEC.

The Operating Limits for the facility have been prepared in rough draft and have been reviewed by Nuclear Safety and Engineering. The draft was revised and will be submitted to the Richland Operations Office.

A two day Operations training course on the design of the PMACS system was conducted by an Astrodata representative. Formal training on programming the computer for both the physics and operations staff is scheduled for April. The operations staff is working part time on the programmed instructions in PMACS digital programs. This is viewed as part of the training for reactor operation.

The central lattice test cell for the $7\frac{1}{2}$ " graphite lattice core for the HTL/TR is being redesigned to use small blocks of graphite. It had been planned to use large graphite blocks from the EGCR production lot, but these blocks are not necessarily going to be made available. The $3\text{-}3/4$ " lattice test core drawings were revised to show heater rod and thermocouple details. The $4\text{-}3/16$ " lattice test core drawings are also being revised.

At the March 1 to 4, 1966 meeting of the Technical Advisory Committee on HTGR development, a presentation was given on experimental work planned for the HTL/TR during the first few years of its operation. Points emphasized were the background of experience derived from past work with the PCTR, and the pertinence of the work with the HTL/TR to both HTGR safety studies and the development of HTGR advanced concepts.

A meeting was held at PNL on March 21, 1966 with a representative of General Atomic to discuss further the experiments to be done at the two laboratories. Information to be exchanged was listed and program modifications were considered which would bring the work at the two areas into better alignment.

Work is continuing on the planning of the HTLTR experiments. The program will be reviewed again by the Technical Advisory Committee in April.

PHOENIX FUEL REACTOR PROGRAM - P. L. Hofmann

MTR-Phoenix Experiment

Cross Section Evaluation

Beginning with the current MTR core design work, an effort to use Master Library cross sections in all codes used has been initiated. As part of this effort, a special GAM library was generated from Master Library cross sections. To compare these GAM-format, BML cross sections with the "traditional" set of GAM cross sections, a PHYSICS CHAIN calculation of the infinite multiplication factor for an MTR plutonium core was made using the two cross section sets. The two sets differ by only 0.001 $\Delta k/k$. Hence the use of BML cross sections in place of the original GAM set is a negligible perturbation - at least for Phoenix type cores.

A similar comparison was made using HRG, non-resonance treatment with the same BML cross section set, and the results were again identical to the GAM results with the same cross sections.

Nuclear Design Calculations

The determination of a three-dimensional power map for the MTR experiment continues to be an important design problem. The splicing together of three 2-D calculations (a rodged region, a non-rodged region, and a fuel follower region) to form a 3-D composite by using a 1-D axial calculation does not appear to be a promising procedure. This method requires that each of the 2-D calculations to be joined is homogenized into an equivalent zero dimensional system with transverse leakage. However, the nuclear characteristics of the Pu and U zones are so different that the homogenization procedure becomes rather arbitrary. Work is now underway to make use of a 2-D calculation for splicing so that homogenization is necessary in only one dimension. When this method has provided an approximate critical rod position, the 3-D code WHIRLAWAY will be used as a check on splicing methods. The splicing process is not only less expensive than a full blown 3-D calculation but a good deal of insight can be gained by examining the more easily understandable 2-D flux and power maps.

CAF-Phoenix Experiment

Preparation for the experiments in the CAF in support of the MTR-Phoenix Fuel Experiment are complete. A safeguards report and Criticality Safety Specification No. B-22 were prepared during the month. Both are awaiting final approval.

MTR Core Thermal Hydraulics Computer Program

Changes of the MTR fuel box configuration effect changes in the core flow rate since the plenum-to-plenum pressure drop in the MTR remains essentially constant. Since it is desirable to investigate fuel-box configurations other than the one now under consideration (sixteen 80-mil thick plates) the appropriate hydraulics calculations were incorporated into the previously written computer program. Also the program was altered to iterate on a maximum permissible clad surface temperature so that a wide range of power capabilities for various horizontal and axial peaking factors can be investigated in a minimum amount of time.

Further calculations will be made as soon as power distributions are received from Engineering Physics.

Fuel Fabrication

The second 2-kg batch (Lot No. RB) of PuO_2 was reduced to Al-20 wt% Pu alloy which was cast into billets and extruded to sheetbars. The extruded sheetbars were cut to length for rolling. These, as well as the sheetbar selections from the first casting lot (RA), are being rolled. The blister problem which occurred in the first lot of extrusions was essentially eliminated in the second lot by modifying the casting procedure. The first scrap recast lot (RC) was completed, and the billets were extruded to sheetbars. These were cut to length and are being rolled. Early analyses showed that alloy Lots RA and RB meet the specification of 20 ± 1 wt% Pu with 19.0 ± 0.5 wt% Pu^{240} . The prototype welding system is expected to be completed in April. A welding mechanism was ordered this month to replace a borrowed unit.

REACTOR FUELS AND MATERIALSFAST FUELS OXIDES AND NITRIDES - W. E. RoakeFTR Driver Fuel Irradiation Tests in EFFBR

A program for irradiating UO_2 - PuO_2 and PuO_2 -stainless steel cermet candidate FTR driver fuel in the Enrico Fermi Fast Breeder Reactor (EFFBR) was formulated.

Eight UO_2 - PuO_2 fuel pins designed and fabricated by PNL are presently scheduled to be part of the USAEC fuels and materials irradiation testing program in EFFBR. Six of these pins will contain mechanical restrictors designed to reduce or eliminate fuel relocation. The other two pins will contain no restrictors and will be used as "controls". Fuel form (powder and pellet) and fuel smear density will be varied. Upon completion of irradiation in EFFBR, fuel pins of each type will be selected for irradiation in TREAT to examine the effect of transient conditions on fuel relocation characteristics. Similar unirradiated pins will also be tested in TREAT to determine the effect of burnup on relocation behavior.

Four PuO_2 -SS cermet pins to be irradiated in EFFBR will contain PuO_2 particles of different densities and different core-cladding bonds. Test pins will be prepared by coextrusion (metallurgical bond between core and cladding) and pelleting (sodium-filled annulus between pellets and cladding). Some of the pins irradiated in EFFBR will be used in TREAT experiments. Similar unirradiated pins will also be tested in TREAT. In addition to the irradiation stability of the matrix and cladding and its effect on thermal expansion, other observations and measurements will be made on fuel pin swelling, structural property changes, fuel-matrix-clad compatibility, and core-cladding bond characteristics after irradiation in both EFFBR and TREAT.

Companion UO_2 - PuO_2 and PuO_2 -SS fuel pins will begin irradiation in ETR/MTR in advance of the EFFBR irradiations. Several of these pins will also be used in TREAT experiments. The ETR/MTR irradiations will aid in interpreting data from the EFFBR irradiations and will point out in advance when, and if, problems are to be encountered in the irradiation of the similar pins in EFFBR.

The twelve fuel pins will be charged into EFFBR at the time (about January 1967) the reactor power level is increased to 110 Mw. Goal exposure for the eight mixed-oxide pins is 50,000 MWD/MT, and for the four cermet pins is 9% burnup of the plutonium atoms.

Work on these tests during FY-1966 is being performed as part of the Fast Fuels Oxides and Nitrides Program. During FY-1967 and beyond, work on these tests will be performed as part of the FFTF Project.

High Pressure Plutonium-Uranium Nitride Studies

Installation of the high pressure (2000 psi), high temperature (3000°C) filament furnace in a plutonium-contaminated glove box was completed. Calibration of the system for temperature and pressure measurement is in progress. After calibration, the dissociation pressure curve for PuN will be extended, and an attempt will be made to determine the congruent melting point.

Modifications were designed for conversion of an existing, uncontaminated glove box to a pure-atmosphere facility. A recirculated argon atmosphere maintained at less than 10 ppm water and oxygen is to be used for work with unstable fine powders of uranium and plutonium compounds, (i.e., nitrides and carbides).

Thermal-Expansion Apparatus

The tungsten - fused quartz sample holder proved successful in reducing the errors resulting from thermal-expansion differentials in the dilatometer. Although an increased accuracy was obtained, the sample holder was further modified to increase the mechanical strength of the combined unit. Subsequent vacuum problems have delayed calibration of the new holder.

Thermal-Diffusivity Apparatus

The laser-pulse thermal diffusivity apparatus was installed in the hood. Calibration with standard UO_2 samples is now in progress. Several glass windows in the apparatus must be replaced with infrared-transmitting materials before measurements below 200°C are possible. This modification should also increase the higher-temperature sensitivity.

A modification of the bias voltage supply for the indium antimonide infrared detectors increased the usefulness of this detection method to above 1500°C. Thermal-diffusivity data were obtained at approximately 1500°C for a UO_2 sample. Previously, detector sensitivity decreased drastically above approximately 700°C until at 900°C small temperature transients could not be measured. The diode detector can possibly be used to look at even higher temperatures by using proper optical filters.

Electron Microscopy of Plutonium Oxides

Samples as low in stoichiometry as $\text{PuO}_{1.72}$ did not show any significant changes on heating in the reflection electron microscope to 700°C. Modifications of the present technique to improve sensitivity are underway. The hydrogen flow introduced into the microscope again proved successful in preventing oxidation of the specimen. The microscope is still uncontaminated.

A program for irradiating candidate FTR driver fuel in the Enrico Fermi Fast Breeder Reactor (EFFBR) was formulated.

BASIC SWELLING STUDIES - R. D. Leggett

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 and thorium-uranium alloys.

A controlled temperature capsule and a tandem assembly of two capsules continue to operate successfully. One of the tandem capsules is operating at a pressure of 1000 psi and 450°C toward a goal exposure of 0.75 at.% B.U.; the other is operating at low pressure (approximately 30 psi) and 525°C toward a goal exposure of 0.4 at.% B.U. The single capsule is controlled at 700°C (beta phase) and is scheduled for irradiation to 0.2 at.% B.U. in the specimens. These capsules, containing enriched and natural uranium specimens, will provide data needed in evaluating the effects of temperature (both alpha and beta phase), pressure, and burnup on the behavior of various fuel compositions.

Construction of a controlled temperature (low pressure) irradiation capsule designed to operate at 450°C toward a goal exposure of 0.3 at.% B.U. is essentially complete. The capsule will provide information on the effects of burnup and minor alloy additives on the irradiation behavior of uranium base fuels.

The possibility of irradiating capsules containing uranium samples of three enrichments which would provide three burnup levels in one capsule is being investigated. Construction has been started on a controlled temperature (low pressure) irradiation capsule which has been modified to allow relatively easy adjustments in the heat transfer fins to be made. Laboratory experiments will be conducted to determine the heat transfer characteristics of future capsules.

Two capsules controlled at 500 psi and 550°C and 625°C respectively, to about 0.16 at.% B.U. were opened and the specimens recovered. Density measurements indicated that roughly twice as much swelling had occurred in the 500 psi tests as had occurred in the 1000 psi tests at comparable temperatures for the high purity uranium specimens. The specimens continue to exhibit less swelling at 625°C than at 550°C. This confirms previous conclusions that swelling in an isotropic material such as uranium is a much more complex process than simple fission gas agglomeration. The swelling observed in the dilute alloy specimens included in these capsules was small and not capable of being distinguished from that

observed in the 1000 psi tests. All of the specimens are being stored for future metallographic examination.

Th-1.0 wt.% U and Th-5 wt.% U specimens irradiated cold to about 1.5 at.% B.U. and postirradiation annealed at 725°C for 100 hours were re-examined after an additional 140-hour anneal at 825°C. A few grain boundary pores are now present and small (approximately 300 Å) matrix pores are starting to appear. The latter are so small they are almost beyond the resolution limits of the metallographic technique employed. The 1.0 wt.% U specimen has now swelled to about 2% whereas the 5.4 wt.% U specimen has swelled about 3%. This is consistent with the lower solidus temperature of the 5.4 wt.% U specimen. The specimens are being stored for additional annealing studies.

NONDESTRUCTIVE TESTING - J. C. Spanner

Detection of Irradiated Induced Shift in Nil-Ductility-Transition Temperature

Critical angle ultrasonic methods are being evaluated to determine their effectiveness in detecting irradiation-induced brittleness in reactor pressure vessel steels. The two main objectives of the present program are to obtain substantiating data on previously examined irradiated specimens and to design a facility for an expanded methods development program.

A facility has been obtained for handling irradiated specimens and modifications are being made to the fixtures which will be used in making the ultrasonic measurements. Two additional irradiated specimens have been made available from the U.S. Naval Research Lab, and shipping procedures have been initiated.

Design of a more permanent facility for future ultrasonic measurements on irradiated specimens has been started. A test tank with manipulator apparatus will be contained within a lead shielded cell with provisions for remote adjustments of transducer alignment.

Electromagnetic Testing

Tests to determine the response of the multiparameter eddy current tester to different input signals of the types associated with a range of typical test specimen conditions were continued. Comparisons were made of the multiparameter tester performance with inputs from two linear signal simulating devices and from a test coil adjacent to standards having fabricated flaws. These tests showed that curvature of the signal loci, due to detector nonlinearity, was small compared to the curvature due to the flaw signals of interest.

Simulated parameters, having linear signals differing by as little as 5 degrees in signal space were separated. However, using test hole standards which give curved loci in the signal space as the test probe approaches the flaw, it was found that additional parameter separating capability of the instrument must be used to separate each type of flaw. This leads to the tentative conclusion that the curved loci can be separated, but that the dimensionality of the test must be greater than the number of nominal (apparent) test parameters. It was also observed that the end points of the curved signal

loci due to scanning holes of various diameters and equal depths lie in line. Although in contrast, the end points of the loci due to holes of different depths do not lie in line. The loci due to a particular hole depth can duplicate, in part, a loci due to a hole having a different diameter. Thus, an adjustment of the instrument which eliminates or reads out this particular hole depth signal would also eliminate or read out the hole diameter. To avoid these effects, circuits are being built to sample each defect signal at its maximum amplitude eliminating the curved locus caused by the scanning of the defect. This will result in signals having less information content but which may be more appropriately characterized by fewer dimensions.

A different transformation (summing) circuit arrangement for the multi-parameter tester is being evaluated in which one parameter at a time is eliminated by a series of adjustments in contrast to the previous method of eliminating three parameters at a time. With this method, the signals can be evaluated at various stages of the summing process as has proved valuable in the present work. Further comparison of the two methods is needed due to the greater circuit complexity of the new technique. The new method was instrumented using an AD-1 analog computer on loan from the Simulation Laboratory.

Further investigation of phase sensitive detectors is being conducted. The approach chosen involves resolving the signal S where $S = s \cos \omega t$ into two components, $S \cos \theta$ and $S \sin \theta$, where θ is the phase angle with respect to some reference. Then, applying $S \sin \theta$ to the vertical of an oscilloscope and $S \cos \theta$ to the horizontal, a display of $|S|$ and θ is obtained.

A 6-diode gate, using 1N914 diodes, has been chosen as the basis of the detector. In the on-state, linearity of the gate is better than 1% for signals from 20 mV peak-to-peak up to 5V peak-to-peak for frequencies up to 500 kHz. Leakage in the off-state is less than 1% up to 250 kHz. At higher frequencies, leakage increases due to diode capacitances. Gating speeds up to 250 kHz have been achieved with only moderate distortion. At lower gating speeds, distortion is negligible. Further work is needed to improve the turn-off characteristic of the gate to allow higher gating rates.

Initial observations of one channel of the phase detector are inconclusive, but appear promising. More conclusive results are expected upon completion of two 6-diode gate circuits when the entire phase detector can be tested.

Infrared and Thermal Research

Development of thermal wave testing methods continued. Most practical applications of thermal wave measurements require accurate detection of small differences in thermal properties of the samples. In previous attempts to apply the sinusoidal thermal system to highly conductive practical samples, such as metals or graphite, poor signal-to-noise ratios made it impossible to detect small differences in the samples. The signal-to-noise ratio was improved by replacing two of the amplifiers in the electronic drive and data acquisition system, and several experimental measurements which had not previously been possible were obtained.

Measurements on aluminum, lead, stainless steel, and graphite, using the miniature thermal transducer, showed that the transducer was not obeying plane thermal wave theory; the transducer thermal impedance appeared to be more nearly a function of thermal diffusivity than of the $KC\rho$ product (thermal conductivity $\times$ heat capacity $\times$ density). In addition, the sensitivity of the transducer, when applied to metals and graphite, was lower than theoretically predicted. These effects are most likely due to heat losses at the end supports of the transducer and to the tendency of the transducer to act as a point rather than a plane source. Both of these effects should be reduced by increasing the size of the transducer, or increasing the thermal wave frequency. (Thermal wave lengths in the samples of metal and graphite at the frequencies used in the present measurements were much greater than the smallest dimension of the miniature transducer.) A larger transducer, which should have higher sensitivity and which should give results that can be interpreted in terms of plane thermal waves in high conductivity samples, has been fabricated. Experimental and theoretical investigations to measure thermal diffusivities with the miniature transducer are continuing.

Present thermal transducers have a gas-filled gap between the transducer and sample surface. This gap provides a convenient and flexible coupling that can be used at high or low temperatures. Due to the low heat capacity of the gas, finite diffusion times and the complication of the accompanying phase shifts in thermal waves arriving at the sample surface need not be considered. However, changes in coupling due to variation in gas gap width and composition will have an effect on phase and amplitude of the transducer's response to the sample. For this reason, methods of eliminating the effects of coupling variations are being studied. Early theoretical work showed that changes in coupling resistance between the transducer and sample should cause the thermal impedance of the transducer to follow a locus on the Argand plane which is different from the locus generated by changes in sample properties. The theory furthermore predicted that for small gaps and low frequencies, the locus generated by changes in gas composition should be the same as that generated by changes in gap width. Hence, it would be possible to eliminate the effects of small changes in gap width and composition of the coupling gas by measuring only the component of impedance difference perpendicular to the width-composition locus. This possibility has been experimentally confirmed using the miniature transducer with gap widths between 0.003 and 0.006 inch and coupling gas compositions between 100% air and 100% helium. The theoretically predicted method of automatically compensating for coupling variations has proved correct for this one case. Experiments to determine applicability of this method will be carried out for other cases as new transducers are fabricated.

Fundamental Ultrasonic Studies

Wave model studies and transducer development work was continued. The analysis of energy flux associated with attenuating media was completed, but the complexity of the analytical expressions prevents simple interpretation.

