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AEC
RESEARCH
and
DEVELOPMENT
REPORT

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

ON
DIVISION
OF
REACTOR DEVELOPMENT AND TECHNOLOGY PROGRAMS

MAY, 1966

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

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Nuclear Technology
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PACIFIC NORTHWEST LABORATORY
MONTHLY ACTIVITIES REPORT

APRIL 1966

on
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of
Reactor Development and Technology

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May, 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 major effort this month was in the direction of documenting various studies with the aim of publishing by July 1, 1966. In some cases this involves some additional computer work. The studies being documented include the following:

- An updating of the Plutonium Value document, HW-72217
- Fuel utilization studies (VESTA)
- Plutonium enrichment in reduced density fuels for various degrees of moderation
- Comparative fuel costs for fast breeders and fast converter.

Code development is nearing completion to provide a complete cycle analysis survey system for fast reactors. This system uses a four-group fast cross section appropriately collapsed for each loading from the various multigroup cross section by the Reactor Physics Department. A paper on crossed-progeny fuels was prepared for presentation at the Thorium Fuel Symposium at Gatlinburg, Tennessee, the week of May 1-5. A paper was accepted for the Denver ANS meeting in June 1966. This paper deals with the economics implications of reactor control when plutonium fuel cores are employed. Computer code capacity is being provided to examine several new isotope production schemes in Hanford reactors.

CONCEPTUAL REACTOR DESIGN STUDIES

Fuel cycle cost calculations for U^{235} fueling of an oxide fueled fast breeder were completed and neutronic calculations for a carbide fueled fast breeder are nearing completion.

USAEC-AECL COOPERATIVE PROGRAM

Debug work has continued on the analog simulation of the BLW-250 and is nearly complete. Simplification of the reactor kinetics model was made to facilitate debugging of the hydraulics model.

Analytical calculations and test section fabrication were continued in the study to determine cross-channel mixing during two-phase flow in rod bundle fuel elements. The final assembly of the test section representing a square-triangular flow channel was undertaken, and the precision welding of the heated portion was completed successfully.

Six zirconium alloys exposed in an ETR G-7 loop low oxygen environment show relatively little flux-induced oxidation, compared to considerable accelerated corrosion in previous high-oxygen exposures.

A 400 hr Zr-2.5 Nb pressure tube creep test was completed on a stress rupture specimen while it was exposed in a 280 °C circulating loop. The stress was changed after 400 hr from 44,000 to 54,000 psi. At the end of the second loop cycle, measured creep will be compared with creep indicated by the in-loop instrumentation.

APPLIED AND REACTOR PHYSICS

PLUTONIUM CRITICALITY STUDIES

A computational study of the criticality of plutonium compounds was continued; the effect of core density change on criticality of PuN-water mixtures was determined as a function of H/Pu atomic ratio.

The effect that finite particle size ($\sim 7.5 \mu$) can have on resonance capture in Pu^{240} and on the criticality of Pu^{239} is under study. A method has been devised for treating resonance capture in Pu^{240} for a homogeneous mixture of micro size PuO_2 particles and moderator.

HIGH TEMPERATURE REACTOR PHYSICS STUDIES

The construction work on the building and utilities for the High Temperature Lattice Test Reactor has been completed. The last Acceptance Test is being performed.

The detailed drawings of the safety blades have been revised slightly to incorporate sheets of rare-earth oxide-nickel cermet in place of slabs of B_4C in graphite.

The Programmed Measurement and Control System (PMACS) has not been accepted even though most of the malfunctions have been corrected. The remaining problem is a noise level which is higher than is acceptable.

A draft of the Operating Safety Limits has been approved by Battelle and submitted to the AEC, Richland Operations Office.

Specifications for high temperature experimental fuels are being studies. Information has been requested from several suppliers to ascertain availability of a wide range of materials in the form of microspheres.

Calculations have been initiated which are a preanalysis of the first experiments for the HTLTR. A new method for iterating to the critical condition for a reactor cell has been devised. The neutron leakage from the cell is the iteration parameter.

A computer program which provides a least squares fitting of data accruing in real time (called a learning loop) has been devised and is being applied to HTLTR operations needs.

PHOENIX FUEL REACTOR PROGRAM

A new procedure for splicing 2-D power maps to obtain a 3-D MTR power map is being investigated. The results are being used to estimate the expected overall power capability of the MTR-Phoenix Experiment and to assist in the design of flux suppressors.

Preliminary work has begun on the evaluation of the safeguards for the PRCF experiments in support of the MTR-Phoenix design.

Welding development was completed and a more economical method to produce the aluminum picture frames for the PRCF elements has been developed.

REACTOR FUELS AND MATERIALS

FAST FUELS OXIDES AND NITRIDES

Work was started on the preparation of a test proposal for irradiating candidate FTR driver fuel (SS-PuO₂ cermet and UO₂-PuO₂) in the Enrico Fermi Fast Breeder Reactor (EFFBR).

Preliminary thermal-diffusivity data were obtained on pneumatically impacted PuO_{2-x} at temperatures from 100 to 1500 °C.

Mass spectrometry and α -energy determinations of the plutonium isotopic distribution in samples removed from various radial positions in irradiated UN-20 wt% PuN fuels were begun.

Batches of 20 g of PuN and 40 g of UN which have well developed crystal structures and no oxide contamination (as indicated by X-ray diffraction analysis) are being routinely produced. Experiments on sinterability and homogenization of 80% UN-20% PuN mixtures are in progress.

BASIC SWELLING STUDIES

A capsule irradiated at 700 °C was discharged after reaching its goal exposure of 0.2 at. % burnup in the uranium specimens. Two additional capsules continue to operate successfully--one at 1000 psi and 450 °C, the other at low pressure (230 psi) and 525 °C. Another capsule is being charged for low pressure operation at 450 °C. The possibilities of obtaining multiple burnups in a single capsule by using specimens of different enrichments is being investigated.

A paper, "Controlled Temperature pressure Irradiation Capsules," was written for presentation at the International Symposium on Capsule Irradiation Experiments.

NONDESTRUCTIVE TESTING

Development of nondestructive inspection methods for detection of irradiation-induced shift in nil-ductility transition temperature in reactor materials continues to appear promising on the basis of recent experimental results. Special "critical angle" ultrasonic tests applied to irradiated A-212B and A-302B steel samples have given a "factor of six" variation in signal response compared to unirradiated control samples. While these results clearly distinguish the irradiated from nonirradiated samples, certain inconsistencies exist in the test results which are now being examined in further detail.

The eddy current multiparameter effort has in the past attempted to extract more useful information from test signals by analysis of their varying response as the test coil is moved past a defect area. Recently, attempts are being made to simplify this procedure by restricting the analysis to the defect signal obtained at its maximum value only. Timing, sampling and pulse stretching circuits have been breadboarded and tested for this purpose. Advantages of the new approach are not yet fully evaluated.

Investigation of miniature sinusoidal thermal wave transducers has shown that the response of sensors smaller than about $1.3 (\alpha/f)^{1/2}$, where α is thermal diffusivity and f is frequency, is highly dependent upon thermal diffusivity. These results suggest a novel method of measuring thermal diffusivity as well as detecting cracks oriented perpendicular to the sample surface. Work with larger transducers (3 in. square) has confirmed earlier observations that sensitivity is much improved as the transducers are increased in size.

NUCLEAR CERAMICS

Postirradiation metallography and autoradiography were completed for pneumatically impacted $\text{UO}_{1.99}$ and $\text{UO}_{2.05}$ specimens and their $\text{UO}_{2.00}$ standards. Examinations of specimens of $\text{UO}_{1.90}$ and $\text{UO}_{2.02}$ were completed last month, and a specimen of $\text{UO}_{2.15}$ is still being examined. The objective of this work is to correlate fuel-temperature profile with stoichiometry in the composition interval $1.90 \leq \text{O/U} \leq 2.15$.

Coulometric analyses of radial samples removed from an irradiated $\text{UO}_{2.00}$ specimen showed no significant variation in radial oxygen concentration. This agrees with earlier measurements on a sintered $\text{UO}_{2.00}$ specimen.

Measurements of grain-growth kinetics for pneumatically-impacted UO_2 as a function of stoichiometry were initiated. Experiments will be conducted in the temperature range 1200 to 1700 °C, and results will be correlated with irradiation behavior.

A B-type monoclinic phase, heretofore unreported for Am_2O_3 , was observed. Confirmation of these preliminary results by experiments in progress will show that Am_2O_3 , like Pm_2O_3 , exists in three forms: cubic, monoclinic, and hexagonal.

An oxide-susceptor induction furnace capable of high-temperature operation under large oxygen partial pressures was successfully developed. A ZrO_2 susceptor can be used continuously at 2050 °C; use of a ThO_2 susceptor should significantly increase the maximum operating temperature.

Changes in the optical absorption spectra of single-crystal ThO_2 were observed after heating in hydrogen or air.

A chemical method was worked out to determine the solubility of PuO_2 in sodium silicate glasses from which the controlled nucleation and growth of PuO_2 single crystals will be subsequently investigated.

Complete 2.5 and 4.0 in. diameter die sets and a hold down plate were obtained from an offsite vendor for less than \$2,000.

Several tests were run on the new vacuum system to determine the efficiency of different size evacuation tubes that are being used on the pneumatic impaction billets.

Mechanical properties of SS cermet fuel pins were determined for pins of varying UO_2 volume percents.

NUCLEAR GRAPHITE

The present capsule H-3-12 is functioning properly after 1-1/2 cycles of reactor operation and the binary-gas system is maintaining maximum samples temperatures at 815 ± 10 °C.

Capsule H-3-22 was successfully disassembled the week of March 21. Eighty-four samples were recovered and 58 have been reinstalled in H-3-23 for reirradiation. The new capsule will also contain four "telephone dial" sample holders to provide information on in-reactor heat-transfer characteristics for this geometry.

Samples of Texas-coke graphite heat treated at different temperatures were irradiated to 16,200 MWd/adjacent ton at about 600 °C. The samples with high heat-treatment temperatures displayed smaller thermal expansion. Samples heat treated at 2570 and 2700 °C displayed a decrease of CTE upon irradiation. The samples heat treated to 1370 °C, which had first contracted and then expanded, displayed an increase in CTE of approximately 300%.

A new method of calculating the thermal expansion of polycrystalline graphite has been developed. The results from the application of the theory are in substantially better agreement with experimental data than previous theoretical analyses.

IRRADIATION DAMAGE TO REACTOR METALS

A liquid-metal, temperature-controlled capsule, designed for irradiating fast reactor structural materials in the core of the ETR, was examined following a one-cycle irradiation. The temperature sensing system malfunctioned midway in the irradiation cycle, and the cause was traced to a thermocouple cladding failure resulting from thermal fatigue.

An in-reactor creep test on annealed 304 SS to be conducted at 840 °C 5,000 psi is accumulating irradiation prior to test loading. The out-of-reactor control creep test at these conditions ruptured after 926 hr and exhibited a total elongation of 26% at rupture. Minimum creep rate during the test was 1.76×10^{-4} in./in./hr.

Out-of-reactor tests at 900 °C, 5,000 psi and 700 °C, 10,000 psi ruptured after 96 and 10,630 hr, respectively. Rupture strain of both specimens was about 28%. The specimens were submitted for metallography.

Tensile tests have been conducted at deformation temperatures between room temperature and 750 °C on AISI 304 and 348 SS irradiated at 650 and 750 °C in a gaseous environment. Exposures at these temperatures to about 1.0×10^{20} nvt (>1 MeV) cause little change in the strength at any deformation temperature examined for either material. The greatest reduction in ductility occurs at deformation temperatures between 500 and 750 °C. No significant decreases in ductility were noted in the range of test temperatures between 200 and 400 °C as were noted when these same materials were irradiated at temperatures below 400 °C. Annealing studies are being conducted on both materials for each irradiation temperature examined.

Transmission microscopy was performed on an irradiated Hastelloy X-280 tensile specimen that was tested at 1350 °F (730 °C). Before

irradiation the specimen was given 20% cold work and a 1000 °F (540 °C) anneal. The specimen shows less stain-induced precipitation than the unirradiated control specimen.