It will be necessary, therefore, to compute by numerical methods the quantities which describe the energy flux as a function of incident angle. Digital computer programming to facilitate this energy flux analysis has not been undertaken to date.

Efforts to resolve the remaining differences between experimentally observed pulse shapes, as compared to those calculated with the attenuating wave model, continued. The frequency spectra of the measured pulses were obtained and compared with the calculated spectra. The frequency spectra of the residual signals were also obtained. From these it appeared that the greatest discrepancies between measured and calculated results were at the lower frequencies. It was suspected that diffraction or beam spreading errors might be responsible since these effects are more pronounced at low frequencies. Also, previously recorded experimental pulses had been obtained at incident angles which were more oblique than was necessary to obtain the desired pulse distortion effects. At the greater incident angles, diffraction errors are also more pronounced. A new series of experimental pulse recordings were obtained at less oblique incident angles to minimize diffraction errors.

Since discrepancies in the low frequencies of the pulses can also be caused by the use of incorrect attenuation data, some variations in the slopes of the attenuation curves were also used in the wave analysis. These experiments have not been fully analyzed, but initial results indicate that there are still some pulse shape differences that remain unexplained.

Transducer fabrication and poling work continued. Six spherically focused transducers were fabricated from ceramic PZT-4 material. Since it was found that the transducer elements distorted appreciably during the poling process, they were prepared in roughly the desired size and shape and were to be poled before final grinding to the desired thicknesses. During the poling process, however, several elements fractured due to the poling forces. The reason for this has not been fully determined, but it is believed to be due in part to the fact that there were undesirable thickness variations. This problem cannot be resolved until equipment for grinding focused elements with uniform thicknesses has been obtained.

Basic studies on the self-focusing properties of intense ultrasonic beams have been initiated. The initial work consists of studying the self-focusing properties of intense laser beams due to nonlinear effects in the effective index of refraction of the material. When the laser beam is focused in the material, quartz for example, the beam diameter remains at the focal point diameter over an appreciable distance (centimeters). The energy density is proportionately higher in this self-focused state. It appears that this same effect may occur with ultrasound when the ultrasonic beam intensity is capable of producing nonlinear effects in the ultrasonic analog of the optical index of refraction.

NUCLEAR CERAMICS - R. E. Nightingale

Pneumatic Impaction of Ceramic Materials

Initial experiments are in progress in impacting at temperatures above 2000°C. These temperatures should be sufficient to allow sufficient plastic deformation in the ceramic during impaction to produce a structurally sound ceramic component.

The three techniques of interest include:

- a. Flame spray coating of Mo as a protective layer during preheating of the impaction can.
- b. Heating of molybdenum to 2200°C in molten cordierite or molten SiO₂.
- c. Slurry coating of a graphite impaction.

Induction heating to 2000 ±100°C of non-coated and flame sprayed Al₂O₃ coated molybdenum rings 2" in diameter x 1/16" thick x 1" high resulted in a burn out failure of the non-coated ring in 10 minutes and a burn out failure of the coated ring (0.008" thick coating) after 57 minutes. Further tests are in progress.

Attempts to heat molybdenum rings in a non-reacted mixture of 153.1 wt% SiO₂, 36.1 wt% Al₂O₃, and 10.8 wt% MgO (cordierite - Mg₂Al₄Si₅O₁₈) resulted in the formation of a thin glassy coating on the molybdenum which prevented rapid oxidation at high temperatures. However, the powder within a half inch of the ring wall was only consolidated, not melted; the powder in the center of the ring was unaffected during the heating. This result raises the question of the temperature distribution within a nupac can when impacting a ceramic.

Further experiments will be conducted using pre-reacted cordierite and immersing the test rings into molten cordierite directly and after preheating.

A literature survey on the coating of graphite for use at high temperatures is in progress. The most promising coating material found to date is iridium metal.

Fiber Reinforced Ceramic Matrix Materials

Two alumel reinforced magnesia composites were pneumatically impacted at 1200°C and 1400°C respectively. The fibers in the 1200°C impaction deformed into the matrix. The 1400°C impaction showed gross reaction between the metal and the MgO matrix.

Further development on pneumatic impaction of ceramic materials appears necessary before ceramic matrix materials can be successfully impacted.

Vacuum System

A new Heraeus-Engelhard high vacuum pumping system has been delivered. This system will be used principally in the pneumatic impaction operation. The system consists of a mechanical pump with a capacity of 26 CFM, and a 6" diffusion pump with an ultimate pumping speed of 1500 liters per second. The system came completely assembled and mounted on a unistrut frame.

Material Development

A titanium shape was fabricated by pneumatic impaction to determine the effect of starting with an unsymmetrical powder preform. The preform was prepared by vibrationally compacting powders into a mold. Minus 100 mesh titanium powder was used. Due to the low particle density of this powder, the bulk volume of the shape greatly decreased. However, no distortion effect was noted due to asymmetry. The pneumatic impaction conditions were a 35 μ vacuum, 352,000 psi pressure and 1025°C.

Coining Studies for Fuel Pin Fabrication

Pellets of 30 vol% UO₂-304L stainless steel were cold pressed, annealed and coined to densities of approximately 90%. An extrusion using the coined pellets is planned. A need for further coining studies will be determined after the extrusion is evaluated.

Cermet Fuel Evaluation

Strain gage and microformer type extensometers have been calibrated and are currently being tested with samples of 304L stainless steel, aluminum, and cold drawn 1020 mild steel.

Tensile samples are being machined from extruded and cold swaged fuel pins for elastic modulus, yield stress, and tensile strength measurements.

Initial results of the cermet mechanical testing program are:

<u>Fuel Pin</u>	<u>Elastic Modulus</u>	<u>Yield Strength</u>	<u>Ultimate</u>	<u>% Elong</u>
10 vol% UO ₂ Swaged 5U-33-S	14,200,000 psi	71,600 psi	78,600 psi	3.0%
10 vol% UO ₂ Extruded 5U-33-E	16,100,000 psi	29,300 psi	56,500 psi	25.0%

A comparison of mechanical strength versus UO₂ content is planned for next month.

Pneumatic Impaction Forming - UO₂ Spheres

The application of essentially an isostatic pressing technique utilizing the pneumatic impaction machine has yielded better than 90% TD UO₂ spheres in preliminary tests.

A low melting aluminum powder heated under vacuum to a highly plastic condition was used as an energy transfer media to densify UO₂ spheres. The UO₂ was formed into spheres by aggregation with a binder, sieved through screens to obtain uniform sizes and blended with the Al powder which formed a continuous metal matrix around the spheres.

The mixture was pneumatically impacted at 550°C and approximately 280,000 psi. The particles were removed by dissolution of the aluminum in caustic.

Results, approximately 90% TD and visually spherical appearing shapes are very encouraging. To optimize the process, the variables to be investigated are the correct relationship between die wall and loaded volume, the volume relationship between ceramic and metal matrix, and the optimum temperature for impaction. Should additional temperature be necessary, an energy transfer media of higher melting point than aluminum may be required.

Materials for Basic Research

Characterized UO_2 single crystal specimens were prepared and sent to Los Alamos Scientific Laboratory, Pennsylvania State University, and Arizona State University, for use in basic properties studies. In addition, UO_2 single crystal specimens were prepared for shipment to the University of Jerusalem, Israel.

High density, pneumatically impacted fuel pellets are needed for electrical conductivity studies. Ultra-high purity PuO_2 (<60 ppm total impurities) will be used in fabricating PuO_2 and $\text{UO}_2\text{-PuO}_2$ pellets. Two pellets were fabricated using UO_2 as a "stand-in" material to develop procedures and hardware requirements, and to establish impaction parameters.

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Stoichiometry Gradients in Oxide Fuels

Irradiation of UO_{2+x} fueled capsules designed to study oxygen redistribution and bulk composition effects on fuel-temperature distribution was completed. Five capsules, containing $\text{UO}_{1.90}$, $\text{UO}_{1.99}$, $\text{UO}_{2.02}$, $\text{UO}_{2.05}$, and $\text{UO}_{2.15}$ were irradiated at 30 kw/ft for 16 hr. Each capsule also contained a $\text{UO}_{2.00}$ internal standard.

Post-irradiation evaluation, consisting of metallography, autoradiography, and radial sampling is in progress. All specimens were sectioned, and metallography was completed on the specimens containing $\text{UO}_{1.90}$, $\text{UO}_{2.02}$, and their standards. Mean thermal conductivities of these materials, deduced from the structure of the irradiated fuel, are as follows:

<u>Specimen</u>	<u>$\bar{k}$, Fuel surface to columnar grain termini (400°-1700°C)</u>	<u>$\bar{k}$, Fuel surface to melt radius (400°-2800°C)</u>
UO _{1.90}	27% >UO _{2.00}	24% >UO _{2.00}
UO _{2.02}	12% <UO _{2.00}	16% <UO _{2.00}

Attempts to remove radial samples from irradiated specimens using diamond core drills were not successful. Apparently the very high density oxide used in these experiments is not as machinable as the sintered pellets drilled previously. New drilling procedures and new drill designs are being worked out.

Electrical Conductivity of UO₂

Preliminary data for polycrystalline UO₂ indicate that the electrical conductivity is a function of the frequency of the alternating current. At temperatures below ~750°K, the conductivity increases with increasing frequency. Between ~750°K and 1650°K, the conductivity is independent of frequency. Above ~1650°K, the conductivity decreases with increasing frequency.

For single crystal UO₂, a similar frequency dependence was observed only in the low-temperature region. Further studies are being made to determine the cause of the frequency dependence.

Mechanical Thinning of Ceramic Materials

The mechanical thinning technique for ceramic materials recently developed has been applied successfully to a single crystal of UO₂. Approximately 50% of the specimen area was transparent to electrons. Features tentatively identified as stacking faults and dislocation loops have been noted. The diffraction patterns obtained from some areas indicated strain in the lattice. These strain effects are apparently generated during specimen preparation. Work is underway to eliminate them.

High-Energy-Rate Impaction: Bonding Mechanism Study

The ten cans of the first (exploratory) series in the high-energy-rate impaction study were opened and the specimens recovered after impaction at 1140°C and 285,000 psi. Evaluation is in progress. Several significant observations have been made to date.

- (1) Good bonding occurred between green powder pellets, and between single crystal and powder pellets. No appreciable bonding occurred between single crystal pairs except where Al_2O_3 (pressure-transfer matrix) was obviously present in the interface. The non-bonding may have resulted from positioning the crystals with their (111) natural fracture planes horizontal, (perpendicular to punch movement and parallel to the usual planes of high-pressure laminations); or use of $\text{UO}_{2.00}$ single crystals instead of UO_{2+x} .
- (2) Similar densification, movements, fracture, etc. occurred whether specimens were contained in a carbon crucible, wrapped in molybdenum foil, or were unprotected from the Al_2O_3 matrix.
- (3) The Al_2O_3 behaved as a very fluid medium in every case, and penetrated to the interior of most of the containers. This penetration probably destroyed interface purity in some specimens.
- (4) Density changes, determined from a limited number of observations, showed that the UO_2 powders, graphite crucibles, and Al_2O_3 matrix materials were all densified during pneumatic impaction.

Growth of PuO_2 Single Crystals

A cursory literature survey was made to identify possible methods for the production of PuO_2 single crystals. Preliminary experiments were initiated to investigate the growth of PuO_2 from a fluxed melt. A 30 wt% Na_2O - 70 wt% SiO_2 glass which will be used to dissolve PuO_2 in air at temperatures in the range 1500-1900°C was prepared. Subsequent nucleation and growth will be investigated by controlled annealing at temperatures below the dissolution temperature.

Thermionic Emission Work Function

The thermionic work function of polycrystalline tungsten foil is being determined in the retarding-potential apparatus. A vacuum of approximately 1×10^{-9} torr has been reached at room temperature; after cleaning the sample by flash heating to approximately 2500°C, the work function will be measured.

Differential Thermal Analysis

Several samples of lithium aluminate were run in the new DTA apparatus. Reactions were detected at 200°C (water dehydration), 725°C (Li_2CO_3 decomposition) and at 1030°C, 1130°C, and 1260°C. The 1030°C and 1260°C are reversible; however, the 1130°C endothermic reaction is not apparent on cooling.

The apparatus was installed in the hood, and measurements on PuO_{2-x} compositions should begin shortly.

NUCLEAR GRAPHITE - R. E. NightingaleNuclear Graphite Irradiations

The long-term irradiation program on nuclear-grade graphites continues to progress satisfactorily. The binary gas system in the H-3-12 capsule presently in the GETR is functioning properly and maximum sample temperatures have been maintained at $815^{\circ}\text{C} \pm 10^{\circ}\text{C}$ for the entire reactor cycle.

Two modifications made to obtain heat-transfer information (discussed last month) were only partially successful. The two graphite end collars at Position 6 lowered the sample temperatures only about half of that desired. It appeared that the contact resistance between the collar and the aluminum can was somewhat higher than originally calculated since the collars as manufactured were slightly under-size. However, the information obtained will provide basic data needed for capsule modifications required to cope with increased reactor power levels in the fall.

The second heat-transfer modification consists of a "telephone-dial" sample container above Position 1. The gamma heating and consequently the sample temperatures were lower than calculated. However, temperatures are higher than would be expected for cylindrical quarter-round assemblies in the same location. The temperatures are quite sensitive to changes in the composition of the binary gas system, indicating a wide range of temperature control is available. Since installation of the "telephone dial", information has been obtained which leads us to expect the carbon felt used as an insulator between the dial and the adjacent sample holder may disintegrate (see EBR-II discussion following). Consequently, the temperatures will probably drift down with time as the insulation deteriorates and the dial becomes better grounded to the sample holder.

Irradiations of "Dimensionally-Stable" Graphites

Capsule H-3-22 for the irradiation of "dimensionally-stable" graphites continues to operate successfully. The binary gas system is working very well and the maximum sample temperature has been maintained at 1000°C . Two thermocouples in the capsule failed early in the present cycle but later one recovered and is still operating. The other six thermocouples are functioning properly. The capsule is scheduled for discharge late in March.

Irradiations of Graphite in EBR-II

Investigations are continuing on capsule designs for conducting irradiation experiments in EBR-II. The basic sample, a cylinder 0.250 in. diameter by 1.75 in. long with spherical ends, will be stacked vertically in a 0.260 in. ID stainless steel tube with small graphite cups between each sample. This tube will be filled with argon and sealed. It will then be slipped into a 0.335 in. ID by 0.375 in. OD stainless steel tube which will be filled with argon and also sealed.

Studies were made on the use of carbon felt pads between the samples rather than graphite cups. A mockup tube was vibrated for about 1.5×10^6 cycles to simulate conditions in the reactor. Upon disassembly it was found some of the carbon felt had completely disintegrated and some had deteriorated badly. This allowed the sample ends to vibrate against each other. This pointed out that the felt should not be used in any situation where mechanical wear may be a problem.

Physics calculations have been started to determine the effect on fission rates and gamma heating rates caused by placing a capsule containing a moderator material (graphite) in the EBR-II fast-flux system.

High-Temperature Graphite Irradiations

Heat-transfer calculations and mockup tests have been completed for a capsule, H-3-12, for irradiation of graphites at temperatures up to 1600°C. The use of carbon felt as a thermal radiation barrier to raise sample temperatures in this capsule has also been analyzed experimentally and is discussed in a later section. Sample preparation and construction of "telephone-dial" sample holders is presently underway along with fabrication of a binary gas system for temperature control.

"Effect of Pre-Irradiation Annealing on Radiation-Induced Dimensional Changes of Graphite"

A report, BNWL-CC-501, on the above subject was issued. The main conclusion is that samples annealed at 1200°C appear to exhibit an anomalous behavior under subsequent neutron irradiation, whereas samples annealed at 600 and 1800°C show the same behavior as those not annealed. No explanation for the anomaly is presently apparent.

Controlled Density Graphite Series

Raw materials used in preparing this series were Continental coke particles (<0.015 in.) and coal-tar-pitch binder. The mix, containing about 12% binder carbon, was molded to form cylindrical billets 5 in.

diameter and 3 to 4 in. long. Subsequent processing included a normal bake, no impregnations, a chlorine purification and graphitization at 300°C. Densities were varied by molding pressure to yield 4 sets of samples with bulk densities of 1.87, 1.97, 2.05, and 2.10 g/cm³. Values of the coefficient of thermal expansion (measured between 25 and 425°C) in units of 10⁻⁶ in/in/°C ranged from 8.6 to 10.8 (I) and 1.8 to 0.4 (II) with increasing density. These measurements show that anisotropy of the samples increased with bulk density.

The samples have been irradiated to a maximum exposure of 8.6×10^{20} nvt ($E > 0.18$ MeV) at a temperature of ~600°C. All transverse samples grew and all parallel samples contracted after 2.5×10^{20} nvt. Both the growth and contraction increased with sample density. At 7.9×10^{20} nvt, the transverse samples had grown from 0.06% to 0.20% and the parallel samples contracted from 0.06% to 0.11% for the 1.87 and 2.10 density samples respectively. The coefficient of thermal expansion (CTE) was greater for lower density material for both orientations. At 5.3×10^{20} nvt the CTE of transverse samples with a density of 1.87 g/cm³ increased 13% and CTE for parallel samples increased 11%.

Isotropic Graphite Irradiations: R Series

Filler materials including Gilso carbon, fluid coke and carbon black were combined with 175°C melting point coal-tar pitch. The samples were then molded, baked, chlorine purified, and graphitized at 3000°C. No impregnants were used. The coefficients of thermal expansion between 25 and 425°C were 4.4 and 3.7×10^{-6} in/in/°C for the transverse and both parallel orientations respectively.

The samples have been irradiated to an exposure of 2×10^{20} nvt ($E > 0.18$ MeV) at a temperature of ~600°C. Each of the three orientations contracted 0.04%. Such a contraction at this low exposure indicates that isotropy as such does not insure dimensional stability in a graphite under neutron bombardment. The coefficients of thermal expansion increased by 4 to 9% at this exposure.