The fracture toughness of annealed Zircaloy-2 irradiated to an exposure of 1×10^{21} n/cm² was determined using the DCB specimen. The results of these tests together with data for lower exposures previously reported indicated that saturation of neutron damage at 280 °C is complete in Zircaloy-2 at an exposure of approximately 3×10^{20} n/cm². Fracture toughness of cold-worked Zircaloy-2 annealed at 475 °C was almost completely recovered after 100 min. At longer annealing times, however, the fracture toughness exhibited a maximum and decreased again slightly.

A joint experiment has been initiated between Atomics International (AI) and Battelle-Northwest to test methods of neutron-spectra determination and fast-neutron dosimetry. This experiment is intended to test and improve the dosimetry and damage-correlation methods used in material irradiations. The foil activation measurements have been completed at all sites and the data are presently being analyzed. Preliminary results indicate that there is good correlation between calculated and measured foil-activation data.

ATR GAS LOOP OPERATION AND MAINTENANCE

Testing of the model heater for the PNL Gas Loop is in progress. A total of 67 hr operation under conditions up to full 500 lb/hr gas flow at 2000 °F outlet temperature has been achieved.

METALLIC FUELS DEVELOPMENT

Six thorium-uranium fuel elements are under irradiation with the highest exposure achieved being 13,400 MWd/ton. This fuel element still shows no volume increase greater than that required to accommodate solid fission product atoms.

A chemonuclear fuel element with a web thickness of 0.00012 in. is being fabricated. Currently a device is being completed to rotate 48 rods in the electron beam vacuum depositor and to permit all 48 rods to be simultaneously coated.

ENGINEERING DEVELOPMENT

UPSTREAM BOILING BURNOUT

Laboratory experiments were continued on a 12 ft long uniformly heat test section to investigate the occurrence of boiling burnout at locations upstream of the test section outlet. In these experiments, the location of boiling burnout was observed to move progressively from the outlet to a point 3 ft from the inlet as the inlet quality was increased to 30%. Also, tests were performed where the heat flux was increased considerably over the burnout values. In these tests, the region of burnout spread over most of the length of the test section, and very large variations in circumferential temperature were noted.

NEUTRON FLUX MONITORING

Continued operational success of the regenerating neutron flux monitor concept is being demonstrated by the two experimental (U²³⁴-U²³⁵) detectors installed in KW Reactor. A prototype chamber based on the self generation of beta current from B¹¹ absorption of thermal neutrons has also been installed in a KW test facility, but results to date have been only moderately encouraging.

CRITICAL FLOW AT HIGH TEMPERATURES AND PRESSURES

Pressure vessel blowdown experiments were performed to investigate the critical flow in a 3 ft long section of 2 in. pipe. During the blowdowns, discharge rates up to 4300 lb/min were obtained using water of 505 Btu/lb enthalpy. The results agreed well with earlier results obtained with smaller test sections.

PLUTONIUM RECYCLE PROGRAM

FUELS DEVELOPMENT

Postirradiation examination of fuel rods from ruptured FE-6504 is essentially complete. The examination showed that four nondefected rods in the outer ring experienced localized overheating in the top end region. Fission gas release from the six rods sampled ranges from 75 to 88% and correlates well with the indicated fuel temperatures during irradiation.

Postirradiation examination of a transverse cross-section of an irradiated, intentionally defected UO₂ fuel rod indicated that the fuel was molten to approximately 20% of the radius during irradiation.

A prototype FERTF fuel element was fabricated and is scheduled for a series of physics, thermal hydraulics, and mechanical tests.

Volume I of the study "Comparative Costs of Oxide Fuel Elements" has been completed.

All High Power Density Batch Core fuel requirements have been met on schedule.

The equipment for rounding PuO₂ particles has been installed in a hood having a dry air atmosphere.

FUELS REPROCESSING DEVELOPMENT

Plutonium has been recovered from three samples of Shippingport Reactor Blanket fuels, decontaminated and sent to Dosimetry Technology for beta-gamma emission measurements.

REACTOR PHYSICS

Critical experiments have been conducted in the Plutonium Recycle Critical Facility using 2 wt% PuO₂-UO₂ fuel elements and moderated with D₂O containing 11 wppm B¹⁰.

Calculations have been made in support of the Fuel Element Rupture Test Facility of the Plutonium Recycle Test Reactor (PRTR).

The feasibility of using the Thermal Test Reactor for measurements of the boron content in D_2O samples taken from the moderator of the PRTR was demonstrated.

A series of experiments in the Critical Approach Facility, using 2 wt% PuO_2 - UO_2 fuel elements and unpoisoned H_2O moderator have been completed.

A paper describing measurements and calculations of the worth of neutron absorbing rods in Pu-fueled, H_2O lattices and one describing measurements and analytical correlations of critical mass, buckling, and reflector savings determinations have been accepted by the American Nuclear Society for presentation at the June meeting.

Spectral indices T and r , which characterize respectively the thermal component and the epithermal component of the neutron spectrum, were measured in the PRCF in a cell consisting of a Pu-Al 19-rod cluster moderated with D_2O . The indices were deduced from gamma-ray activities of lutetium foils. Similar measurements were made earlier in the PRTR.

Analysis of calculations of burnup of a mixed oxide fuel in various light water environments has continued.

The complication introduced by the moderation of neutrons by oxygen in PuO_2 - UO_2 rods is being studied to assist in theoretically predicting the effect of particle size on reactivity in such rods.

Input requirements of the NIT code, which calculates resonance integrals by the Nordheim method, have been simplified by modifying the code to read resonance parameters from an HRG data tape.

Updating of the B-NW Master Cross Section Library continued with inclusion of new data for Np^{237} , U^{233} , U^{235} , Pu^{239} , and Pu^{241} . The Brookhaven ENDF library has been imported for general use and computer routines are being written to convert the ENDF data into the B-NW library format.

The performance characteristics of the modified Rotating-Crystal Spectrometer have been measured. The modifications result in an increased source intensity of 50% for the same resolution while the time resolution, angular resolution, signal-to-noise ratio, and detection efficiencies have been substantially improved.

Data obtained on the double-differential neutron scattering from H_2O have shown for the first time a clearly resolved component of the molecular vibrational excitation at about 0.2 eV broadened only slightly by multi-phonon coupling with the lowest energy modes of the system.

Measurements and review of H_2O measurements made with the triple-axis spectrometer have continued.

Measurements of MeV neutron total cross sections of H^1 , H^2 , and boron were compared in detail with results obtained by other investigators.

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

REACTOR ENGINEERING DEVELOPMENT

Calculations were performed to determine the time interval required to develop the hot spots detected by metallographic examination on the downstream end of the fuel element which failed in the FERTF. It was found that approximately 2-1/2 sec would be required to develop sufficiently high temperatures to cause the observed changes.

MATERIALS DEVELOPMENT

Corrosion tests with A 245 carbon steel in a concentrated solution of H_2SO_4 - HNO_3 - H_3PO_4 produced data showing corrosion losses at 100 °C and 130 °C were satisfactorily low. Teflon impregnated asbestos valve packing maintained suitable properties in the same solvent for about 4 hr. The behavior of graphitar pump bearing and asbestos packing in the H_2SO_4 - HNO_3 - H_3PO_4 was unsatisfactory. Treatment of these valve packing and pump bearing materials with concentrated H_3PO_4 showed that they could be used effectively in a H_3PO_4 solvent system.

Recent studies indicate that the sensitivity and reproducibility of the BNW method for measuring trace concentrations of H_3BO_3 in water can be improved significantly by increasing the indicator concentration and simultaneously decreasing the alkaline buffer concentration.

A two-step chemical decontamination procedure was employed to reduce the fuel contamination imbedded in the FERTF shroud tube from about 1 g to about 0.45 g. The solution used was primarily a mixture of hydrogen peroxide with oxalic and gluconic acids.

Prefilming of the PRTR primary coolant system is scheduled to begin very soon with the purpose of reducing corrosion product concentrations in the coolant at reactor start up. Specimens have been installed in the in-line corrosion loop to determine the efficiency of this process.

Laboratory evaluations were performed to determine the effect of velocity on carbon steel in citrate-oxalic acid mixtures. In the range of 1 to 10 ft/sec the corrosion rate increased linearly with the calculated velocity, doubling with an increase of 6 ft/sec.

Pilot plant testing of a PRTR dump valve has continued to assure operability of the valve during use of boric acid in the PRTR moderator system. Pilot-plant corrosion testing in the EDU Shim Test Loop has revealed no corrosion problems to the shim well with refreshment flow to the bottom of the well.

Aluminum and carbon steel corrosion probes in the PRTR have indicated negligible corrosion rates during the past month.

Laboratory corrosion tests are being conducted to evaluate aluminum corrosion resulting from the PRTR primary shield coolant contacting the moderator system shim wells and piping.

The removal of low temperature hard water scale and corrosion products from carbon steel pipe in 10% Diammonium citrate with 0.1% of a proprietary inhibitor was examined. Sections of a typical scaled pipe were

completely cleaned in 8 hr at 85 °C and 16 hr at 60 °C. Corrosion loss as measured by a blank was less than 1/4 mil. The deposits (100 mg/cm²) appeared to dissolve rather than undercut and slough off.

TEST REACTOR OPERATION

Work in April was directed toward reactor startup with the batch core loading. Completion of primary system activities, including replacement of the primary pump seals and a hydrostatic test to 1500 psi was accomplished. Cleanup of the moderator system was completed and the reflector coolant was transferred to the moderator system.

EBWR DEMONSTRATION PROGRAM

Three capsules with short sections of Zircaloy-clad EBWR rods, which were fabricated with vibrationally-compacted, pneumatically-impacted, UO₂-1.5 wt% PuO₂ have attained a maximum burnup in the MTR of 5.8×10^{20} fissions/cm³ (23,200 MWd/MT of fuel) at an estimated maximum rod power of 535 W/cm (16.3 kW/ft).

ADVANCED SYSTEMS

U²³³ FUELING OF A FAST COMPACT REACTOR

Sizing studies for liquid metal-cooled reactors with internally cooled- and pin-fuel cores were completed, using a revised computer program. Physics studies using a transport theory code were also completed. Kinetics computations have begun.

NUCLEAR SAFETY

CONTAINMENT SYSTEMS EXPERIMENT

Construction activity continued on the liquid sampling system and aerosol sampling system. Design for installation of the reactor simulator in a test stand outside the containment vessel was initiated. The shipping date for the simulator vessel was revised to September 9 by the seller.

A steam jet was used in tests to develop a method for injecting an aerosol into pressurized tanks. Results were not completely satisfactory, apparently because of liquid droplet formation in the jet. Superheating the motive steam gave improved performance; further tests were planned.

The multichannel analyzer code for determination of radionuclides in aerosol samples is operational in basic form, and storage of standard spectra was started.

PRESSURE VESSEL CRACK MONITORING

Analysis of acoustic signals emitted by a stressed 304-L SS tensile specimen has revealed two distinct types of signals. One is a random burst of relatively high amplitude energy, and the other appears to be a continuous frequency signal whose amplitude varies directly with specimen stress.

Burst signals, chosen arbitrarily from the vicinity of yield point loading, fit well into a triangulation pattern which includes sensing points and

the point of metal deformation. The primary mode of signal transmission in the metal appears to be shear wave. Velocities, calculated from known distances and the measured elapsed time, agreed with each other and with theoretical shear wave velocities to within 1%.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENTS

Installation of the WSEP is greater than 95% complete.

Melt compositions for satisfactory spray calciner-melter operation have been developed.

Interpretation of the data on the phosphate melt system continued.

LOW AND INTERMEDIATE LEVEL WASTES

Calcium and strontium nitrates added singly or together to glass melter condensate wastes before the ferricyanide-mixed oxide scavenging procedure considerably improved the Sr^{90} separation but only if the strontium carrier were present above a certain minimum concentration.

RADIOACTIVE SHIPMENT HAZARDS

Using melts of glass made from actual high level wastes and diluted with nonradioactive material similarly prepared, it was found that Sr^{90} was fractionally released at 1100 to 1200 °C to ca. 10^{-6} while Ce^{144} was released below its detection limit per hour.

FISSION PRODUCT AEROSOL CONTAINMENT

The sensitivity of detection of methyl iodide by the gas chromatograph using the electron capture detector was found to be 3×10^{-12} g in 25 μl . Certain paint thinners react with I_2 to give ethyl and probably methyl iodide in the presence but not in the absence of ultra violet radiation.

REACTOR SAFETY ANALYSIS AND EVALUATION

Two subsize vessels were inserted in the M-3 position of the ETR. These will make a total of four that have been irradiated.