Fracture Mechanisms in Graphite

Selected areas of two cylindrical graphite samples were studied under tensile stress using a metallographic microscope. Observations on these areas were recorded photographically as the amount of stress was increased. Beginning at low levels of stress, cracks were formed that, in general, appeared to be parallel to the layer planes of the adjacent graphite crystallites. With increasing amounts of stress, these cracks became more distinct and new cracks were formed. In one case, where the stress had been localized by introducing a notch, failure occurred through the area being observed. The failure crack path preferentially followed cracks formed during tensile stressing. After failure had relieved the tensile stress on the observed areas, cracks formed in these areas did not all close up. Instead, a number of cracks remained which had not been present before stressing. Such a phenomenon could be responsible for the permanent set occurring in graphite.

Size-Effect Experiment

Results from the first post-irradiation length measurements on samples from several bars of the size-effect experiment indicate no definite correlation between magnitude of dimensional change and sample size for an exposure of approximately 4600 Mwd/At at 600°C. These results were expected, particularly for transverse orientations, in which the transition from radiation-induced growth to radiation-induced contraction results in very small length changes for exposures from 3000 to 5000 Mwd/At. This portion of the size-effect experiment is scheduled to be recharged into the irradiation test facility during the next reactor shutdown period.

The Effect of Oxidation on the Sonic Modulus of TSX Graphite

It has previously been pointed out that at temperatures up to 780°C, the rate of air oxidation is of minor importance in determining the rate at which the strength of TSX graphite, as measured by sonic modulus, decreases with increasing extent of oxidation. The dominant factors in controlling the modulus loss are the orientation and porosity of the samples.

It has now been determined that, at temperatures of 800°C and above, the air oxidation rate does affect the rate of modulus loss. Three parallel samples were alternately oxidized at a relatively low temperature, 650°C, where the utilization factor is near unity and at high temperatures of 800°C, 850°C, and 900°C, respectively. At 800°C, even after 15% oxidation, the utilization factor is somewhat less than 0.2, whereas at 850°C after 15% oxidation, the utilization factor is less than 0.1. These values indicate that oxidation is primarily confined to the external surface. In all three cases, logarithmic plots of the relative modulus, E/E_0 , versus the fractional residual weight, w/w_0 , showed a greater modulus loss with oxidation at the lower temperature than at the higher temperature where oxidation is confined to the outer surface. As expected, the average slope of these curves decreases as the upper oxidation temperature is increased, again because the oxidation is increasingly confined to the external surface and hence has a lessened effect on the modulus. This is in contrast to the earlier experiments which showed that below 780°C, the rate of modulus change with weight change was constant irrespective of the oxidation temperature.

Oxidation of Boronated Graphite

To determine whether or not the presence of B_2O_3 , which may be formed during the manufacture of boronated graphite, affects the rate of oxidation of this material, a sample containing a nominal 5 wt% boron was leached in boiling water before rate measurements were performed. Visual inspection during subsequent evacuation led to the discovery that as the water reached the outer surface of the sample, it rapidly evaporated leaving a white

deposit. As a result, the sample was again leached in boiling water, but upon re-evacuation only a negligible deposit was formed. After carefully removing this deposit from the surface, the sample was thoroughly outgassed and reweighed. The weight lost by the sample as a result of leaching amounted to only about 0.2 mg. It is not known whether the white B_2O_3 deposit was formed during the manufacture of the boronated graphite or by hydrolysis of the B_4C present. The latter process is thermodynamically feasible but may not occur at a significant rate because of the inaccessibility of the B_4C to the boiling water.

Subsequent oxidation rate measurements showed that the initial rate of the graphite - oxygen reaction was little affected by the preleaching. However, with continued oxidation the rate increased more rapidly but did not reach as high a maximum as was observed for a sample which was not preleached. This result apparently indicates that the rate of formation of B_2O_3 was altered by the preleaching which in turn alters the rate of the graphite - oxygen reaction. In the absence of B_4C , the rate of the graphite - oxygen reaction is unaffected by leaching. This was determined by "leaching" a sample of TSX graphite. Comparison of its oxidation rate to that for "unleached" TSX showed that within experimental error the two rates were the same in the temperature range 576 to 728°C.

The rate of oxidation of the preleached sample of boronated graphite as determined from the concentration of CO_2 in the effluent gas was greater than that determined from the weight loss, as expected, because of B_2O_3 formation. On the other hand, the reverse was found when this sample was again leached and its rate of oxidation measured. This result has been noted before (January 1966, Monthly Report) and, as before, remains unexplained. No carbon monoxide nor any other compound which could account for this discrepancy was found in the effluent gas.

IRRADIATION DAMAGE TO REACTOR METALS - A. L. Bement

Alloy Selection

Design was completed and working prints drawn for the ATRC specimen container with surrounding sleeve. The unit must duplicate, as nearly as possible, the neutron absorption and scattering within the reactor core caused by the specimen assembly. The container will be fabricated from a thermally stable nickel alloy such as Inconel 600. The sleeve will be fabricated from columbium simulating the ATR gas loop sleeve. Since initial ATR gas loop operating temperatures are tentatively planned to be less than 1800°F (980°C), nickel-base alloys will comprise the first loading. To this end, the first ATRC assembly will contain a typical loading of three nickel-base alloy groups which have high, medium, and low nickel content, respectively. The final ATRC design eliminates much detail and only retains the spatial arrangement and the equivalent cross sections between the mockup and the core assembly. Drawings of the ATRC components and assembly are being submitted to Phillips at NRTS for approval.

Thirty-two tensile specimens, recently discharged after a high temperature (740°C) irradiation, were tensile tested at room temperature and 300°C. Ex-reactor control testing is in progress to generate support data for the irradiation testing program.

Efforts to procure a quantity of AISI 316 stainless steel have continued. Close liaison has been maintained with Allegheny Ludlum to expedite the order placed for a low-boron heat of material which is expected to be on-site prior to the end of the fiscal year.

Irradiation Effects in Structural Materials

Stainless Steels

The purpose of this phase of the program is to determine the combined effects of irradiation and environment on the mechanical properties of stainless steels. Radiation-induced property changes will be determined from irradiations and tests conducted at various temperatures on several alloys. Particular emphasis will be placed on determining the existence of metallurgical instabilities and the mechanisms by which they are enhanced in a nuclear environment.

A liquid metal temperature-controlled irradiation capsule was exposed in the ETR core during Cycle 78. The capsule contained specimens of AISI 304 and 348 and operated at temperatures between 1050 and 1300°F (565 and 705°C) during the first half of the cycle. Malfunctioning of the capsule thermocouples was observed near the end of the cycle. The capsule was discharged at the end of the cycle and shipped to Radio-metallurgy for examination and recovery of the test specimens. The approximate exposure for this capsule was 2.5×10^{20} ($E > 1$ Mev).

Several tensile tests have been conducted on AISI 304, 348, and 406 stainless steels irradiated in a gaseous environment at 650°C to an exposure of approximately 1.1×10^{20} fast fluence. Tests were conducted between room temperature and 750°C. The results of all the data obtained to date on these materials and AISI 410 and AM-350 stainless steels will be presented at the ASTM Symposium on the Effects of Radiation on Structural Materials in June of this year.

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 preirradiation thermal and/or mechanical treatments and are screened by tensile tests, stress-rupture tests, and metallographic examinations.

Transmission microscopy of Hastelloy X-280 has identified the rod-shaped precipitate resulting from cold working and annealing (1500°F or 815°C) before irradiation to be of the same type caused by either irradiation or plastic strain subsequent to an anneal at 1500°F (815°C). This phase, $M_{23}C_6$, has a lattice parameter of 10.6 Å, and is essentially stoichiometric $Cr_{23}C_6$ (10.638 Å). Furthermore, it was determined that this phase, which was previously observed on edge (100 orientation) by optical and replica microscopy, has hexagonal morphology (110 orientation). Transmission microscopy has also identified the triangular precipitate induced by tensile strain at 1350°F (730°C) to be M_6C' with a lattice parameter of 10.85 Å.

Postirradiation stress-rupture data on Hastelloy for all the pre-irradiation treatments tested have been obtained. Specimens were tested in air at 1350°F (730°C) at a stress of 18,000 psi (the 1000 hr rupture stress for standard treated unirradiated material). Compared to standard treated material, improvements of up to 106% were found in rupture life, up to 100% in duration of secondary creep, and up to 37% in duration of tertiary creep. These improvements are due to a greater tolerance of the microstructure formed by preirradiation thermal-mechanical treatments for plastic strain after irradiation, which was also demonstrated by the improved necking strain and total elongation in short term tensile tests reported earlier. The stress-rupture data indicates that optimum rupture properties for material to be used in structural and cladding applications will result from an anneal at 1500°F (815°C) for 24 hrs prior to placement in the reactor.

Fracture Mechanics

Two irradiated DCB annealed Zircaloy-2 specimens, one of rolling direction orientation and one of transverse direction orientation, were tested following neutron exposure to approximately 6×10^{20} n/cm² ($E > 1$ Mev) in the ETR, G-7 loop. The fracture toughness (K_{Ic}) values for the TD orientation after this exposure were practically identical to those obtained for 3×10^{20} n/cm² exposure over the temperature range of -100 C to 80 C. Results for the RD orientation were similar; however, the toughness values for temperatures above 40 C were slightly lower for the higher neutron exposure. These results provide a strong indication of a saturation of the neutron damage effect on the fracture toughness of Zircaloy-2 at about 3×10^{20} n/cm². Verification of this tentative conclusion will be possible when the results of tests on specimens irradiated to 1×10^{21} n/cm² are available. These latter specimens are presently being tested.

Fracture testing of 20% and 40% cold worked Zircaloy-2 for both the TD and RD orientations has been completed. The results indicate that below -20 C, there is a slight decrease in toughness due to cold working. Above this temperature, the toughness of annealed material exceeds the upper limit for plane strain fracture. In the case of cold worked material, however, plane strain fracture occurs at temperatures as high as 100 C. In addition, a considerable difference in toughness between the TD and RD orientations develops due to cold working.

Tests to study the influence of hydride platelets on the fracture toughness of annealed Zircaloy-2 have commenced. Specimens (RD and TD orientations) charged with 100 ppm hydrogen revealed little effect of hydrides on fracture toughness at temperatures below -100 C. The toughness for both orientations decreases with increasing temperature, however, resulting in a considerably lower toughness for hydrided material than unhydrided at room temperature and above. This inverted dependence on temperature is in contrast to the behavior normally observed for Zircaloy-2. Interpretation of this anomalous behavior will rest on data to be obtained from specimens containing 400 ppm hydrogen.

As reported previously, the environment, whether it be humid air, dry argon, or vacuum, was found to dramatically affect the sub-critical crack growth rates in D2 tool steel. Tests were recently conducted in which three organic solvents were separately introduced into the crack. As the solvents benzene, acetone, and ethyl alcohol were applied to the crack tip, a dramatic increase in crack growth rate occurred. Ethyl alcohol had the largest effect causing the rate to increase by more than one order of magnitude. Further experiments are planned to determine if the activation energy for the crack growth process is affected by the addition of these and other organic liquids.

A paper entitled, "Applications of Fracture Mechanics in Evaluating the Initiation and Propagation of Brittle Fracture in Reactor Structural Components" was written and submitted for presentation at the ASTM

"Symposium on the Effects of Radiation on Structural Materials."

Advanced Metallurgy Studies

Fast Neutron Applications

An analysis has been made of the possible effects of fast neutron irradiation on fundamental hydrogen damage processes in fast reactor cladding and core structural materials. It is concluded that no significant increase in the normally minimal susceptibility of the austenitic stainless steels and nickel-base alloys to hydrogen damage should occur in a fast neutron environment.

The use of a cyclodial-focusing mass spectrometer has been obtained for the experimental phase of the neutronic gas behavior studies. Necessary repairs and modifications of the instrument have been initiated.

High Pressure Studies

The article, "Miniature Diamond Ultra-High Pressure Cell Adapted for In-Reactor Irradiation Studies," has been accepted for publication by the Review of Scientific Instruments. The above mentioned cell has since been adapted for ex-reactor, high pressure (700,000 psi) X-ray analysis. Designs have been completed for extending the working temperature of this nonhydrostatic device from -40°C to 300°C . Work is progressing on a hydrostatic in-reactor device capable of 450,000 psi for extended periods of time. The effect of pressure on atomic displacement energy and on the subsequent isochronal annealing of lattice damage is being investigated. It is anticipated that pressure will significantly increase the displacement energy.

Damage Mechanisms

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

During this period efforts were expanded to bring the resistivity apparatus to the desired level of sensitivity. Measurements with a reproducibility of 1 pt/10,000 were made on a 20 mil-iron hairpin sample. It is hoped these can be improved to 1 pt/500,000 in the next period. A resistivity ratio of 10/1 was measured for $\rho(295\text{K})/\rho(77\text{K})$ on this sample.

Activation volumes for iron purified in a ZrH_2 system show significant changes at carbon levels of 1 to 10 ppm. These effects are most obvious in the slope of the $\Delta\tau$ versus strain curves, where $\Delta\tau$ is the stress change accompanying a 10 fold increase in strain rate. Similar results

have been observed in the work of Stein. The origin of the effect is unknown but is obviously associated with the influence of impurities on dislocations. It is not apparent at this time whether this interaction is influencing dislocation motion or dislocation multiplication or is an indirect effect due to grain boundary segregation.

The testing of iron single crystals in bending has continued. Analysis of the slip traces produced on these crystals has shown slip systems that are consistent with systems reported in the literature. A capsule is currently being fabricated to irradiate several iron crystals to study any irradiation produced changes in their slip systems.

Etch pits on the (110) face in a single crystal of iron purchased from the Materials Research Corporation have been studied. The etch pit density in this crystal is lower than in crystals grown by the strain-anneal technique. Three distinct types of pits are formed: an elongated, oval pit; an elongated, hexagonal pit which appears dark under direct lighting; another elongated, hexagonal pit which is much lighter under the same lighting. The elongated axis of each type of pit is coincident with the [100] direction. The sides of the hexagonal pits line up with the [111] and [100] directions.

Attempts to produce observable movement of these pits with plastic deformation have been unsuccessful; however, an increase in pit density has been observed, possibly indicating some type of dislocation multiplication. At this time it cannot be said that the pits formed in this crystal are associated with dislocations. Further work is required to determine the nature of the pits and the differences in pit geometry.

Theoretical Determination of n, α Cross Sections

The need for more accurate cross section data for n, α reactions which cause high-temperature embrittlement in metals has become apparent. The rates of helium generation by (n, α) reactions which have been calculated for fast-reactor studies are uncertain by factors of 2 to 3 because of the lack of cross-section information. The fast-neutron cross section data for seven of the ten isotopes, which are primary sources of helium from fast-neutron transmutation, are difficult to obtain experimentally because the daughter nuclide is stable or emits no measurable gamma ray. Therefore, it appears that we must employ theoretical methods for the necessary cross section data. Those methods which appear to offer the best chance of success have been selected and work is now in progress to determine the fast (n, α) cross sections.

ATR Gas Loop Heater

The ATR gas loop heater is being designed to heat helium flowing at a rate of 4140 lb/hr to a temperature of 2100°F (1150°C). The heating is accomplished in two stages with electric resistance heating elements. The gas first passes through an annular ring of nickel-chromium tubes, makes a 180° turn, and then passes through a center core of molybdenum tubes. Both tube banks are supported with aluminum oxide ceramic. The heater internals are insulated from the outside conditions with metallic foil insulation.

The calculations were made to determine if the present heater design will meet EBASCO's specification M-3003, which defines the heater operating limits. The results state that the heater tube temperatures will stay below 2000°F (1093°C) for the nickel-chromium tubes and 3000°F (1649°C) for the molybdenum tubes provided the power split between the tube sections is adjusted for the inlet gas temperature. At the maximum flow rate of 4140 lb/hr and an inlet temperature of 1500°F (816°C), 44.4% of the power is provided by the nickel-chromium tubes while 55.6% is provided by the molybdenum tubes. At the low flow condition of 1800 lb/hr with an inlet temperature of 1300°F (704°C), 50% of the power is provided by each tube section.

Work is needed to define problems during heater transient operation. This will provide information on start-up and shut-down procedures, control requirements and short-term electrical power demands. This work is now being planned.

Model Gas Loop

The model gas loop is a 2000°F (1093°C), 500 lb/hr flow, helium recirculating loop that is approximately a one-tenth model of the ATR helium loop that is designed for installation on the Advanced Test Reactor (ATR) at Idaho Falls. The heater design represents a significant departure from the commercial practice, therefore, a similar type of two stage, compact heater is being built for the model loop.

The prototype gas heater has been installed in the model loop. Welding of the primary piping has been completed and all new welds have been x-rayed. The loop has been leak checked at 375 psig. A few minor leaks were repaired. All electrical power connections have been made. A protective cage is being constructed around the power leads from the transformers to the heater.

Instrumentation work progressed to a point where it was possible to turn on the power to the heater and check out the system. Power to each stage of the heater was turned on separately. Approximately 6000 watts or 10% of full power was applied to the nichrome elements and approximately 2500 watts or 4% of full power was applied to the molybdenum elements. Both stages operated satisfactorily while helium was recirculated. Upon completion of the cooling water jacket system, elevated temperature operation of the loop will be started.

A report on the design and construction of the model heater is being prepared.

ATR Gas Loop Support - Dynamic Materials Testing Apparatus

When superalloys are exposed to high-velocity helium in the DMTA, a sharp increase in evaporation rate takes place between 330 and 500 fps flow at 2050°F. It was suspected that a transition from laminar to turbulent flow might take place in the test section in this range of gas velocity to account for the observed increase in evaporation rates. A series of flow rate versus pressure drop studies were made on various loop components to determine if the flow- ΔP curves showed similar discontinuities.

A flow- ΔP curve for the gas cooler section of the loop immediately following the test section showed a definite increase in ΔP for flow rates greater than 50%. The preheater, in front of the test section, showed a similar, but not so pronounced, increase in ΔP in the same flow range. There was no discontinuity in the flow- ΔP curve for the test section with sample in place. The increased evaporation rates cannot be ascribed to a change in flow mechanism in the test section itself. The matter is still unresolved.

In-Peactor Corrosion of Zirconium Alloys

Hydrogen analyses were performed on samples of four zirconium alloys exposed in the ETP G-7 loop for 36 days at $\sim 280^\circ\text{C}$ in high-oxygen (~ 1 ppm) pH-10 LiOH. Oxidation data on the same specimens were previously reported (BNWL-207).

Hydrogen pickups, as a percent of the corrosion hydrogen generated, were 10% or less for Zr-4, Zr-3 Nb-1 Sn and low-nickel Zr-4. NPP Zircaloy-2 showed pickups of 13 to 26%; 6515 Zircaloy-2 showed pickups of 25 to 39%. The fractions of corrosion hydrogen absorbed were in general agreement for samples at in-flux and out-of-flux positions, except for Zr-2 specimens. Hydrogen pickup fractions were nearly a factor of 2 higher for 6515 Zr-2 specimens at in-flux positions. Attempts will be made to identify metallographically the reason for the accelerated hydrogen pickup on the 6515 plate.

There was no effect of prefilming on the in-flux oxidation of the alloys. However, for three of four etched/preautoclaved pairs of specimens, hydrogen pickup was higher for the preautoclaved specimen.

ATR GAS LOOP OPERATION AND MAINTENANCE · G. A. LastATR Gas Loop Heater

The ATR gas loop heater is being designed to heat helium flowing at a rate of 4140 lb/hr to a temperature of 2100°F (1150°C). The heating is accomplished in two stages with electric resistance heating elements. The gas first passes through an annular ring of nickel-chromium tubes, makes a 180° turn, and then passes through a center core of molybdenum tubes. Both tube banks are supported with aluminum oxide ceramic. The heater internals are insulated from the outside conditions with metallic foil insulation.

The calculations were made to determine if the present heater design will meet EBASCO's specification M-3003, which defines the heater operating limits. The results state that the heater tube temperatures will stay below 2000°F (1093°C) for the nickel-chromium tubes and 3000°F (1649°C) for the molybdenum tubes provided the power split between the tube sections is adjusted for the inlet gas temperature. At the maximum flow rate of 4140 lb/hr and an inlet temperature of 1500°F (816°C), 44.4% of the power is provided by the nickel-chromium tubes while 55.6% is provided by the molybdenum tubes. At the low flow condition of 1800 lb/hr with an inlet temperature of 1300°F (704°C), 50% of the power is provided by each tube section.

Work is needed to define problems during heater transient operation. This will provide information on start-up and shut-down procedures, control requirements and short-term electrical power demands. This work is now being planned.

Heater Development

An investigation of joining methods applicable to the molybdenum components of the heater continued. The present objective in this work is to determine if joining can be accomplished reliably by solid state diffusion without recrystallizing the molybdenum parts. This limits the processing temperature to a maximum of about 1000°C. Techniques which are known to produce solid state diffusion bonding at higher temperatures are being investigated for possible use at the comparatively low temperature of 1000°C. These methods include moly-to-moly joints with particular attention to surface preparation and atmosphere control, and the use of another element to initiate diffusion. In the latter case, titanium, whose effectiveness in this regard has been established, is currently being investigated. It is applied to the molybdenum surfaces as a vapor deposit by use of electron beam evaporation.

Uncoated tube and connector samples were cleaned, assembled and heated to 1000°C for 1 hour in a vacuum of 1×10^{-6} torr. A stainless steel plug is used in the bore of the tube to apply mechanical pressure on the joint during heating. No bonding occurred in this test. A similar test conducted on a tube sample coated with a 0.001 inch thick film of vapor deposited titanium showed metallurgical bonding around 30% of the area of the tube. Titanium diffusion had proceeded into both molybdenum surfaces over 100% of the mating surfaces, but had partially broken apart through the titanium film due to impurities in the film caused by the particular deposition cycle used. With a proper deposition cycle and thinner titanium films, high quality, 100% bonding should be obtainable.

Vendor Data Review

Contract submittals on the ATR Gas Loop through 3-15-66 have been reviewed and comments issued to EBASCO Services. Contract submittals included complete review of vendors data on items such as the Gas Purification System, Circulators, and Heat Exchangers.

Pre-Bid Conference

A pre-bid conference was held at Idaho Falls on the Analytical Instrumentation for the ATR Gas Loop. PNL explained the requirements of the Analytical System and answered technical questions from the vendors. The meeting clarified the intent of the specifications for measuring the impurities in the helium coolant.

METALLIC FUELS DEVELOPMENT - J. E. Minor

Irradiation of Thorium-Uranium Fuel Elements

The irradiation of two tubular Zircaloy-2 clad thorium - 2.5 wt% uranium, 1.0 wt% zirconium fuel elements continued successfully in the ETR-P7 loop. The fuel element having the highest exposure, 13,200 MWD/T, has shown a fuel volume increase of 2.1% and is currently operating at a maximum core temperature of 465°C . A second fuel element is operating at a maximum core temperature of 585°C and a surface flux of 9×10^5 BTU/hr/ft². The latter fuel element has accumulated 3000 MWD/T exposure and has shown a fuel volume increase of 0.5%.

One high exposure thorium alloy fuel element that suffered superficial damage by accidental dropping at the ETR was returned to PNL. This element had accumulated 11,850 MWD/T exposure (1.37 at.% burnup) and experienced 1.6 vol% fuel swelling. Radiometallurgy examination confirms some superficial damage to the closure weld. In all other respects the fuel element appeared to be in excellent condition with no evidence of corrosion, distortion, or surface filming. Measurements of the element show no change in the ID and an 0.007-inch increase in the OD. Detailed post-irradiation testing and evaluation of the element is planned.

Th-U-Zr Alloy Oxide Dissolution Studies

A series of exploratory dissolution tests have been run in the past month to determine if there are procedures that might be used for de-contamination following a Th-2.5% U-1% Zr alloy fuel failure. Oxides generated by the high-temperature water reaction with the Th-2.5% U-1% Zr alloy were dissolved completely in 85% H_3PO_4 (inhibited with ferric ion) at 155°C within 1 hour. Earlier laboratory studies have shown that it is possible to reduce H_3PO_4 corrosion on AISI 304 stainless steel to less than 0.5 mil/day at 155°C by using suitable amounts of ferric ion as an inhibitor. Dilution of the H_3PO_4 carrying the dissolved alloy oxide with large amounts of water yielded a gelatinous precipitate dissolvable in room-temperature 10M HNO_3 . Undesirable precipitation reactions in a loop might be avoided by following the H_3PO_4 treatment with a concentrated HNO_3 flush prior to the introduction or rinse water. Other studies are required if a procedure is required for the cleanup of a system containing both Zircaloy-2 and stainless steel. Phosphoric acid at 155°C causes an accelerated attack on Zircaloy-2.

Experiments with solutions of 10M HNO_3 -0.1M Na_2SO_4 -0.1M $\text{Na}_2\text{C}_2\text{O}_4$, 10M HNO_3 -0.1M Na_2SO_4 -0.1M Na_3PO_4 , 90 vol% concentrated H_2SO_4 -5 vol% concentrated HNO_3 -5 vol% concentrated H_3PO_4 and 95 vol% concentrated H_2SO_4 -5 vol% concentrated HNO_3 at elevated temperatures were not successful.

ENGINEERING DEVELOPMENT

NEUTRON FLUX MONITORS - G. C. Fullmer

Regenerating Detectors

The demonstration of the regenerative chamber concept, in which neutron absorptions in fertile material replace burnout of fissioned atoms, continues to be successful. Correct operation is still being achieved with the two experimental, regenerating (U^{235} - U^{235}) neutron flux detectors, which are installed in a test facility at the KW-Reactor. Plans are being made for fabrication of two advance-model chambers.

Beta-Current Generator Detectors

The problems of low signal to background ratios encountered in the B^{11} self-regenerating work continued to require laboratory development. One B^{11} detector was completely assembled and is presently being evacuated, during application of heat, in an attempt to improve the insulation resistance. Two quartz tubes are employed in the balanced (twin-lead) chamber assembly with one tube containing B^{11} and the other containing pure carbon. In addition, quartz components were fabricated for two more assemblies for later use. The most recent information indicates that the B^{11} thermal neutron cross-section is only about 5 millibarns, considerably less than previously anticipated, making successful operation more difficult to achieve. The selection of materials

for the balanced chamber assembly is critical in minimizing troubles from possible reaction products. Thus, the assembly case is fabricated from pure iron, and the weights of the B^{11} and pure carbon have been determined carefully on the basis of electron population (0.979 grams of B^{11} and 0.890 grams of carbon). These techniques are intended to achieve appropriate background current subtraction thereby permitting accurate measurement of the neutron-flux-caused signal current from the B^{11} portion of the assembly. A current in the order of 10^{-8} to 10^{-9} amperes is expected.

Microwave Detectors

Microwave frequency changes related to neutron flux level may be obtained from a cavity containing a neutron-absorbing gas, but the amount of frequency shift is excessively inhibited by use of quartz capsules in the cavity. Difficulties became apparent in the attempt to employ a single piece of dielectric material as a sealant over the coupling holes in a microwave cavity. This approach permitted energy coupling to occur from the boundaries of both dielectric and cavity, resulting in frequency-shift measurement problems due to reflections. Further experiments indicated that the use of two pieces of alumina, separated by a metal shield, should provide better operation. Various coupler configurations are being investigated to minimize possible adverse reflection effects at the detector.

CRITICAL FLOW AT HIGH TEMPERATURES AND PRESSURES - F. R. Zaloudek

Six blowdowns of a 1000-gal pressure vessel were performed to investigate critical flow of steam-water mixtures discharging from pipes. The test sections used had the following dimensions:

- 0.5 in. ID and 7 in. long
- 0.5 in. ID and 14 in. long
- 1.0 in. ID and 7 in. long.

Each test section had pressure taps along the full length to measure the pressure profile during critical flow conditions.

Experiments were performed with water heated in the tank to 500, 550, and 585°F (260, 290, and 310°C). The pressure in the vessel was maintained at 100 psi over the saturation pressure during the blowdown by introducing high pressure compressed air into the vessel. This was done to 1) maintain single-phase flow through the differential pressure flowmeter and 2) enable the study of steady-state flow conditions. A valve was provided upstream of the test section to introduce various restrictions in the flow path. The critical pressure was determined by extrapolating the pressure measured along the test section length to the discharge.

The results obtained from the 0.5 in. test sections 7 and 14 in. long agreed closely. This indicated the 7 in. long test section was sufficiently long to eliminate the effects of metastable flow conditions which are sometimes observed in short flow passages. The critical flow rates observed in the experiments on the 0.5 in. test sections were from 5 to 15% less than values predicted by the Fauske critical flow model. The data from the 1.0 in. ID test section are being analyzed, and the results are not as yet available.

UPSTREAM BOILING BURNOUT - D. R. Dickinson

Laboratory experiments to study boiling burnout occurring upstream of the exit location were continued with a 12 ft long electrically heated test section. One of the primary objectives of this series of tests is to obtain data for comparison with results from a 6 ft long test section run previously. This 12 ft long test section is an Inconel tube with water flowing through the 0.435 in. ID hole at a mass velocity of 3×10^6 lb/hr-ft² and a pressure of 1500 psig. Results to date show that with 6% inlet coolant quality (33% outlet), boiling burnout occurred over the downstream 2 ft. With 32% inlet quality (62% outlet), burnout occurred 3 ft from the upstream end. This is similar to the behavior in previous tests with the 6 ft test section. However, the burnout heat flux increased slightly with increasing inlet quality, whereas there had been a slight decrease with the 6 ft test section. Several tests remain to be run, and no final conclusions can be drawn yet.

MICROWAVE DETECTION OF COOLANT IMPURITIES AND MEASUREMENT OF IN-REACTOR TEMPERATURES - G. C. Fuller

Microwave Techniques

The frequency change expected in a microwave resonant cavity due to the moisture contained in a typical coolant gas is a direct function of both the absolute frequency and the change in dielectric constant of the gas. For a change from a pure gas to a mixture containing N_2 molecules of a second gas, the change in dielectric constant, $\Delta \epsilon$, equals $4\pi N_2 \Delta \alpha$ where $\Delta \alpha$ is the change in gas polarizability. Substituting the approximate value of $\Delta \alpha$ of 2.24×10^{-23} per molecule into the equation $\Delta f = f/2 \Delta \epsilon$ one obtains a sensitivity of 7000 parts per million water vapor per MHz change in frequency; this moisture concentration corresponds to about 25% relative humidity at room temperature.

Laboratory comparisons of dry air and air at 33% relative humidity indicated that shifting of approximately 1 MHz occurred. In order to measure environmental changes an order of magnitude finer with the desired reliability, further stabilization of the microwave generator will be required.

Infrared Techniques

Assembly of the infrared analyzer system, which incorporates the quadrant filter wheel concept, was essentially completed and laboratory testing initiated.

Testing and calibration of the variable pressure saturation apparatus for producing gasses with known moisture concentrations continued.

PLUTONIUM RECYCLE PROGRAM - F. G. Dawson

FUELS DEVELOPMENT

Fuel Flux Monitors

The design of the PRTR flux monitor for operation with the HPD fuel elements, Mark I-P₂, was approved for fabrication and assembly. All materials are available and fabrication can commence.

High Exposure Plutonium

Plutonium recovered from the first of three samples of Shippingport fuel, reported last month, will be available for beta-gamma measurements by April 1, 1966. A second sample has been dissolved and is being processed.

Plutonium Handling Practices

A paper entitled "Plutonium Handling and Control Practices at Pacific Northwest Laboratory" (BNWL-SA-525) was presented at the March 1-2, 1966, Plutonium Symposium at Washington, D. C. This symposium was sponsored jointly by the American Nuclear Society, Atomic Industrial Forum, and the Edison Electric Institute in cooperation with the Atomic Energy Commission.

A revision of the paper was prepared for publication in the proceedings of the symposium. A copy of the revised paper will be sent to the AEC Depository Library at the Gmelin Institute so that copies will be made available, on individual request, to other interested scientific libraries and organizations.

PRTR Fuel Fabrication

Fabrication of the final twelve 2 wt% PuO₂-UO₂ PRTR HPD fuel elements is in progress. Fuel for one-half of the rods has been loaded into Zircaloy tubes and these fuel rods have been x-rayed to determine pellet and fuel location and condition. Fabrication is on schedule to meet an estimated completion date of April 29, 1966. Fabrication was started of an additional 15 UO₂ fuel rods to be completed in early April, 1966.

Molten Fuel Defect Test

Initial results of the postirradiation examination of a defected (1/16-inch diameter hole) fuel rod revealed no rod distortion and some cladding discoloration downstream of the defect. The defected fuel rod, part of the seven-rod cluster GEH-12-39, had successfully operated in the P-7 loop in ETR for 29.7 effective-full-power-days. The test element was designed to operate at 26 Kw/ft with fuel molten to 55% of the radius. Postirradiation examination is continuing.

FERTF Test Element Design

A special FERTF test element is being designed for irradiation experiments at rod power generations up to 30 Kw/ft during the Batch Core Experiment. The basic design consists of a basket assembly containing a six-rod-ring of Mark I-R UO_2 - PuO_2 fuel rods. Several spacer grids along the length of the basket position and support the fuel rods while allowing expansion and contraction of the rods. Simplified underwater removal and replacement of fuel rods as well as a centrally located flux wire is an attractive feature of this design. Hardware design and development is continuing and a prototype basket with UO_2 rods is scheduled for flow tests during April 1966.

PRTR Fuel Element Performance

Postirradiation examination continued of fuel rods from FE-6504, which ruptured during irradiation in the FERTF.

Four fuel rods positioned in the outer ring of FE-6504 show evidence of localized overheating of the cladding on the outside surface of the rods at the same axial position (8-1/4 inches down from the square shoulder on the top end caps). Examination of one of the rods (FB-06) that exhibited localized overheating and which was positioned adjacent to the ruptured rod FR-4 was reported previously.

Postirradiation examination of rod FB-59, which was two positions distant from the ruptured rod, revealed phenomena similar to those previously observed in the top end region of rod FB-06, i.e., local collapse of the overheated cladding (beyond the Zircaloy transformation temperature of approximately 850°C) with sintering and grain growth of the fuel extending to its outer surface in the affected region. Diffusion between the depleted- UO_2 pellet and the UO_2 -4% PuO_2 powder made their interface indistinguishable in a photomacrograph. The interface is discernable, however, on the autoradiographs and indicates that the central

void formed and fuel melting occurred to within 0.05 inches of the pellet. Molten fuel flowed outward along radial cracks and may have contacted the cladding, without causing any observable core-cladding interaction. This has not been observed in any of the previously examined rods. Measurements indicate that the fuel column moved the depleted UO_2 pellet upward 0.5 to 0.6 inches against the downward force of the plenum spring. (Pellet movement also occurred in rod FB-06 which exhibited localized overheating in the top end region.)

Postirradiation measurements indicate a maximum diametral increase of 0.7% over an axial distance of approximately 20 inches in the lower portion of rod FB-59. There is some evidence that the postirradiation balance point is displaced in the downward direction, indicating that fuel relocation has occurred. In addition, examination of a transverse section 30 inches down from the top of the fuel rod, which is at the plane of the defect in the ruptured rod, indicates an unexpectedly large amount of fuel melting (to a maximum of 64% of the radius versus an expected 50%). The combination of fuel relocation and a greater than expected degree of melting may have caused the observed fuel rod swelling. More detailed analyses of these phenomena are in progress.

The high volumetric average fuel temperature that existed in this rod during irradiation is also reflected by the higher than normal fission gas release of 88.5%. Rod FB-55, which was two positions distant in the opposite direction from the ruptured rod and which exhibited a more severe locally overheated region is presently being examined.

Postirradiation stoichiometry of UO_2 -4 wt% PuO_2 fuel material contained in rods from FE-6504, determined by coulometric titration, are listed in the following table. The O:U ratios of the fuel from non-defected rods and the irradiated sample that was exposed in an out-of-reactor loop are essentially unchanged from pre-irradiation values. One of the irradiated samples (Item 4) was exposed to out-of-reactor test conditions that approximated the FERTF coolant chemistry, temperature and pressure during irradiation of FE-6504 prior to the rupture. The test conditions consisted of exposure to flowing lithiated water (pH 10.5) at 260°C for approximately 60 hours and included two thermal cycles. This treatment had apparently little effect upon the O:U ratio of the fuel. The analyses, however, indicate that the O:U ratio of the fuel in the ruptured rod increased to approximately 2.1 during irradiation. Such an increase could have a profound effect on the operating conditions of the fuel by decreasing the effective thermal conductivity and melting point. An analysis of these effects is in progress.