A metal bladder was found to be an effective way to make a vessel pressure tight for the purpose of making compliance tests on through wall cracks.

Fatigue growth studies were started on vessels containing cracks starting from the inside surface.

CUSTOMER

PHILLIPS PETROLEUM COMPANY

Sixteen additional fuel element pins, loaded with 8 wt% enriched UO_2 pellets, were fabricated and shipped to the customer.

ATOMICS INTERNATIONAL

A trip was made to Atomics International, Canoga Park, California, to aid them in setting up a process for producing large numbers of end closures in SAP fuel tubes.

BROOKHAVEN NATIONAL LABORATORY

Two Brookhaven National Laboratory cermet capsules irradiated in the ETR were received for postirradiation examination. The pellets (34 from each capsule) were photographed and the length, diameter, and density of each pellet was measured.

DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY04 PROGRAM - F. W. ALBAUGHCIVILIAN POWER REACTORSSPECIFIC FUEL CYCLE ANALYSIS - E. A. EschbachApplication of Linear Programming

Additional attempts have been made to determine the minima of non-linear equations other than the previously described $x^2 - 10xy + 1000 = 0$ curve, with varying success. This approach has been completely successful only for curves which are basically of the form $y = ax + b/x$. Other means of forcing the solution for all types of equations onto the curve will be explored.

Four Group ALTHAEA - FCC Coupling

The FCC fast fuel cycle code modifications were completed. It now generates upon call the necessary reduced cross sections and other relevant input information necessary for operation of the four group ALTHAEA fuel cycle code.

Plutonium Value Document

A new Plutonium Value document is well underway at this point. Work is proceeding at a very satisfactory rate on this document.

Fast Reactor Economics

A large number of economics cases has been run on the Allis-Chalmers burnup simulation. Several results for this were presented last month. This larger parameterization is to go into a document on this reactor which is currently being written jointly by members of the Advanced Concepts and Analysis Group and the Engineering Physics Group.

Fuel Utilization Studies

The ZODIAC simulation of the Allis-Chalmers design for a fast breeder was used as input to VESTA. The reactor was simulated as a breeder with a plutonium core, and as a converter with a uranium core. The effect of such a converter on the uranium demand was considered. Three cases were considered; the first described the capacity which results when either simulation is used in conjunction with an HWR and a 7 yr doubling breeder; the second describes the capacity for a combination of the simulations and with the added assumption that the converter is shut down when plutonium availability limitations permit;* and the third shows the uranium required to sustain the capacities for these three cases. The great increase

* The converter cannot be shut down much earlier than 2005, within this framework, unless more thermal reactors are built, because the plutonium stockpile, while great enough to supply the demands on inventory for new capacity is insufficient to supply in addition the inventories needed to replace the converter with breeders before that time.

for continued uses of the converter is due to the large amount of natural uranium needed to supply the high enrichment of the converter. The dip observed for replacement of the converter represents the converter inventory being freed for other uses.

Code Development

The one-dimensional mode of the four group version of the ALTHAEA code has been debugged using a thermal cross section set. One series of calculations using a fast cross section set has been completed and appears correct. Rate of convergence was increased by inclusion of the Aitken acceleration technique to predict fission density at each mesh point. On simple problems the saving may be only 10 to 20%, while on complex problems the saving is 60 to 70%.

The JASON code has been revised to operate with the four group ALTHAEA code, but has not yet been tested.

CONCEPTUAL REACTOR DESIGN STUDIES - J. C. Fox

Fast Reactor Fuel Cycle Calculations

Allis-Chalmers Model, Economics Calculations

Fuel cycle costs have been assessed for batch operation (to near equilibrium) of a one dimensional oxide fueled fast breeder reactor model. This is a model which approximates the Allis-Chalmers 1000 MW_e Study Reference Design. Separately evaluated versions employ plutonium and U²³⁵ enrichment in the core for startup and batch feed. Neutronics and mass balance results for these systems have been given in previous monthly reports.

The total fuel cycle costs were obtained for a variety of cost assumptions. The U²³⁵ values were held at \$12.01 per gram, and various plutonium values were assumed, ranging between 83 and 250% of that values. The necessary computing has now been done to provide additional cost estimates for other U²³⁵ values.

The economics computations were performed using the QUICK code for both the U²³⁵ and plutonium enriched cores.

Fuel inventory interest charges were assumed to be 4.75, 8, 10, 12, and 14%. Other working capital charges were assumed at 8, 10, 12, and 14%.

Combustion Engineering Model

Neutronics mass balance information relating to the batch cycle operation of the Combustion Engineering 1000 MW_e Reference Design model has been obtained for four batch cycles (33,000 MWd/MTM core fuel per cycle).

The core was zoned (as in the reference design) in two regions to obtain radial power flattening. The inner and outer core zones contain 4485 and 4428 kg plutonium, enriched to 10.68 and 15.28% fissile content respectively. The core is surrounded by 0.3% U²³⁵ (depleted) in U²³⁸ carbide

blanket and a reflector composed of 80% sodium and 20% SS. Corresponding calculations for U²³⁵ core enrichment fueling provide 1.6 times the amount of fissile atoms at startup and in batch feed.

The feed enrichment was composed of fuel exactly like the original core load, with one-third the core volume being replaced (the most exposed batch) at the end of each cycle.

Region and total breeding ratios were computed at the beginning and end of each burnup cycle for both systems.

USAEC-AECL COOPERATIVE PROGRAM - F. W. Albaugh

Hydraulic and Nuclear Stability in Parallel Flow Channel Systems

The analog description of the hydraulic behavior of the BLW-250 is nearly complete. Some debug work is still required to remove drifts in calculated coolant densities. Modifications have been made that should remove this problem but have not been tested on the computer as yet. Completion of debug work on the hydraulics model is currently anticipated by mid-May.

In order to facilitate debugging of the hydraulics model, the reactor kinetics model has been simplified from four regions to a simple point model. This change will also permit comparisons to be made with results obtained previously based on the AECL point reactor study. Return to the four-region nuclear model is planned following completion of debug work on the hydraulics model and final checking of the present system.

A meeting was held with personnel of AECL and AEC-Headquarters to review progress and outline the future schedule for the program. Parameters to be varied for the parametric study were identified and plans were made for a brief study of side-to-side stability in the BLW-250.