Rods From FE-6504	Postirradiation O:U
1. Ruptured Rod FR-4	2.101
	2.070
2. Non-defected Rod FB-83 Adjacent to Ruptured Rod FR-4	2.017
3. Non-defected Rod FR-6	2.003
4. Non-defected Rod FR-6 After Exposure in Ex-reactor Loop	2.007

A regression study was completed using data from two analytical laboratories to obtain a predictor of %-Pu in fuel elements.

MATERIALS DEVELOPMENT

Ceramic Fuel Dissolution Studies

Ceramic fuel dissolution work during the past month has been centered on (a) exploring the dissolution behavior of arc fused and Sol-Gel thoria in candidate decontamination reagents and (b) determining the corrosion behavior of stainless steel in concentrated acid solvents.

Experiments with arc fused and Sol-Gel thoria in 85% H_3PO_4 at 155°C produced satisfactory dissolution. Over 90% of the thoria was dissolved in 24 hours. Inhibition of the H_3PO_4 with ferric ion to minimize the corrosivity of the solution appeared to have little if any effect on its thoria dissolution efficiency. Earlier experiments showed that corrosion losses on AISI 304 stainless steel in 85% H_3PO_4 inhibited with ferric ion were less than 0.5 mil/day. Dilution of the H_3PO_4 carrying the dissolved thoria with large amounts of water yielded a gelatinous precipitate but this was dissolved in room temperature 10M HNO_3 .

Corrosion experiments with AISI 304 stainless steel in 90 vol% concentrated H_2SO_4 -5 vol% concentrated HNO_3 -5 vol% concentrated H_3PO_4 that were begun in February were continued during the past month. Steady state corrosion rates for stainless steel appear to be about 0.08 mil/day at 100°C and 170°C. Average corrosion losses in 24, 48, and 72 hours at 100°C were 0.12, 0.17, and 0.29 mils. Losses in like periods at 170°C were 0.28, 0.39 and 0.46 mils.

Corrosion and Water Quality Studies

H_3BO_3 Analysis

A simplified procedure for analyzing dilute H_3BO_3 solutions in water was evaluated. This procedure involves addition of d-mannitol to the solution to complex the weakly dissociated H_3BO_3 . H_3BO_3 complexed with d-mannitol can be titrated as a strong rather than a weak acid, and thus can be analyzed simply and accurately when boron concentrations are ≥ 100 ppm.

This effort was devoted to assessing the possibility of using electrical conductance measurements in the mannitol- H_3BO_3 solution as a direct indication of the H_3BO_3 concentration. Preliminary results confirmed that a direct relationship does indeed exist between conductance and H_3BO_3 concentration; however, the magnitude of the effect appears to be so small that achievement of highly sensitive and accurate measurements of trace boron concentrations by this method appears unlikely.

H_3BO_3 Removal by Ion Exchange

Removal of H_3BO_3 from dilute solutions by conventional strongly basic anion exchange resins is difficult due to the weak acid nature of H_3BO_3 . Recently however, a modified resin has been developed which reportedly selectively absorbs boron in the presence of other anions, and which is not subject to the strong dependence of exchange capacity on solution boron concentration as in the case of conventional resins. The properties of this material are being evaluated to determine whether it can be profitably employed to recover B^{10} from liquid wastes and/or from regenerant solutions used to treat spent ion exchangers at PRTR.

Preliminary results indicate that this resin is highly selective for boron in the presence of bicarbonate and hydroxide ions, and moderately so in the presence of chloride or sulfate. The boron capacity of the resin compares favorably with that of conventional ion exchange materials at moderate solution boron concentrations; however, results for dilute H_3BO_3 solutions are not yet complete.

Corrosion Evaluations in Citrate-Oxalic Acid Mixtures

Laboratory tests indicate that the presence of a magnetic field drastically effects the corrosion rate of carbon steel in citrate-oxalic acid mixtures (similar to the mixture used to decontaminate the PRTR primary system). The attack in a magnetic field is non-uniform; severe pitting and grain boundary attack were observed. This phenomenon was observed with coupons and corrosion probes.

The coupon in the magnetic field lost approximately one-half the weight of the control coupons (no magnetic field) however, the attack occurred over 10 to 15% of the coupon area. Further testing will be conducted to determine the effect on other ferrous and non-ferrous alloys to better define this phenomenon.

Inhibitor Evaluations for Citrate-Oxalic Acid Mixtures

Laboratory evaluations were conducted to evaluate selected inhibitors for carbon steel in citrate-oxalic acid mixtures (composition similar to the mixture used to successfully decontaminate the PRTR primary system). The inhibitors tested were propargyl alcohol, phenylthiourea, diethylthiourea and pertechnetate ion. Tests were conducted at 25, 60 and 80°C and inhibitor concentrations of 0.01M. Corrosion measurements were made with corrosion probes and coupons.

At low temperature (25°C) all the inhibitors were effective, however as the temperature increased the pertechnetate ion appeared to lose its inhibiting power. The other inhibitors remain protective to varying degrees. Phenylthiourea was the most effective followed by diethylthiourea and propargyl alcohol.

Chemical Shim Corrosion Studies

Pilot plant testing of a PRTR dump valve is being conducted to assure operation of the valve during use of chemical shim in the PRTR. The valve has continued to operate normally during three months testing in a moist helium atmosphere over a 160 ppm boron solution.

PRTR Corrosion Monitoring

Aluminum corrosion probes in the moderator and reflector as well as carbon steel probes in the primary shield coolant lines are being used to continuously monitor material corrosion in the PRTR. These probes indicated uniform corrosion rates of less than .01 mils/mo during the past month.

PRTR 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, and crack propagation tests at various temperatures. The respective objectives are to describe elastic behavior of an anisotropic tube, and to provide crack propagation data. This work is continuously improving the basis for judgment of irradiated specimens.

The components of stress and strain and the radial displacement of a pressurized orthotropic cylindrical tube depend upon six elastic constants. These constants can be calculated provided strain data from three separate tests are available. The three tests are: axial compression (or tension) of a tube, tensile test of a longitudinal specimen taken from the tube, and a pressure test of the tube under closed end conditions. Using appropriately selected specimens, the first two of this group of three tests have been performed. This work will provide an indication of the degree of anisotropy in PRTR Zircaloy-2 tubes.

REACTOR PHYSICS

D₂O Moderated Systems

Analysis of PuAl Burnup Data Initially Containing High Exposed Plutonium

The plutonium atom ratio data from 2 wt% PuAl fuel containing initially 16% Pu²⁴⁰ and the Pu²³⁹ atom density data, N⁴⁹, have been analyzed by fitting analytic functions to these data. The functions used are of the form

$$\frac{\text{Pu}^i}{\text{Pu}^{239}} = \sum_j A_j \frac{1}{N^{49} \times 10^6}^j,$$

$$i = 240, 241, 242.$$

Subsequent integration procedures utilizing the fitted functions to determine plutonium cross section ratios ($\hat{\sigma}_c^{41}/\hat{\sigma}_a^{49}$ for example) initiated further scrutiny of the basic Cs¹³⁷ data and α counting data. The reason is that one expects the ratio

$$\hat{\sigma}_c^{41}/\hat{\sigma}_a^{49} = \frac{\Delta N^{42}}{\int \frac{N^{41}}{N^{42}} dN^{49}}$$

to be essentially a constant (0.280 to 0.285) throughout burnup of the plutonium fuel. This is the case for the data except where the initial plutonium has been depleted to 40-50%. In this burnup region the $\hat{\sigma}_c^{41}/\hat{\sigma}_a^{49}$ ratio dropped to 0.256. Seven data points or seven samples are involved and by deleting one point, $\hat{\sigma}_c^{41}/\hat{\sigma}_a^{39}$ was increased to 0.27. The parameter that changed was DN^{42} and by recalculating, it was noted that Pu^{242}/Pu^{239} or $N^{49} \times 10^6$ would only need to increase by 2% (in the 40-50% depletion range) to increase $\hat{\sigma}_c^{41}/\hat{\sigma}_a^{49}$ to 0.285. A check of the basic data has shown that, in the region in question, the percent α counts attributed to Pu^{238} (a correction to the total α counts) for the seven samples are in question when all of the data available is plotted as a function of the Pu^{240}/Pu^{239} atom ratio. Three of the points indicate that the α counts due to Pu^{238} is 19-21% and another three indicate 21-24% or a 10-15% disagreement. The percent α counts due to Pu^{238} is used to correct the total α counts of a sample, e.g.,

$$\left(\frac{100 - \alpha\% Pu^{238}}{100} \right) \text{ total } \alpha \text{ count} = \text{corrected } \alpha \text{ count.}$$

The plutonium in a sample is then found using the value for the corrected α count. Due to the 10-15% disagreement in $\alpha\% Pu^{238}$, a 2-3% change in α corrected can result which in turn can propagate a 2-3% change to N^{49} , which will vary the results of the integration procedure. Two back-up samples are being analyzed to establish just how well the $\alpha\% Pu^{238}$ can be determined at this depletion range and to check the results obtained previously.

Correlation of PuAl Burnup Data

Correlation of analytical and experimental data for PuAl fuel containing initially 1.8 wt% Pu^{240} has been completed using Program S. Of primary importance in the correlation is the capture to fission cross section ratio

$$\left(\frac{\sigma_c}{\sigma_f} \right)_{239} = \alpha^{239}$$

of Pu^{239} . Since there is approximately 10% difference in α of Pu^{239} between the Leonard cross section data and the Sher cross section data, it was hoped the experiment would verify a particular set. The analytical calculations were performed using the Leonard cross section data. Analytical values of α for Pu^{239} agreed very well with the experimental values within the experimental uncertainty. A very good correlation was also obtained for values of the ratios

$$\frac{\sigma_c^{240}}{\sigma_a^{239}} \quad \text{and} \quad \frac{\sigma_a^{241}}{\sigma_a^{239}}$$

as a function of burnup.

Comparison of the theoretical values of the concentrations of the plutonium isotopes as function of burnup with the experimental values shows very good agreement for Pu^{239} and Pu^{240} over the 50% burnup range studied. The comparisons for Pu^{241} and Pu^{242} show good agreement up to about 30% burnup, which corresponds to the Pu^{240} being about 30% of the Pu^{239} . Above 30% burnup the analytical values of Pu^{241} differ from the experimental values by as much as 10%. The Pu^{242} values differ by as much as 15%. Since experimental uncertainties are quoted as about 3%, it is concluded that the difference is due to inadequacies of the theoretical model.

HPD Core Experiment in the PRTR

A number of calculations have been made in support of the fuels testing program in the high power density experiment in the Plutonium Recycle Test Reactor. In addition to those reported last month, one calculation was of PuO_2 enrichment as a function of density (85 to 95% TD) in pellet fuel to produce the same power as UO_2 -2 wt% PuO_2 vipac fuel rods (86% TD). A second calculation was of the change in flux shape over the core as a result of burn-up--in particular, the increase in flux in the Fuel Element Rupture Test Facility (FERTF) as a function of time.

Other calculations are in progress, including the effect of using a Hastelloy-C process tube in the FERTF. Planning of start-up and critical experiments is continuing.

PRCF Experiments with 2 wt% Mixed Oxide Fuel

Subcritical measurements using 19 rod clusters of PuO_2 - UO_2 fuel in a 13.86 in. hexagonal lattice were completed. The D_2O moderator contained ~ 5 wppm B^{10} during the loading and subcritical measurements. Nineteen clusters were loaded and multiplication data taken for extrapolation to critical. This loading was an estimated 60% of the critical number of fuel clusters. To obtain an undistorted radial flux traverse it was necessary to remove the safety rods from the reactor. The control rods had been removed at the start. Indium foils were attached to each of five clusters across the loading to obtain a radial neutron flux distribution measurement. The axial flux distribution was measured with a BF_3 counter. The analysis of subcritical measurements are about 20% completed.

The core was returned to an 8 in. lattice spacing using 13 clusters. Foils containing Pu^{239} , U^{235} and U^{238} were activated and counted. The foils were placed between the ends of separable rods in each of three rings of the 19 rod cluster in the core center.

Cadmium ratios were measured for the two types of uranium foils. Analysis of the foil counting data is in progress.

Experiments are beginning at higher concentrations of boron in the D_2O moderator. The core size will be increased to 28 clusters with sufficient B^{10} added to be critical at a moderator level between 75 and 80 inches.

Bowing of Fuel Rods in the FERTF

The computational methods used in analyzing the effect of bowing of fuel rods in the Fuel Element Rupture Test Facility (FERTF) have been evaluated using experiments conducted in the Plutonium Recycle Critical Facility (PRCF). The experiments used a 19-rod, UO_2 -4 wt% PuO_2 cluster in the center fuel channel, 509, of the PRCF. In one case the cluster was moved a distance of 0.027 in. from the center of the channel. In a second case a single rod was moved a distance of 0.051 in. toward the process tube. Flux distributions around the rods were measured with gold foils in both cases, as well as in the unshifted case. Reactivity effects of the two shifts were also determined.

Calculations of changes in flux due to shifts in rod positions have been performed using program THERMOS. Correlation factors have been derived to relate the results of the calculations to the experimental measurements. Using these correlation factors, calculations have been made for rod bowing in the FERTF for a 19-rod, UO_2 -4 wt% PuO_2 cluster in the FERTF as previously used; and for a 6-rod, UO_2 -6 wt% PuO_2 cluster in a stainless steel basket as proposed for future FERTF use.

Data - Theory Correlations

Two papers have been submitted for publication in the proceedings of the Conference on Neutron Cross Section Technology held at Washington, D.C. on March 22-24. The titles are Sensitivity of Reactor Multiplication Values to Cross Section Uncertainties of Thermal Systems and Sensitivity of Thermal Reactor Parameters to Scattering Model. A summary of the first of the above papers was presented orally at the Conference.

Theory Development

Grain Effects

A study of the effect of particle size on reactivity for PuO_2 grains in a natural uranium oxide rod and graphite lattice has been undertaken. Some necessary extensions of existing resonance collision theories are being made to account for moderation by the oxygen in the core, fast fission effects and possibly space dependence of escape probabilities. The effects of resonance shielding for grains ranging from 50 μ to 700 μ diameter is of prime importance, but does not appear to be very large, as the total PuO_2 content of interest is only 2%.

Automated Learning Logic

Theoretical Physics work on improvement of least squares fitting methods has led to discovery of a computer loop capable of learning (within limitations), much as the human mind learns, by sifting linear correlations from a sequential flow of observations. The essence of the idea here lies simply in discerning that least squares fitting methods do in fact provide a quantified representation of the fundamental logic of learning.

Practicality hinges critically upon exploitation of the additional observation that the coefficients of the normal linear least squares fitting equations accrue additively when imbedded in a data stream. Consequently, the minimum peripheral memory demand for storage of all accrued knowledge relevant to a given linear theory is determined solely by the complexity of the theory (and not by the voluminousness of the raw data).

Stripped of frills, a primitive learning loop consists essentially of four equations accruing normal coefficients, and arranged as follows to minimize loss of significant figures and risk of register overflow when traversed by an endless data stream:

Learning Loop (B-W Program Wit)

Measurement Count

$$m = m + 1$$

Mean Value·Value

$$\langle v.v \rangle = \langle v.v \rangle + [v.v - \langle v.v \rangle] / m$$

Mean Value·(Parametric Gradient)

$$\langle v.\partial v/\partial p \rangle = \langle v.\partial v/\partial p \rangle + [v.\partial v/\partial p - \langle v.\partial v/\partial p \rangle] / m$$

Mean (Parametric Gradient)(Parametric Gradient)

$$\langle (\partial v/\partial p)(\partial v/\partial p) \rangle = \langle (\partial v/\partial p)(\partial v/\partial p) \rangle + [(\partial v/\partial p)(\partial v/\partial p) - \langle (\partial v/\partial p)(\partial v/\partial p) \rangle] / m$$

Such a learning loop inevitably becomes dubbed as a "wit", to imply a primitive quantum of intellect not far above that of a computer "bit".

Calling for a progress report summarizing what has been learned leads to explicit construction of the linear theory by traverse of four more equations:

Theory Parameters

$$\underline{p} = \langle v \cdot \partial v / \partial \underline{p} \rangle \cdot \langle (\partial v / \partial \underline{p}) (\partial v / \partial \underline{p}) \rangle^{-1}$$

Measurement Variance

$$\sigma^2 = [\langle vv \rangle - \langle v \cdot \partial v / \partial \underline{p} \rangle \cdot \underline{p}] / (m-k)$$

Parameter Variance

$$\langle (\delta \underline{p}) (\delta \underline{p}) \rangle = \sigma^2 \langle (\partial v / \partial \underline{p}) (\partial v / \partial \underline{p}) \rangle^{-1}$$

Theory

$$v = \underline{p} \cdot \partial v / \partial \underline{p} \pm [(\partial v / \partial \underline{p}) \cdot \langle (\delta \underline{p}) (\delta \underline{p}) \rangle \cdot (\partial v / \partial \underline{p})]^{1/2}$$

The scope of the theory spanned by any specific study is controlled by providing the loop with the number of theory parameters k and with an equation for calculating the theory gradient $\partial v / \partial \underline{p}$.

Potential applications close at hand range from real-time processing of intricate reactor measurements into evaluated reactor theory, to continual optimization of intricate managerial decisions. Memory demand for storage of relevant accrued knowledge is so low that the loop can be programmed on even a small board-wired computer. The learning equations are so simple that on a large computer the loop can be attached to a stream of data as a trapping parasite without significantly affecting data processing time.

Code DevelopmentHRG*

A recent modification in the 1107 master version of HRG shows promise in preliminary tests of significantly improving the accuracy of broad group cross section calculations as they are usually done. The cases affected are those containing isotopes using the resonance integral calculation. Heretofore, the σ -effective for each resolved resonance was found by dividing the resonance integral for that resonance by the integral of a $1/E$ flux over a fine group. Implicit in this procedure are these assumptions about the integral of the fine group flux seen by the resonance absorber: 1) it does not depart appreciably from that of a $1/E$ flux (negligible energy self-shielding) and 2) it does not depart appreciably from that of the average flux in a heterogeneous system (negligible spatial self-shielding). The possibility has always existed of providing, as additional input, self shielding factors by nuclide and by fine or broad group to correct for situations in which these assumptions were invalid. In practice, this provision has been largely unused, possibly because of lack of knowledge of the appropriate factors. The recent modification redefines the σ -effective for each resolved resonance, using in the denominator the integral of the fine group flux seen by the resonance absorber instead of the integral of a $1/E$ flux. The fine group flux used in the denominator integral is determined in a manner compatible with that used in calculating the resonance integral in the numerator. In effect, the σ -effective thus determined already includes first order corrections for spatial and energy self-shielding. The self-shielding factors are still available as additional input and can be used for further correction, if desired.

This new modification is available on a trial basis only. It will not be included in the PHYSICS CHAIN version until it is more thoroughly evaluated.

BNW Master Library*

Fluorine-19 was added to the UNIVAC 1107 version of library as isotope number 408. The neutron cross sections and cards were supplied by the Critical Mass Laboratory. The data came from a publication by Buckingham et al, AWRE-O-28/60 (1961).

The g's or statistical weight factors of the Breit-Wigner single level resonance parameters for U^{232} and U^{234} were updated. The g's were in the library as 0.5 and should have been 1.0.

*Partially funded by O2.

U²³⁸ was updated to eliminate a small processing problem. The resonance parameter region used to begin at 5 eV and started out partially on the "peak" of a resonance instead of the 1/v tail. Since HRG can use resonance parameters in its calculations, the end result was one of adding the contribution in the region just below 5 eV twice, once as part of the resonance calculations, and secondly from the fit to the cross section data. The contribution problem was solved by reducing the lower energy of the resonance parameter region from 5 eV to 3 eV.

Sodium-23 was updated using the resonance parameters derived by Theoretical Physics personnel. The resonance parameters were calculated by using the computer program MISFIT-III to obtain a best fit to neutron cross section data with a theory similar to that of the B-W single level.

Nickel-58, 60, 61, 62, and 64 were updated for energies above .4 MeV using KFK-120 neutron cross section data for natural nickel. Using the natural nickel cross sections will cause a discontinuity in the data; however, isotopic information is not available in the fast region.

High-Precision Calculation of the Error Function and Its Complement on the UNIVAC 1107*

The error function

$$\text{Erf}(z) = \frac{2}{\sqrt{\pi}} \int_0^z e^{-t^2} dt$$

and its complement

$$\text{Erfc}(z) = \frac{2}{\sqrt{\pi}} \int_z^{\infty} e^{-t^2} dt$$

occur in expressions for the scattering kernel of an ideal gas, and elsewhere in thermalization theory. While the commonly-used Hastings approximations for these functions result in small absolute errors, the resulting relative errors may be as large as a few percent for values of z at which either function is quite small. In order to circumvent this problem, separate FORTRAN routines have been written to generate these functions (for real z) by using the Taylor series for $|z| < 1.51$ and a continued fraction expansion for $|z| > 1.51$. For $z > 4.17$, $\text{Erf}(z)$ is assigned the value 1.0, which is correct to eight figures. For $z > 9.2$, $\text{Erfc}(z) < 10^{-38}$, and it is assigned the value zero. The computing time for either function is approximately 10^{-3} seconds, and the maximum error observed during several test runs is two digits in the eighth decimal place.

*Partially funded by O2.

HFN

Computer code HFN has been submitted to the Argonne Code Center. Henceforth offsite requests for HFN should be referred to ANL.

HAMMER*

Work has begun on converting SRL-BNL computer lattice analysis system HAMMER to the UNIVAC 1107. HAMMER consists of a multigroup, multiregion, integral transport theory code for infinite lattice calculations, followed by a homogenized cell Fourier transform leakage calculation followed by a multiregion, one dimensional diffusion theory reactor calculation. Current plans include the generation of library tapes for HAMMER from the B-NW basic library.

SUBROUTINE INPUT*

The computer subroutine BCDRD was modified to allow users of the computer Program RBU to create a problem input tape using the standard Fortran system on the IBM 7090 computer. Previously the SOS system computer program PRE-INPUT was used for problem input to RBU.

The only major difference in treatment of BCD card input between PRE-INPUT and BCDRD is in the treatment of integers. Fortunately, BCDRD could be modified without a great deal of difficulty to allow integer input and the modified version was then named INPUT. INPUT will read any input deck used in the past RBU runs; however, INPUT contains all of the options available in BCDRD that allow the user to store repetitious input numbers in consecutive locations of memory.

UPDATE*

The computer program UPDATE has been modified to include in the output the date a record was updated on the B-NW Master Library tape.

The buffered tape copy routine contained an error in logic that caused three records to be copied whenever a copy request for a single record was made. The resulting update was flagged as containing an error and the new tape was not released for use until the problem was corrected.

The master library conversion routine BCDRD was modified to include several of the NAMELIST input options and they are: (1) Storage of "N" numbers by using input data of the form N*1.0; (2) Data field termination with a comma; and (3) When changes (1) and/or (2) are used, all 80 columns of the data card can be used for input. The conversion scheme was also changed and resulted in a 10% decrease in the time required to convert record two into binary form from the BCD card input.

*Partially funded by O2.

CONVRT*

The computer program CONVRT which is used to convert the B-NW Master Library from the Univac 1107 treatment of negative numbers to that of the IBM 7090 has been modified to include the date each record was updated as part of the output. If the record has been updated since the last publication date of the library, the record is flagged by printing "***NEW" at the end of the line of output. Users of the master library can now determine if the isotopes that they are going to use in a neutronics calculation have been updated and if so, they can use the new data in their calculations.

ABACUS-2*

The local optical potential model computer program ABACUS-2 is still not operating properly on the Univac-1107. As a test, ABACUS-2 was compiled on the IBM-7090 to see if the problem was a computational one; however, the 7090 run duplicated the test case exactly, and indicated a computer dependence. At this time the error in the Univac 1107 version has not been found; however, additional testing indicates that the radial wave equations are not being solved identically on both computers.

TRUNK

The TRUNK code to calculate the buckling of truncated spheres is complete and has been taped for customer use. A document is being issued.

RBU Code

The isotropic scattering portion of the RBU Monte Carlo was tested. All the results indicate that isotropic scattering collisions are handled correctly. Neutrons were started at a point with an energy of 10 Mev. and were slowed down to 1000 ev. The medium had a mean free path of 1 cm and a mass of 19 neutron masses. The average distance to collision, the average logarithmic energy decrement, the average cosine of the scattering angle, the neutron age, and the collision density were calculated and compared with the theoretical calculated values. All results were within statistical limits.

Neutron Physics

Slow-Neutron Scattering

The Rotating-Crystal Spectrometer was put into operation with a different monochromator crystal and rotor and modified chopper. The crystal rotor and chopper were tested for long runs at 9000 rpm and the phase stability was determined to be satisfactory. Other operational characteristics of the system were studied. At the end of the month additional slits were added to the chopper to attempt to improve the signal-to-noise ratio of the new system.

A survey measurement with an initial energy of 0.6 eV was performed to study the counting rates to be encountered in the excitation of the high-energy vibrational levels of H_2O .

Several measurements on the inelastic scattering of neutrons from H_2O were performed on the triple-axis spectrometer as part of the program to reduce systematic errors in previous measurements.

A powder diffraction study of a zirconium hydride sample was made with the triple-axis spectrometer. The observed diffraction pattern showed that the sample was not pure delta phase as hoped for but contained a component of epsilon phase. A more refined analysis of these results is required to establish the epsilon phase fraction more exactly and the suitability of the sample for further measurements.

Computer program SCS which calculates double- and single-differential scattering cross sections from point values of the scattering law has been adapted for the Univac 1107 system. Programs LEAP, ADDELT, and SCS have been modified to permit either independent or chain operation. Checkout of the chain operation is in progress.

Fast-Neutron Cross Sections

The PDP-8 computer for the Van de Graaff accelerator facility has been received by Radiological Physics. Specifications for the interface design necessary to use the PDP-8 with the fast-neutron time-of-flight system have been worked out with personnel of the Computers and Controls Unit. Design and construction of the interface circuitry is in progress by Computers and Controls.

The changes in the 1107 version of the total cross-section data reduction program BIG NED which include covariant errors have been completed. The running time per case has been reduced by about a factor of three primarily through the use of a largest-residual-pivot matrix-inversion routine instead of the Crout-with-iteration routine previously used. The calculated covariances are being studied.

The data reduction program BIG NED now includes a statistical test on the validity of the calculated cross section per channel which includes covariance. The results of this test are being studied and used to spot statistically inadmissible data in the input.

CAF Operation

Measurements were completed for 1.32 and 1.143 inch lattices using 2 wt% $\text{PuO}_2\text{-UO}_2$, 16% Pu-240 fuel and also 2 wt% $\text{PuO}_2\text{-UO}_2$, 24% Pu-240 fuel. The worths of a series of poison rods were measured in each of the four lattices.

TTR Operation

The TTR was operated one day for the Center for Graduate Study.

A series of reactivity measurements was made to investigate the possibility of determining B-10 concentration in D_2O . Preliminary data, in the range of 0 to 5 ppm, indicate a sensitivity of 1.3 cents/ppm B-10/liter of sample.

PRCF Operation

Exponential experiments were completed on the shim-free 13.86" lattice using 2 wt% mixed oxide fuel and 4.80 ppm B^{10} in D_2O . Axial traverses were made using a BF_3 chamber with the neutron source in and out of the core. An indium foil irradiation was made in the radial direction.

Fission foil irradiations were made in the High Power Density Lattice using the separable fuel element. Foils irradiated were bare Pu^{239} , U^{235} , depleted U and Cd covered depleted U.

The HPD critical test core size is being increased from 13 fuel elements at 4.80 ppm B^{10} to 28 fuel elements at about 12 ppm B^{10} for further physics tests. The present reactor state is 28 fuel elements at 10.4 ppm B^{10} with a critical moderator level 3.5 inches below the top of the fuel meat. Additional boron will be added to compensate for moderator level increases during moderator worth measurements and boron worth calibrations.

04 CYCLE ANALYSIS

Reactor Dynamics

A paper summary, "Short Term Dynamic Effects upon the Utilization of Plutonium as a Reactor Fuel," was submitted to and accepted by the American Nuclear Society for presentation at the June meeting. The paper will report details and the kind of results obtainable from the kinetic package which is being included into the ALTHAEA Fuel Cycle Analysis chain.

Code Development

Debugging is proceeding with the four-group one-dimensional version of the ALTHAEA code. To date two supporting "chain" programs PREP and CASE GENERATOR and the four-group one-dimensional routine collapsed to zero dimension are operable. The calculations appear correct but convergence to the proper flux is slower than desired. A third supporting code, SUPER CASE GENERATOR, has been coded, but a test case has not yet been tried.

Cross section libraries for use with the present systems on the 1107 and 7090 which include Cs target isotopes have been prepared.

Fuel Rejuvenation

Extension of the fuel rejuvenation cost calculations, to include "spiking" the fuel with U-235, were commenced.

Code Development

The 1107 version of the Fuel Cycle Analysis chain is now operable for two-group one-dimensional and four-group zero-dimensional burnups including the option for plotted output from the Bensen-Lehner plotter. Special parameters for inclusion on plots are now being calculated in the s prop subroutine of PLOTTER. The JASON code has been revised to give one-dimensional input suitable for problems in which a portion of the moderator is included in the fuel on the target regions with the rest of the moderator in a separate region.

Reduced Density Plutonium Enriched Thorium in Water Reactors

Thorium derives a large portion of its cross section from thermal reaction as opposed to U-238 which is predominately resonance reaction. (The thermal cross section for thorium is 7.4 barns, whereas U-238 is only 2.7 barns while the dilute resonance integral for thorium is 83 and U-238 is 282.) Thus effective

U-238 cross section which is a combination of these factors is much more responsive to neutron spectrum than is that of thorium. In fact, it is better to modify the conversion ratio of thorium fuel by removing thorium from the fuel element (reducing the effective density) than to modify the neutron spectrum by changing the moderator-to-fuel ratio. When thorium exist at full density in a water reactor (PWR, or BWR designed for U-238,) the enrichment requirement is usually too high to make this fueling method competitive with slightly enriched uranium fuel. However by reducing the thorium density, the enrichment requirement is reduced resulting in a fuel cycle cost reduction. This reduction in thorium density increases the fuel fabrication cost per kilogram. Thus if fuel fabrication costs \$100/kg at full density, it is reasonable to assume that the cost at 50% density would be \$200/kg. However, the fuel cost can be reduced by reducing the density even with this fabrication penalty. It should be noted that by reducing the density a useful matrix material may be introduced such as a good scatterer or good heat transfer material which is not evaluated here. The table shows the effect for various fabrication costs.

At the \$100/kg fabrication price a reduction to 6.6 g/cc (from 8.8 g/cc) can achieve a 0.15 mill/kWh_e fuel cost reduction. However, at \$25/kg fabrication cost, a gain of 0.20 mills/kWh_e is observed. For the 4.4 g/cc density an indication of even greater reduction is possible at still lower fabrication costs. The use of U-235 enrichment in reduced density thorium fuel is now being investigated.

PLUTONIUM-THORIUM FUEL CYCLE COSTS FOR VARIOUS FABRICATION COST

Fabrication Cost at Full Density (\$/kg)	Fuel Cycle Cost, Mills/kWh _e		
	$\rho = 8.8^*$ (g/cc)	$\rho = 6.6$ (g/cc)	$\rho = 4.4$ (g/cc)
100	2.150	2.007	2.108
75	2.052	1.901	1.948
50	1.949	1.793	1.797
25	1.838	1.690	1.642

* Density of thorium plus plutonium

FUELS REPROCESSING DEVELOPMENT

Recovery of Plutonium from High-Exposure Fuels

Two bundles of Shippingport Reactor Fuels (designated No. 6 and No. 14) have been dissolved. Plutonium has been recovered and partially decontaminated from both dissolver solutions by successive extraction extraction, scrub and strip cycles using trilauryl amine-Solvesso-150 as extractant. Principal contaminant of the recovered plutonium is uranium. It is expected that one further solvent extraction cycle with TBP-Soltrol as extractant will produce a plutonium product satisfactory for beta-gamma measurements. Plutonium recovery from Bundle 6 was about 95% and yield was 4.94 grams. Plutonium recovery from Bundle 14 was greater than 99% based on alpha count. Gravimetric data are not yet available. At present, the third bundle, No. 45, is being dissolved.

REACTOR ENGINEERING DEVELOPMENT

Plutonium Utilization in Power Reactors

The core heat transfer, pressure drop, and temperature limitation equations have been generated for computer programming.

A code to check steam table data against accepted values has been written with the intent of writing a steam table subroutine. This routine will accommodate the 1107 or 7090 computer and will lend itself to methods of easy programming. The code and the subroutine have been completed.

The present status of the study is that the program flow diagram is complete and equations pertaining to the above variables have been generated.

PRTR Rupture Loop Separator

The particle trajectory program has been modified for compatibility with the 1107 computer. Additional modifications to normalize the separator velocity profiles have been completed and are being verified. This modification will reduce input preparation time by about four hours per separator geometry.

Reactor Components Development

Chemical Shim

The dump valve test continued during the month. The valve has now operated a total of 90 days in an aqueous solution containing nominally 160 ppm boron. No detrimental corrosion is apparent.

On March 16, the shim rod was removed from the environmental test facility, and the shim well inspected with a borescope. Minor corrosion was observed. Accumulated test time at this inspection was 29 days. Nominal boron concentration was 50 ppm. Coupons were removed from the storage tank at the same time and inspected. No unpredicted results were noted. The shim rod and coupons were reinstalled and testing resumed.

Heat Transfer Calculations for the FERTF

Heat transfer calculations were made to determine the temperatures which might be expected in a pressure tube or a basket following the rupture of a molten core fuel element in the Fuel Element Rupture Test Facility (FERTF). Transient heat transfer calculations were made to determine the temperatures of a Zircaloy basket, a stainless steel basket, a Zircaloy pressure tube, and a Hastelloy "C" pressure tube might reach if they were contacted with molten UO_2 .

The model chosen for the calculations was a cylinder of molten UO_2 at 5500°F (3040°C), 1/2 in. in diameter and 0.3 in. thick protruding from a ruptured rod of the fuel element. It was assumed that the end of the cylinder would bridge the annular gap containing the coolant and would come to rest against the basket or pressure tube. It was assumed that no heat would flow into the UO_2 cylinder from the ruptured rod. Heat was assumed to flow from the circumference of the molten UO_2 to the coolant at a rate of $250 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. The heat transfer contact between the UO_2 and the metal plate was assumed to be perfect. For the basket cases, heat was assumed to flow from the UO_2 into the basket and then to the coolant from both surfaces of the basket. A heat transfer coefficient of $5000 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ was assumed for the heat transfer from the basket to the coolant. For the pressure tube cases, heat was transferred to the coolant only from the same metal face as was in contact with the molten UO_2 ; the opposite face was assumed to be adiabatic. The UO_2 was assumed to generate heat at a rate 50% greater than that of a rod operating at 30 kW/ft . The transient calculations were made using the TIGER V heat transfer code on the 1107 computer.

The analysis of the computer results is not complete, but tentative conclusions may be made.

The calculations gave the temperatures in the pressure tube or basket walls within the UO_2 cylinder as a function of time. The calculations showed that there was a very steep temperature gradient in the metal of the pressure tube or basket and that 3/16 in. away from the molten UO_2 , the metal temperatures were within a few tens of degrees Fahrenheit of the coolant temperature. The metal temperatures reported in the succeeding portions of the report will be the average temperatures through the metal directly adjacent to the UO_2 .

The calculations show that a Zircaloy-2 basket would reach a peak temperature of approximately 1480°F (804°C) about a second after the contact with the molten UO₂, and then would decrease within the next 2 to 3 sec to a near steady state value of 1100°F (593°C). At these temperatures, the basket is believed to have enough strength to retain the molten UO₂. A stainless steel basket would reach a peak temperature of about 1200°F (650°C) which would decrease to nearly a steady state value of 1000°F (540°C) within 6 to 8 sec.

A Zircaloy pressure tube would reach a temperature of about 1850°F (1010°C) about 3 sec after contact and would then level off to a temperature between 1800 and 1850°F (980°C and 1010°C). At these temperatures, a 3.25 in. ID Zircaloy-2 pressure tube at 1100 psia pressure would be expected to fail rapidly.

A Hastelloy "C" pressure tube would reach a temperature of about 1600°F (870°C), and would level off at that value. At this temperature, a Hastelloy pressure tube would be expected to remain intact for several hours.