Cross-Flow Mixing Between Parallel Flow Channels During Boiling

Analytical work to determine cross-channel mixing during two-phase flow in rod bundle fuel elements is being continued. Modifications to the computer code which calculates local flow rates and enthalpy in multirod bundles were undertaken in an effort to eliminate solution stability problems encountered earlier. A new iterative technique is being tried to evaluate the proper diversion cross flow between adjacent channels that will satisfy both the longitudinal and transverse momentum equations.

Assembly of the test section representing the square-triangular type flow channel was undertaken, and the precision welding of the heated portion was completed successfully. All other machined components for this test section are on hand, and completion is expected in several weeks. This section will be used for experimental determination of turbulent and diversion mixing of coolant during two-phase flow conditions.

In-Reactor Corrosion Evaluation of Zirconium Alloys

Samples of six zirconium alloys were exposed 25.5 days in the ETR G-7 loop in pH-10 NH₄OH at dissolved oxygen levels less than 0.05 ppm. The bulk outlet water temperature was 280 °C. The samples were exposed at two in-flux positions ($\sim 1 \times 10^{14}$ and $\sim 2 \times 10^{13}$ nv, > 1 MeV) and two out-of-flux positions, located in the loop, ~ 1 sec and ~ 1 min after the stream

leaves the high flux zone. Flux-induced oxidation on etched specimens was small or absent (see Table); however, preautoclaving appeared to sensitize VALLOY, Zircaloy-2 and Zircaloy-4 toward enhanced in-reactor oxidation. VALLOY-S showed relatively high weight gains but no significant flux-induced oxidation. Metallography on specimens of each alloy shows oxide thickness in general agreement with weight gains.

Corrosion Behavior of Zirconium Alloys Exposed in the ETR
G-7 Loop

Alloy	Maximum Weight Gain, mg/dm ²	Weight Gain Ratio In-Flux/Out-of-Flux ^(a)		Hydrogen Pickup, ppm
		$\frac{2 \times 10^{13} \text{ nv}}{\text{Fast}}$	$\frac{1 \times 10^{14} \text{ nv}}{\text{Fast}}$	
Zircaloy-2 annealed, etched	15.3	1.2	1.2	10
Zircaloy-2 annealed, preautoclaved	11.9	1.7	3.4	
Zircaloy-2 80% CW, etched	16.9	1.3	1.5	20
Zircaloy-4 annealed, etched	13.7	~1	1.4	
Zircaloy-4 annealed, preautoclaved	10.9	~1	3.0	19
VALLOY ^(b) annealed, etched	12.3	~1	~1	4
VALLOY ^(b) annealed, preautoclaved	111.0	6	17	49
VALLOY-S ^(c) annealed, etched	28.7	~1	~1	
VALLOY-S ^(c) annealed, preautoclaved	20.3	~1	~1	86
Zr-3Nb-1Sn annealed, etched	15.7	~1	1.6	3
Zr-3Nb-1Sn annealed, preautoclaved	5.0		~1	
Zr-2.5 Nb ^(d) quenched 20% CW aged, etched	12.5	~1	1.2	2
Zr-2.5 Nb ^(d) quenched 20% CW aged, preautoclaved	8.0	~1	1.2	
Zr-2.5 Nb ^(d) quenched 30% CW aged, etched	10.7	~1	~1	
Zr-2.5 Nb ^(d) quenched 30% CW aged, preautoclaved	3.9	~1	~1	7
Zr-2.5 Nb ^(d) annealed 20% CW etched	10.7	1.2	~1	2
Zr-2.5 Nb ^(d) annealed 20% CW preautoclaved	7.5	<1	<1	

(a) Ratios were calculated using average in-flux and out-of-flux weight gains.

(b) Zr-1.2 Cr, 0.08 Fe, supplied by Vallecitos Laboratory, General Electric Co.

(c) Zr-1.2 Cu-0.28 Fe, supplied by Vallecitos Laboratory, General Electric Co.

(d) Supplied by Chalk River Laboratory, Atomic Energy of Canada, Ltd.

Hydrogen pickup data for in-flux specimens are shown in the last column of the table. Samples analyzed to date show relatively low hydrogen pickups, particularly in niobium-bearing specimens. An exception is the relatively high pickup for VALLOY-S.

This is the third in a series of G-7 loop exposures at varying oxygen levels, ~1 ppm, ~0.1 ppm, and <0.05 ppm. There is an apparent decrease in flux-induced oxidation as the oxygen level decreases. However, final interpretation of the latest exposure must await investigation of the high weight gains on preautoclaved VALLOY. A 4-cycle G-7 loop exposure of the TAC alloys will be completed in May to investigate the effect of the low-oxygen environment on breakaway phenomena previously observed in high-oxygen exposures.

Evaluation of Zr-2.5 wt% Nb Pressure Tubing

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

Work is in progress to develop an apparatus for continuously measuring the diameter creep strains in a stress rupture specimen while it is exposed to PRTR irradiation inside of a standard PRTR pressure tube. A mockup of the apparatus is now in series with a loop operating at 280 °C in circulating water. Following a 400 hr run at a hoop stress of 44,000 psi, a second run at a stress of 54,000 psi has begun. Subsequent tests will incorporate modifications to improve the strain sensitivity of the apparatus.

APPLIED AND REACTOR PHYSICS

PLUTONIUM CRITICALITY STUDIES - E. D. Clayton

Criticality of Plutonium Compounds

In processing plutonium fuels, compounds of plutonium are encountered over a range of densities and H/Pu atomic ratios. Experimental data are lacking on the criticality of these compounds in the undermoderated range. Calculations have been completed for PuN-H₂O solutions having H/Pu ≤ 20. Constants were determined in a critical mass equation that may be used for predicting the critical mass over a range of densities for any given H/Pu ratio in the range H/Pu ≤ 20. These constants were determined from computed critical radii of water reflected spheres of PuN-water solutions having densities ranging from theoretical to 0.1 theoretical for five H/Pu ratios.

The critical mass equation is,

$$M_c = A \rho_{\text{core}}^{-m} \rho_{\text{refl.}}^{-n},$$

where ρ_{core} and $\rho_{\text{refl.}}$ are the densities in g/cm³ of the core and reflector and M_c is the mass in kg. The following data were obtained for the PuN-H₂O solutions.