These calculations contain many assumptions which will affect the conclusions. One assumption that has a large uncertainty factor is the calculational model which describes the failure of a fuel rod such that molten UO₂ contacts a basket or pressure tube in the manner described. These assumptions prevent placing a high level of confidence in the accuracy of the temperatures. However, if it is believed that the assumptions are reasonable, the calculations show that a Zircaloy pressure tube would fail, and a Hastelloy pressure tube probably would not fail if it were contacted with molten UO₂. A Zircaloy-2 basket and a stainless steel basket would probably retain the molten UO₂.

TEST REACTOR OPERATION

Operating Experience

The work in March continued to be directed toward those activities required for reactor startup with batch core loading. Replacement of the process tube-to-nozzle gaskets was completed and good progress made on the fabrication of the chemical shim system piping. Work continued on cleanup of the reactor with vacuuming and water flushes. Containment vessel decontamination continued to be a problem of holding contamination levels in various work areas to acceptable values. Progress during the month included the following:

1. The packing from three angle valves was removed to check for caustic content that might have been introduced during the primary system decontamination. No appreciable caustic residue and chloride were found. The packing on the remaining angle valves was tightened and additional packing added where required.

2. The bottom shield penetration coverplates for shim wells 1349, 1352, 1447, 1553, 1648, and 1651 were replaced in March. A lead-filled shielding block was added to the seven coverplates that were removed to reduce the radiation originating in these shim wells. Current lower access space radiation levels (at the inlet jumper elevation) range from 500 mr/hr at the angle valves to 1,800 mr/hr at the center of the reactor face.

3. Smearable contamination in the containment vessel continues to be controlled at relatively low values in all areas except the hot shop and process cell. A daily check is made of the containment vessel smearable contamination and the "clean" areas decontaminated when required.

4. The flush of the moderator inlet plenum area was made by introducing water into the outer plenum region, and pumping from the inner plenum region with one of the top and bottom shield pumps, and back through a filter to the moderator storage tank. The amount of fuel material removed with this flush was not determined, but contact radiation levels on the filter downstream of the top and bottom shield pump increased 4.8 R/hr during the flush.

5. Further efforts at cleaning the 1550 shroud tube and process channel were made in March. The lower shield plugs were removed from the process channel and discarded after cleaning fuel material from them was found impractical. The shroud tube and bottom primary shield penetration were cleaned with water and wire brush. Autoradiographs taken of the shroud tube opposite the process tube failure, before and after the cleaning, disclosed that contamination was reduced from an estimated two grams to one gram of fuel material. Further cleaning with a chemical decontaminating solution is planned.

6. Fifty-one process tube inspections were made during the month. Preliminary evaluation does not disclose any significant damage. Eighty-four process tubes were inspected in the tapered section. Process tube 5696 in channel 1556 and process tube 699 in 1946 showed an unknown circumferential mark at ~7" below the taper transition. Process tube 5696 will be removed and prepared for burst testing (exposure - 742 MWD) and to investigate the mark in the tapered section.

7. The storage basin cleanup program was successfully completed in March. A total of nine shipments of waste material from the storage basin were made to the "Wye" burial ground. Continuous flow is being maintained through the storage basin demineralizers (now being used only as filters) which keeps the water turbidity at low levels.

Maintenance Experience

Significant maintenance items accomplished were the completion of both top and bottom face hardware connections, repair of 7 shim recess bellows, and partial installation of the boron shim system. A total of 3,399 craft manhours were utilized during the March outage as follows:

FERTIF Fuel Failure	617
Decontamination	262
Repair	636
Modification and Improvement (mostly capital work)	1 234
Operations Assistance	169
Preventive Maintenance	481
	<u>3 399</u> manhours

Improvement Work StatusWork Completed

LAS Shielding (Shim Wells)

Work Partially Completed

Hydraulic Thermal Barrier Lift
 New Instrument Power Supply
 Traveling Flux Wire Probes
 Reactor Hall Crane Work Platform
 Moisture Detection System
 Effluent Monitor System Calibration Modifications
 Diesel Well Pump - Remote Start
 Calandria Measurements
 Instrumented Fuel Element Installation
 Core Blanket Filters
 Primary Pump Auxiliary Piping
 Moderator Filter
 Primary System Filters
 PRTR Critical Test Instrumentation

Design Work Completed

Instrumented Fuel Element Installation
 Core Blanket Filters
 Primary Pump Auxiliary Piping
 Moderator Filters
 Primary System Filters (Process Tube)
 Relocate IX-4

Design Work Partially Completed

Fuel Transfer System Improvements
 Analysis of HX-1 and Primary Pump Supports
 Reflector Change to Light Water
 Rupture Loop Decontamination Equipment

Process Engineering and Reactor Physics

The general plans for the critical tests have been prepared and the detailed test procedures completed.

A review of the trip recalibration data for the linear high flux trips revealed that there was little or no evidence of drifting trips since the completion of NPC-291-1 in July 1965. It is therefore planned that these trips be reduced from their normal 130% of indication to 115% of full power indication in order to reduce the severity of any transients. Under such conditions, a manual positive deviation trip will not be required.

PRTR Test 124 (Fission Gas Pressure Measurement in HPD Fuel Elements During Irradiation) was written during the month. Detailed calculations of radiation problems resulting from a gas leak were prepared.

Purchase of 5-micron size filters (instead of 2-micron size) to remove crud from the primary system was accepted in order to obtain the filters in time for use during the prefilming of the primary system. Minimum flow restriction is planned for the non-fueled process tubes containing these filters, to provide up to 100 gpm flow through each tube.

Technical Guidance - Project CBB-001 - Increased Power Level

Engineering work is essentially complete with all drawings having been approved. Some rework of the boron analysis system arrangement will be required when that system is delivered to meet specific reactor requirements.

Acceptance test procedures have been prepared and issued on the boron shim system (including removal, addition, and analysis subsystems), the HX-1 pressure control valve (S-80), MCC No. 5, MCC No. 12 EE, and the power calculator.

The boron shim system has been temporarily installed in the mockup area for conducting the ATPs. Installation of S-80 valve, part of phase I, is complete at month end.

A new containment vessel penetration was prefabricated and bench tested successfully at 15 psig.

Technical Guidance - Project CAH-119 - PRTR Storage Basin Addition and Experimental Facilities Modification

During the month the critical facility shop and offices and the operating supplies area were accepted for occupancy with minor exceptions.

PRTR Facility Additions

PRTR Reflector System

Process design change DC-273, which would permit operation of the reactor with light water reflector, was issued for approval. This change

includes isolation of the heavy water moderator system and the proposed light water reflector system, and provision for introduction of helium into the reflector vessel to sweep out hydrogen formed in the vessel during reactor operation. An appropriation request and equipment purchase requisitions were prepared.

Traveling Flux Wire Probes

The guide tubes on the top face have been completed and drawings have been prepared for an in-core neutron probe.

Review of PRTR Gas Loop for Gas Cooled Fast Reactor Use

A technical review of the PRTR Gas Loop was presented to representatives of General Atomics and ORNL. The GCFR fuel experiments were discussed.

EBWR DEMONSTRATION PROGRAM

Irradiation Testing of EBWR Prototype Fuel Rods

Three test capsules, fabricated with short segments of EBWR rods, continue to be irradiated in the MTR. The capsules contain vibrationally-compacted, pneumatically-impacted, UO_2 -1.5 wt% PuO_2 in Zircaloy cladding (0.064 cm thick). The maximum burnup attained in the fuel is 21,900 MWD/MT_{fuel} (5.4×10^{20} fissions/cm³). Goal burnups to 27,500 MWD/MT_{fuel} are planned. The three are the last of the irradiation test group of 32 capsules (bulk fuel densities, 77-85% TD).

Irradiation of production-run EBWR rods (bulk fuel density, 86-89% TD) continues. The rods are 1.07 cm OD by 148 cm long; including a plenum, 146 cm long. Of 58 rods, 13 have been discharged (one examined). Irradiation of Fuel Element No. 6501, a 21-rod bundle with a burnup to date of 0.7×10^{20} fissions/cm³, will continue when the PRTR resumes operation. The remaining rods are under irradiation in the form of three-rod clusters in another reactor. Goal burnups to 27,500 MWD/MT_{fuel} (6.8×10^{20} fissions/cm³) are scheduled for the rods. A data package for the cluster tests was received which includes daily information on the flux, coolant flow, coolant temperature drop, and reactor operating history. A cursory evaluation of the data discloses that the clusters operated initially at predicted linear power ratings and that the change in linear power with burnup agrees with the calculational model within ~6%.

EBWR Experiment

Reactor Burnup Calculations

A two-dimensional, two-group diffusion theory burnup code, EQUIPOISE BURNOUT, which recently has been made available has been evaluated using codes now in use. The results from one-dimensional reactivity calculation agrees within 4 mK with the results from HFN for the same calculation. The results of a zero-dimensional isotopic depletion calculation agree within one per cent with the results from a calculation using ALCHEMY for the same system, a mixed oxide fuel exposed to about 3000 Mwd/T. EQUIPOISE BURNOUT has provisions for varying the resonance cross sections as a function of nuclei density. The equation used is not adequate for obtaining the Pu^{240} cross section of the mixed oxide system mentioned above if cross sections calculated using HRG are assumed to be correct.

ADVANCED SYSTEMS

U-233 FUELING OF A FAST COMPACT REACTOR - J. C. Fox

A programming error was found in the core thermal hydraulics code which necessitated recalculation of the steady state performance of the alternate internally cooled fuels. Working graphs to construct pumping power, core void, coolant passage diameter and maximum temperatures have been plotted from the corrected data.

The pin core modifications to the thermal hydraulics code have been completed. A case has been programmed to compare the computer results with a previously proposed high-temperature thermal reactor before proceeding to the steady state comparison of the alternate pin fuels. Documentation of this effort has commenced.

The TIGER V transient heat transfer code is being evaluated for the purpose of determining the thermal reactivity feedback of the alternate fuels. It is hoped that this code can provide with reasonable programming effort and flexibility, the thermal hydraulics transient transfer functions for the nuclear kinetic comparison of the alternate cores.

The results of a first programming run has indicated that a 20% nodal mesh does not yield sufficient accuracy in temperature to calculate doppler coefficients and thermal strains with good confidence. Programming has begun on a fueled pin to determine the required mesh size which would achieve good confidence in the transfer functions.

Fuel inventories will be recomputed for a wide range of core materials compositions using the Russian 25-group cross section set, which appears to be most suitable for this study. For the previous study on the 710 reactor, the 15-group Hanson-Roach set was used. The size survey using this set is now in progress.

NUCLEAR SAFETY

CONTAINMENT SYSTEMS EXPERIMENT - J. M. Batch

Major Facilities and Equipment

Construction of the containment vessel ventilation systems was completed except for acceptance testing and installation of the air cooler. The drain piping system for the containment vessel, drywell and wetwells was also completed. Construction work started during the month on the aerosol sampling system, aerosol injection line, and the liquid sampling system. Completion of these systems is expected by the middle of May.

Design drawings of the Maypack cluster assemblies, the cluster and sensor support system, and the sight glasses for liquid level measurements were completed and approved for construction. Design efforts in progress include the heating system and insulation for heated air leakage tests, the electrical system for the solenoid valves of the Maypack clusters, and the Maypack filter holders.

Scope studies were started on a hydraulic dummy core and on a possible facility for installation of the reactor simulator vessel outside the containment vessel. An engineering study and report of the installed spray nozzle arrangement was completed. Alternate plans were reviewed to improve the fog spray efficiency.

Containment System Instrumentation

Specifications were prepared for CSE strain gage load cells and weldable encapsulated strain gages. Specifications for pressure transducers are now being written. The jet impact transducer was redesigned and the fabrication drawings altered to consolidate the SM-2 and CSE impact probes. This modification was made because of the high cost estimate (over \$800) for the original SM-2 impact assembly.

A readout circuit was developed and tested for the SM-2 prototype conductivity type liquid level probe. The circuit makes use of a 12.5 KHz astable multivibrator to excite the four spaced probes. The readout was arranged to incrementally "step" down in 0.4 volt increments as the water level passed each set of probes. This voltage will be satisfactory for use as an analogue readout during SM-2 tests, but plans are to redesign the circuitry to provide digital readout for the CSE reactor simulator conductivity probe. Some grounding problems remain to be resolved before the SM-2 conductivity probe and readout system can be installed on the SM-2 vessel.

Aerosol Development Studies

The effect on subsequent behavior of aerosols of the method used to inject them into a pressurized containment vessel was studied. Four tests, using three different injection techniques, were made. Although airborne behavior was nearly identical in these tests, material balances were poor, and minor differences were observed which suggest that further testing is necessary.

Aerosol behavior in tests where single-component aerosols were produced was compared with the behavior in Run 53 where a five-component mixture was generated. The results showed that the complexity of the aerosol composition did not significantly affect behavior in steam-air containment atmosphere. Tellurium, ruthenium, and barium oxide aerosols behaved similarly, being largely particulate, depositing on solid surfaces, and not washing off with continued steaming. Iodine and cesium also behaved similarly until organic iodine forms became predominant. These two materials were soluble in steam condensate, and after an initial period of rapid deposition on surfaces, they were washed off into the condensate.

Dummy Hydraulic Core

Work started during the month on scope and engineering requirements for a dummy hydraulic core for the reactor simulator vessel.

Multichannel Analyzer Code

A gamma energy analytical program has been written for the PDP-7 computer to economically process the large number of radiochemical analyses associated with the CSE. The generalized system consists of four shielded detector systems coupled with four analogue-to-digital converters, the outputs from which can be simultaneously stored by the computer. As written, the computer program can solve for a maximum of eight isotopes. Radionuclide concentrations are obtained via a least square fit of the unknown spectrum profile and a model profile related to the individual isotope standard spectra.

Debugging the multichannel analyzer codes continued. The matrix inversion routine was checked out. The details of how the results of a given isotopic analysis are to be reported on the teletypewriter were decided. Upon completion of the above, the coding of the statistical analysis section of the program and the report generation section were coded. A typical isotopic analysis report will look like the following:

ANALYZER #1

RUN NO.	SAMPLE ID	SAMPLE LOC	SAMPLE TIME
Al-235	M4F	AXZ	2360

NUCLIDE	COUNTS	REL STD ERROR	DPM
Ce-137	53235	.013	432.75
I-131	45729	.047	527.62

The meaning of the data reported is:

ANALYZER - Which analyzer is reporting

RUN NO. - A six-character run identification

SAMPLE ID - A three-character sample mnemonic

SAMPLE LOC - A three-character coordinate which locates position of sample in the vessel

SAMPLE TIME - Time after some reference time at which sample was collected

NUCLIDE - Isotopes for which analysis was made

COUNTS - Counts in sample resulting from nuclide

REL STD ERROR - Standard deviation of counts divided by counts

DPM - Disintegrations/minute for nuclide.

Containment Vessel Tests

The vessel was released for leak testing March 17, and pressurized to 1 psig. A 2 in. pipe cap had been left off the inner penetration panel; this was replaced, and the vessel was pressurized again to 8 psig, and two other significant leaks were located - an O-ring on the recently installed instrument flange, and a large butterfly valve which has been found to be faulty.

Heavy polypropylene bagging material is now being used to isolate leaks for measuring. Exposure to 270°F (150°C) metal surfaces shows this material to be superior to the polyvinyl material previously used, although it is not as pliable.

PRESSURE VESSEL CRACK MONITORING - J. C. SpannerDetection of Metal Overstress by Acoustic Emission

The purpose of this program is to develop a monitoring system to detect yielding and/or flaw propagation in reactor pressure piping by detecting the attendant acoustic emission.

Work to date has been concentrated on developing an understanding of the significant characteristics and propagation modes of the signals involved. Two recent tests performed with type 304-L stainless steel specimens have provided tangible and encouraging results. A 24 inch square by 3/8 inch thick plate with 3 inch by 4 inch tabs at the center of each edge was tested in a tensile machine by straining one tab to about 200% of yield. In another test, a strip 30 inches by 3 inches by 1/2 inch with a weakened section near one end was bent approximately 100° thereby straining the weakened section well beyond yield. In both cases, the acoustic emissions were monitored with three high sensitivity accelerometers located on the plate at various distances from the strain area. The output from these transducers was stored on magnetic tape for subsequent analysis.

In both tests, burst type emission was in evidence well before yield ($\approx 70\%$ of yield) and continued throughout the balance of the test. Analysis of these bursts indicates that the predominant frequency is about 50 kHz. In one test, the propagation velocity was calculated to be 100,000 inches per second and in the other 120,000 inches per second. This discrepancy may appear large until one considers that the maximum distance involved was only 23-1/2 inches which would tend to magnify analytical errors. Calculated velocities are near the theoretical shear wave velocity for this material (118,000 inches per second); however, better consistency is needed. The feasibility of locating the acoustic emission source by triangulation also looks promising.

The large plate exhibited what appears to be a steady high frequency pulse in addition to the burst signal. The amplitude follows a very logical pattern when related to the specimen loading sequence. However, the characteristics of this pulse have not yet been fully identified. Both of the above types of emission are consistent with earlier research investigations of acoustic emission in metals by others.

Thorough analysis of the above test data will be completed to form a basis for logical modification of the measurement system such as introduction of band pass filters to improve the signal-to-noise ratio. Analysis of the test results, in light of classical wave propagation theory and evaluation of specialized sensors, is being initiated.

Transducers have been installed to monitor a forthcoming hydrostatic test of the secondary shell and attached piping of a new N-Reactor heat exchanger. The primary objective of this test is to define signal conditioning requirements to determine the extent of the background noise problem under actual field conditions using high sensitivity sensors.

NUCLEAR FACILITY MONITORING GUIDE - I. C. Roberts

A manual of power reactor effluent and environmental monitoring requirements and practices is being prepared for the U. S. AEC Division of Reactor Development and Technology and is nearing completion. Some additional work is required to make the material relate to established practices of the U. S. AEC regulatory organizations. This additional work results from the circulation of a draft of the Guide at the end of last year. The additional work is, itself, now 50% complete. Final editing in preparation for printing the document began during the month.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENT - A. M. Platt - G. J. Alkire

Waste Solidification Engineering Prototypes (WSEP)

Installation of the WSEP is continuing on schedule. Approximately 90% of the work is complete. Coupling of instrument and electrical services is about 70% complete. Preliminary testing of instruments, furnaces and piping is well underway. Performance of the six-zone resistance-heated furnace for the receiving pot was satisfactory for heating and cooling.