<u>H/Pu</u>	<u>A</u>	<u>m</u>	<u>n</u>
0	454.8	1.47	0.53
1	332.7	1.50	0.50
5	108.9	1.59	0.41
10	39.5	1.63	0.37
20	10.1	1.66	0.34

The core density exponent, m , for PuN solutions is nearly the same as that given in TID-7028 for plutonium atoms in water at $H/Pu \leq 1$. At an H/Pu ratio of 20, however, the exponent for plutonium atoms in water is 1.53, as compared with 1.66 for PuN-water solutions. These results were obtained with DTF transport theory code, using multigroup constants (18 energy groups) obtained from the GAMTEC-II computer code.

Computed values of criticality have agreed well with experiment for PuO_2 -polystyrene compacts with an H/Pu atomic ratio of 15 for 2 wt% Pu^{240} . Similar calculations on material containing 8 wt% Pu^{240} have consistently predicted a slightly larger critical mass than measured. The effect that micro size PuO_2 particles (7.5 μ) has on resonance capture in Pu^{240} may be the reason, and this is now being investigated. Present methods of computing resonance absorption in finite geometries assume the absorber is embedded in an infinite moderating media. Dancoff shielding corrections are usually applied for nearby resonance absorbers. However, for the high density PuO_2 -polystyrene compacts the Dancoff correction would be inaccurate. To a first approximation, the resonance shielding in the particles is computed by assuming a finite size sphere embedded in a homogeneous absorbing media. The mathematical representation of the above problem has been derived by extending the Adler-Nordheim approximation. The GAMTEC-II code has been modified to perform the required calculations.

HIGH TEMPERATURE REACTOR LATTICE PHYSICS STUDIES -

R. E. Heineman

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

Reactor and Building Construction

The reactor building and associated utilities will be completed by the contractor on about May 4. Only a few parts of the Acceptance Tests on the heating and ventilation system remain to be performed. The offices, control room, and other selected portions of the building will be given to Battelle for beneficial use.

The Phase II contractor (reactor construction) is installing thermal insulation inside the steel shell and is constructing the doors and hatch covers for the shell. The piping and electrical work has been reduced, awaiting the delivery of additional equipment for installation.

The graphite lintel bars, which support the stack above the experimental cavity, are still scheduled for delivery on June 20. The rough machining for all graphite heating rods has been completed. These blanks have been shipped to Battelle for final machining.

All but three of the Acceptance Test Procedures (ATP) for the reactor and its systems have been written and approved.

Mockup and Materials Studies

The four samples of lanthanide oxide-nickel cermet which were obtained for evaluation as a poison material in the HTLTR safety rods have been described in previous reports along with the results of reactivity and corrosion resistance tests on them.

Two of these samples, those which are in the form of 1/16 in. sheet containing 4 wt% each of Gd_2O_3 and Eu_2O_3 , have been submitted for tensile tests following their removal from the 200 hr corrosion test. Sample 3 is the same as 4 except for a dispersion of the oxides. These tests provide the following comparisons between the two grades of cermet and TD-nickel sheet similarly exposed:

	Tensile (psi)	
	Room Temperature	1000 °C
TD-nickel	67200	18600
Cermet (Sample 3)	64500	10200
Cermet (Sample 4)	56000	7400

Similar tests performed by Sylvania (the material supplier) at 1800 °F provided the following data:

	Yield (psi)	Tensile (psi)
Sample 3	8500	9300
Sample 4	4800	7000

This compares favorably with the BNW values.

A review of all test results indicates that the material appears to be acceptable for the use intended and hence has been recommended for procurement.

Reactor Equipment

The responsibility for development and procurement of the two experimental oscillators which move lattice and material samples in the reactor have been assigned to Battelle. The major effort during the month was directed toward analysis of different types of drive systems and drafting of oscillator details. During the month eight drive systems were analyzed. The preliminary choice was one which utilizes an oil shear, constant slip, variable speed drive unit with position type feedback. Other systems have not been ruled out but major effort will be applied to the constant slip, variable speed system.

The first set of detail prints for the oscillators were completed and checked. Modifications required for the second set of drawings are 80% complete.

An estimated 30-day delay in the procurement of the oscillators has resulted from lack of funding. Delivery of certain vendor supplied drive system components may prove to be critical in completing the work as scheduled.

The detailed drawings of the safety rods were revised to incorporate the coprecipitated, rare earth-nickel sheets in place of the B₄C plates. The control and safety rods have been received from the vendors and are being assembled for functional testing.

The modifications to the gas sampling systems have been approved by the AEC for incorporation into the design. These changes are needed to accommodate the water removal system, for which cost estimates are now being sought. Final procurement action is being initiated on the teletypewriter, the neutron generator, and the flux monitoring equipment. All equipment specifications have now been approved.

Data and Control System

Acceptance Tests

The ac power supplied to the Programmed Measurement and Control System (PMACS) has been improved. The harmonic content appears to be less than the 3% allowed in the specifications. This was done by changing the hookup of the voltage regulator which supplies PMACS. The new configuration increases the sensitivity of the regulator units to variations of the line voltage but reduces the harmonic content of the output. The response to variations of the line voltage has not been measured yet.

The intermittent problems that were occurring in the paper tape reader and the teletype have been corrected. The paper tape reader required adjustment of the capstan pinch roller and the teletype was corrected by an adjustment of the spacing mechanism.

The alphanumeric display failed several times during the month. Two video amplifiers and the "Ø" character decode card required transistor replacements.

The analog display horizontal and vertical amplifier signals remain noisy, even after the ac power harmonics were reduced below that required by the PMACS specifications.

The video amplifiers were rebuilt again to provide transistor protection in the circuits. The amplifiers have now operated several days without failure. The noise in the analog display signals was traced to the +18 volt power supply. The operational amplifiers in the digital-to-analog converters for the analog display do not reject enough of the +18 volt noise. It appears that, with proper filtering, the noise can be reduced, possibly to an acceptable level.

Programming

A systems program to permit on-line debugging of HTLTR programs was completed. A program to provide a symbolic print-out of Funny Format binary tapes was written to be used in debugging and making changes to

programs where symbolic listings are not available. The Teletype Output Package was revised for use with the KSR-33 Teletype only. The original written for the Type 28 Teletype was longer than necessary and did not work properly in all cases.