Heat dissipation tests with waste shipping cask showed that greater than eight hours may be tolerated without artificial cooling of the waste solution. Greater than 50 kilowatts may be removed by the cooling coils.

Process Chemistry

Investigation of the melt system $PW-1:Na_2O:P_2O_5$ was continued. Data obtained were used to evaluate potential spray calciner-melter flowsheets for (1) carbon steel melt receivers assuming that solid additives would be added to the melter and (2) stainless steel melt receivers assuming no additives would be added to the melter. Melts which have slump temperatures suitable for use in carbon steel receivers were found to have the following drawbacks: they corrode carbon steel at about 25-50 mils/day at the slump temperature; they would require a specially prepared material as additive to the melter; the amount of additives required lower the PW-1 content of the product by a factor of two; the melts are 5 - 10 times more water soluble than melts which can be collected in stainless steel receivers. The data indicate that minor changes in the previously used PW-1 flowsheet (for use with stainless steel receivers) should permit holding the receiver pots at 50° C lower temperature during filling than with melt.

Corrosion tests showed that Corronel 230 is not a satisfactory melt pot material for holding PW-1: P_2O_5 melts in the 1000-1100° C range.

Glass Studies

Mathematical reduction and correlation of the data on the factorial study of the phosphate melt system is in progress (by Statistics). Meanwhile a statistically designed laboratory study of the mixed phosphate-silicate-borate system has been undertaken and is about 1/3 complete. With this system, a second liquid phase is formed in certain composition ranges, apparently as a consequence of the introduction of silica, and larger storage volumes are required to avoid phase splitting. Thus the use of silicate glasses, vice phosphate, for waste storage may involve an economic tradeoff between a cheaper storage container (carbon or low alloy steel containers in place of the more expensive stainless steel) and increased storage volume.

Most of the features of the new Thermal Analysis system have performed well at temperatures up to about 100° C, but have shown control instabilities at higher temperatures. In addition, the dilatometer furnace has shown a greater thermal gradient, from end-to-end, than desired. The manufacturer has been contacted regarding these deficiencies and their correction.

Additional details of the month's work are included in the January-March Waste Solidification Quarterly (A. M. Platt, Editor).

Wiped Film Evaporator

Two runs were made with the wiped film evaporator. Good performance was demonstrated in an eight-hour run while producing a highly viscous product. A second run of 40 hours was made with PW-2 feed to evaluate the system. Feed rates from 20 to 50 liters/hour and steam jacket pressures of 30 to 85 psig were tested satisfactorily.

Waste Container Measurements

Use of a 1/8-inch-diameter, diamond tip, core drill to prepare nonradioactive solidified waste samples for thermal conductivity measurements was demonstrated.

Waste Cask Monitoring

A new readout system was developed for use with capacitance probes to measure liquid level in waste casks. The integrated system, which uses capacitance to pulse height to DC conversion, was installed and calibrated in 324 Building.

Waste Densification

Study of pneumatic impaction of aluminum oxide as a means of densifying waste from the fluoride volatility process continued. Because of the eventual possibility of fission product volatilization, two 2-1/2 inch diameter cans were impacted at lower preheat temperatures at 1000° F (538° C) and 1500° F (816° C). Final core densities were 3.06 g/cc and 3.41 g/cc, respectively, compared with 3.56 g/cc when preheated at 2400° F (1316° C). Impaction conditions were identical, except for preheat temperature, in all cases. The effect on the resultant thermal conductivity is being investigated.

REACTOR SAFETY ANALYSIS AND EVALUATION - R. G. Wheeler

Subsize Vessel Failure Analysis

This portion of the program is concerned with evaluating the failure behavior of irradiated low alloy steel reactor pressure vessels from tests on subsize-vessel specimens. The subsize-vessel specimens are tubes of approximately 2-1/2 inches outside diameter, 1/4-inch wall thickness, and 18 inches long. The materials currently being tested are ASTM A106B and ASTM A212B. Fracture mechanics equations relating subsize-vessel specimen failure to fracture toughness (K_{IC} or K_{IC}) are being determined experimentally by compliance tests. These equations will also provide a means of obtaining fracture toughness data from subsize-vessel burst tests and will be used extensively for analyzing the failure of the irradiated A212B specimens.

During this period several adjustments were made in the compliance testing procedure and the specimen dimensions. The compliance testing of the subsize-vessel specimens requires an accurate measurement of the change in volume of the specimen with change in internal pressure. A problem with reproducibility caused by wall thickness variations was corrected by machining the pipe to a uniform wall thickness (± 0.002 inch). Techniques and equipment were developed for measuring the subsize-vessel specimens wall thickness, at any position on the vessel, to an accuracy of ± 0.0005 inch. The effective crack length of a subsize-vessel specimen is the ratio of actual crack length to specimen length. This ratio was changed, hence the number of specimens was reduced, by adjusting the separation of ring clamps on the specimen and thereby changing the specimen length. At the present time it is possible to obtain compliance data representative of five different crack lengths from tests on a single subsize-vessel specimen. The effect of variations in material properties, from specimen to specimen, on the compliance data is being reduced by compliance testing the specimens before and after a crack is placed in the specimen. Data variations caused by data reduction technique, temperature change, and atmospheric pressure change have been reduced by increasing the number of pressure-volume measurements per specimen, and by using computer programs to digest the pressure-volume data and curve fit the data. This increase in the number of data measurements and use of computer techniques will also result in a significant decrease in technician and engineer time.

During this period arrangements were made for the irradiation of two more A212B subsize-vessel specimens. These specimens will be inserted in the M3 position of the ETR during the next scheduled reactor cycle in April 1966.

Ultrasonic Monitoring of Burst Tests

Monitoring of the ultrasound emitted by a flaw as it approaches critical length has been performed on Al06B steel tubes as they were burst at low temperatures. Five tests have been monitored at temperatures ranging from -60° to -30°C . The tests were recorded on magnetic tape after suitable signal modification and are now being studied by replaying the tapes through an oscilloscope at various speeds. As more tests are recorded, attempts will be made to establish an ultrasonic failure pattern for Al06B.

COLUMBIA RIVER SEDIMENTATION STUDIES - J. M. Nielsen

Results of about 5000 analyses for radionuclides in 622 samples of Columbia River bottom sediment were turned over to the U. S. Geological Survey. Measurement of ion-exchange capacity in these samples was begun and initial values ranged from 2 to 20 meq./100 g. Suspended-sediment samples were found to have substantially higher ion-exchange capacities ranging from 32 to 37 meq./100 g., reflecting their smaller particle size and the presence of more organic matter. This latter quantity showed a significant seasonal variation as expected from the growth cycle of river micro-organisms and represented 20% of the dry weight for suspended sediments in May and October but only 7% in February. In bottom sediments, the amount of organic matter decreased with particle size from 3.6% for fine sediment in the McNary pool to 0.02% for coarse samples.

RADIOACTIVE SHIPMENT HAZARDS - L. C. Schwendiman

The fractional release rate of Cs^{137} and Ru^{106} from a pilot plant product of calcined high-level Purex waste was measured at $1100-1200^{\circ}\text{C}$. A sample of the product from a hot cell pilot plant run was pulverized and diluted with similarly prepared non-radioactive calcine to give samples containing about 1 mCi of total radioactive fission products. The release during primary fusion at 1200°C and the subsequent release rate during periods of holding for four hours at 1100°C and two hours at 1200°C were measured. The release of Cs^{137} during the initial fusion was 0.14% with the subsequent rate increasing from 0.6 to 1.2% per hour at 1100°C , and to about 1.7% per hour at 1200°C . These rates are somewhat higher than the rates obtained when Cs^{137} was added to a prepared simulated high level waste calcine.

About 1.5% of the Ru^{106} was released in the initial fusion, but subsequent heating at 1100°C released only 0.03 and 0.01% per hour for the first and second two-hour periods. The rate for the two hour period at 1200°C was 0.01% per hour. These rates, too, are somewhat higher than the limited data obtained earlier on the release of Ru^{106} added to prepared non-radioactive calcines and glasses.

FISSION PRODUCT AEROSOL CONTAINMENT - L. C. SchwendimanStudies of the Methyl Iodide-Hydrazine System

Kinetic investigations of the reaction between methyl iodide and hydrazine continued with an examination of gas chromatographic, potentiometric, conductometric and spectrophotometric analytical methods. Of these, the potentiometric method does not seem applicable at low concentrations of reactants. The conductometric method is more applicable when low concentrations of reactants are used, and the effect of salts and bases will be determined. Evidence for some catalytic decomposition of hydrazine on the platinum black electrodes was obtained. In the gas chromatography method the technique for sampling the reaction phase (gas, gas-liquid, and liquid) was investigated. A satisfactory procedure for the gas phase was found. A strong absorption of methyl iodide in the ultraviolet region permits the spectrophotometric investigations of the gas phase reaction over a range of methyl iodide partial pressures from 100 mm Hg to 0.6 mm Hg.

The kinetic data obtained by the nmr method during December for the reaction between methyl iodide and hydrazine was treated with a modified least-squares program according to three models: (a) a pseudo-first order mechanism, (b) one-step second order mechanism, and (c) two consecutive second order reactions such that a steady state condition would exist. These analyses indicate that the reaction rate appears to be first order in hydrazine and that the methyl iodide concentration follows a pseudo first order dependence.

The stability of vapor mixtures of 35% hydrazine hydrate and methyl iodide was demonstrated by generating an electric spark in atmospheres containing air, hydrazine vapor, and methyl iodide. Vapors of hydrazine hydrate in equilibrium with the liquid at 25, 75 and 100°C showed no evidence of reaction when an electric spark was passed through the vapors. Addition of a small quantity of methyl iodide into the system did not affect the results. Since the vapor compositions were even more susceptible to ignition than the concentrations anticipated in the future studies, it is almost certain that explosive combinations of the constituents will not be encountered in these studies.

Carbon dioxide may suppress the vapor phase reaction of methyl iodide and hydrazine as indicated in early aerosolization studies at Harvard. Methods to measure CO₂ in air were reviewed to help select one most promising.

Delays in making the large aerosol chamber available prompted preparing the 114-liter chamber used in scouting experiments for a series of experiments. Methods for handling and sampling methyl iodide will be perfected in these experiments which should also furnish further design data for the large scale experiments.

Reaction of Organic Compounds with I₂

A study of the interaction of iodine with paint has been initiated to gain an understanding of the mechanism and kinetics of alkyl iodide formation.

Since the interaction of iodine with organic compounds at room temperature has been reported to be very slow, the process could be accelerated with a radiation field. To test this, a vial containing natural gas (predominantly propane) at atmospheric pressure and I₂ was placed in the field of a Co⁶⁰ source. After 37 hours the mixture had received an integrated dose of 47.3×10^6 r. Traces of methyl and ethyl iodides and significant amounts of 2-propyl iodide were found in the mixture by gas chromatography. Mass spectrographic analysis verified the presence of the propyl iodide.

To determine which, if any, of the constituents of paint were responsible for the production of volatile organic iodides, Phenoline[®] thinner, Phenoline 300 pigment and catalyst, and Phenoline 302 pigment and catalyst were each placed in a Co⁶⁰ radiation field under an I₂ atmosphere. Using gas chromatography methyl iodide was observed only from samples of the irradiated Phenoline thinner. It was observed that the height of this peak increased linearly as a function of radiation dose. Some of the individual constituents of the thinner were also exposed to molecular I₂ in a radiation field. However, it was not possible to establish which of these, if any, were responsible for the production of methyl iodide.

The new electron capture detector for the gas chromatography permits detection of concentrations of methyl iodide on the order of a few ppm from micro liter size gas samples. With this detector methyl iodide, ethyl iodide, and what appears to be isopropyl iodide have been observed in samples of Phenoline thinner which were irradiated in the presence of I₂.

LOW AND INTERMEDIATE LEVEL WASTE STUDIES - G. J. Alkire

Various ferricyanide precipitates were studied to reduce the Cs concentrations from simulated waste solidification condensate. The precipitation of Cs as the ferricyanide was effected concurrently with mixed oxide (hydrous ferric oxide and hydrous MnO₂) scavenging. Some of the cations investigated for use as ferricyanide precipitates were Ce, Zn, Ni, Cu and Co. Removal of Cs was quite effective with all of these cations, although Ce, Zn and Ni appeared to be slightly superior. For each of the ferricyanides, Cs decontamination factors were > 1000. Radioruthenium decontamination factors in the same solution were about 100 at pH = 8, when the mixed oxides were concurrently precipitated with the ferricyanide.

The concurrent mixed oxide - ferricyanide precipitation process appears to be effective in substantially reducing fission product concentrations in a simulated waste solidification condensate. Experiments presently underway will attempt to determine optimum radionuclide removal for process conditions.

[®] Trade name of Carbolene Company, St. Louis.

Separation of sulfate ion from glass melter condensate by sodium sulfate crystallization was investigated. Low temperature crystallizations were conducted with simulated condensate which contains 4.4 M H_2SO_4 , 5.3 M HNO_3 , 0.18 M H_3PO_4 and small concentrations of metal ions to represent 0.1% entrainment and 2% ruthenium evolution from the phosphate glass melter. Radioactive tracers were added and the condensate partially neutralized with NaOH. Sodium sulfate ($\text{Na}_2\text{SO}_4 \cdot 10\text{H}_2\text{O}$) was crystallized at 0°C and -10°C . The crystals were separated from the supernatant liquid by filtration, washed with water, and the amounts of radioactive tracers remaining with the crystals were determined. Results indicated that only 2 - 5% of the Sr, Ce, Ru and Ce remains with the crystals. The bulk of the fission products may be recycled to the glass process with the supernatant liquid.

DISPOSAL OF REACTOR OFF-GAS INTO SOIL SYSTEMS - W. A. Haney

Work is underway on defining the soil system parameters which should be determined prior to an actual reactor off-gas injection test. This work is related to the exploration of a site, near the LOFT experiments at NRTS, for predisposal cold tests and ultimately an actual radioactive gas injection test using containment vessel gases. It is desirable to obtain as much information as possible about the characteristics of the subsurface environment (permeability, porosity, etc.), so that accurate results will be obtained from solution of a mathematical model of these injection experiments.

CUSTOMER

ASSISTANCE TO OAK RIDGE NATIONAL LABORATORY

Carbon Felt as a Thermal Radiation Barrier

Tests were conducted to determine the effectiveness of carbon felt in reducing thermal radiation. A heater suspended in a stainless steel tube was used as the radiation source. The heater was raised to the same temperature with and without carbon felt surrounding it, and the difference in heater power necessary to attain this temperature under both conditions indicated the carbon felt reduced thermal radiation by 90%. These results indicate use of carbon felt would be an effective means of elevating sample temperatures in irradiation capsules.

ASSISTANCE TO PHILLIPS PETROLEUM COMPANY - W. L. Hampson, Jr.

Four fuel element pins, loaded with 8 wt% enriched UO_2 pellets, were fabricated and delivered in time to meet a critical charging date in the MTR. Sixteen additional pins will be fabricated later. The pins will be used in the Contamination-Decontamination experiment at the National Reactor Testing Station (NRTS), Idaho Falls, Idaho.

UNITED STATES NAVAL RESEARCH LABORATORY

A spectral analysis has been completed of the Army SM-1A reactor. The purpose of this project was to provide a basis for correlating damage to the SM-1A pressure vessel and to steel samples irradiated above the reactor core with data from other radiation tests. These calculations are made to assist the United States Naval Research Laboratory in the pressure-vessel surveillance program.

The neutron spectrum and the damage calculations at the inner surface of the pressure vessel are based on a multi-energy, multi-region, one dimensional, transport-theory calculation. The reactor core including the pressure vessel was analyzed at the reactor peak flux plane using a 21-energy group structure with fine group structure in the fast-neutron range. Results show that although the spectrum at the inner edge of the pressure vessel is much softer than in core positions in test reactors such as the Low Intensity Test Reactor (LITR) and the Brookhaven Graphite Reactor (BGR), damage can be correlated between these diverse spectra within 15% if exposures are normalized to the integrated flux above 0.5 MeV, $\phi(E > 0.5 \text{ MeV})$.

Spectral determinations at the sample positions above the core (in line with the edge of the core and some 21 inches above core center) required a calculational method which permits a quasi 3-dimensional representation. Program DDK (which is also a multi-group, multi-energy transport-theory code) was initially selected, because of its capabilities of calculating neutron spectra in r,Z geometry; however, the computer time required to perform an analysis with this code in cylindrical geometry can be excessive. As an example a relatively small calculation with 11 energy groups, 4 radial zones, and 14 axial zones took nearly an hour of 7090 time for convergence; thus, the expense of calculating spectra for the SM-1A or a similar reactor could be prohibitive. Other codes are being studied and tested, since more studies requiring r, Z geometry are anticipated. Spatial representation of the SM-1A is being simplified to achieve accurate spectral determination with reasonable expenditure.

WESTINGHOUSE 25-ROD DNB STUDY

All fabrication necessary for Phase II of the test section to study DNB with a 25-rod bundle has been completed. The test section assembly should be finished during the first part of April, and installation in the vertical test section area will begin following the completion of the current experiments in the facility.

Phase II of the Westinghouse DNB study differs from the initial phase in that the test section has a variation in cross-sectional heat flux distribution. The 9 inner rods of the 25-rod square array have a flux higher than the outer 16 rods. The designed flux peaking ratio of inner rods to outer rods was 1.15 while resistance measurements of the assembled rods indicate the ratio will be 1.20. This difference falls within the allowable tolerance of the wall thickness of the tubes used for the heater rods.

Preparation of the components for the test section entailed both repair and new fabrication. All rods needed for the outer 16 positions were available from rebuildable rods of the first section. New rods were fabricated for the inner 9 locations. As before, all rod assemblies were hydrostatically tested for leaks at 3400 psig for a minimum of 30 min. No leaks were observed. Each of the 100 thermocouples (4 per rod) were checked after all rod fabrication; one thermocouple was found bad and the rod was replaced with a spare. The flow housing, which provides a square enclosure for the bundle, required removal of the remaining ceramic coating which caused trouble during Phase I. Sand blasting removed the coating; the blasted finish was removed by planing the plates.

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