A preliminary program was written and tested for the alphanumeric display of control variables. Tentative decisions have been made concerning memory storage space, buffer locations, and calling sequences.

The heating rods for the HTLTR core are divided into four circuits for proper distribution of heat. The power to these circuits will be controlled by PMACS, which acts as a controller. A transient, heat transfer program, TIGER V, has been used to obtain the relationship between the heat input to the graphite stack and the resultant temperatures throughout the graphite, the insulation, and supporting concrete as a function of time. The first calculation of the transient response, which utilized a large number of spacial nodes in three dimensions, was not reasonable. This model was too complicated to determine whether failure was caused by poor convergence or errors in input. A model using a smaller number of nodes in two dimensions gave reasonable results. Calculations were then made in which variations in the number, spacing, and dimensionality of the nodes were tried. A satisfactory configuration that will be used for future work consists of 36 internal nodes, 12 surface nodes, and 4 boundary nodes. There are 89 connectors between nodes. The program output consists of matrices of 52 temperatures, one for each node. These matrices are printed once every 10 iterations. For 250 iterations over a real time of 5×10^5 sec, the program running time is 52 sec. Calculations showing the step function response between the process inputs and outputs will be obtained. These data will then be used to define the type of control algorithm that will be used for the heater control. The control algorithm will then be programmed for the PDP-7 computer in PMACS.

Reactor Operations

Two members of the operations staff have been assigned to assist in writing and debugging computer programs which will control process variables and display or log process or experimental data. In addition, a formalized 12 hr course on programming was attended by all personnel.

A draft of the Operating Safety Limits was completed and submitted to the AEC, Richland Operations Office.

Spare parts for control rods and PMACS have been set up in General Stores. The control rod spares are now on hand and procurement of instrument spares has begun.

Test equipment for the maintenance of various instrumentation and control systems has been obtained.

Reactor Physics Program

Experimental Program

The experimental program for the HTLTR was reviewed for the HTGR Technical Advisory Committee during their meeting in Richland, April 20-21, 1966. Plans for the first 2 yr of experimental use of the reactor were

covered. The following main features of the program were pointed out. The work is to include studies of natural uranium and urania-233-thoria lattices in a graphite moderator. Measurements of η and various cross sections of several fissile, fertile, and absorbing materials will be made as a function of temperature. Both geometrical dimensions and material composition will be varied in the experiments with lattices, but the temperature of the system, or parts of it, will be the main parameter over which a broad survey will be made. With the fuels now projected, and with the graphite parts already constructed, it will be possible to assemble experimental lattices having moderator to fuel atomic ratios ranging from 450 to 5×10^4 . This contains the region of interest for high temperature gas cooled reactors, which is from 2000 to 6000.

Preliminary discussions and correspondence with fuel fabricators lead towards procurement of the special fuel, moderator, and poison materials needed for this experimental program. The plan is to have the materials made into microspheres. This form is adaptable to high temperature use and would also permit the separation of the fissile, fertile, and moderator materials by a simple sieving process whenever it is necessary to change their relative proportions.

A series of experiments utilizing the unique features of the HTLTR to measure thermally averaged cross sections of a number of materials over a wide range of temperatures are being planned. Materials under consideration include ones which are useful in basic reactor measurements, ones which have immediate interest for design of high temperature reactors, and ones which have probable future uses in high temperature reactors.

To aid in the planning of these experiments, a study of the HTLTR neutron flux and adjoint flux spectra over the temperature range from 100 °K to 2000 °K has been initiated. Specifically, the driver ring radius necessary to obtain a Maxwellian spectrum at the center of the test region, methods of calculating thermal adjoint flux spectra in the test region, and the available methods of calculating the temperature dependence of the cross sections were determined. The most expeditious way of obtaining thermal adjoint spectra appears to be through a modification of THERMOS to permit multithermal group cross section averaging at the upper and lower energy ends of the THERMOS macrogroup. These cross sections would then be used in HFN to calculate a thermal adjoint flux spectrum. The use of HRG to calculate Doppler-broadened epithermal cross sections is subject to the limitation that in the case of extremely high temperatures, resonance integral approximations valid for narrow resonances may become invalid and the NR1A approximation should be used. The magnitude of this effect can always be estimated by comparing the HRG resonance integral value with exact values obtained from the ZUT or NIT codes, and no doubt some such inter code comparisons will be necessary to cover the desired temperature range accurately.

Pilot calculations, all at room temperature, were initiated and are about half complete after some difficulty running HRG on PHYSICS CHAIN was encountered.

The lattice experiments in the HTLTR are being planned to use lattices of interest which have not been poisoned to a neutron multiplication of one.

Thus, the neutron flux will have a nonzero gradient at the edge of the cell. In order to examine the detailed magnitudes of various small corrections to the experimental data and to have preliminary estimates of the parameters to be measured, the computer code HFN was modified to iterate on the value of $\Delta\phi/\phi$ at the cell boundary for a given value of k_{eff} specified in the output. In a completely periodic heterogeneous lattice, this parameter should be energy independent, and hence it represents a very convenient boundary condition for lattice cell calculations.

Because of its low melting point (321 °C) cadmium cannot be used as a spectral filter for any experiments in HTLTR except those at the lower temperatures. The most suitable substitute appears to be gadolinia; hence, preliminary discussions with laboratory personnel have been held concerning the feasibility of producing and sealing gadolinia containers.

Initial indications are that it is probably feasible to produce these filters by combining the gadolinia with a suitable binder and enclosing them in a nickel sheath to simplify closure and ensure containment of the gadolinia.

Automated Learning Logic

As reported last month, Theoretical Physics work on improvement of least-squares fitting methods has led to the discovery of a computer loop capable of learning much as the human mind learns, by sifting linear correlations from a sequential flow of observations. Two such learning loops are being formulated on the PDP-7 digital computer at the console of the High Temperature Lattice Test Reactor, for exploratory use during the initial approach-to-criticality and control-calibration runs.

Each learning loop is designed to provide real-time processing of reactor measurements into evaluated reactor theory, at a speed of about twice the normal measurement pace. The logic is evolving along the following lines, for these first two learning loops.

1. Learn Critical Mass (BNW Program Learn-I)

Observe subcritical inverse multiplication as a function of fuel loading, for up to five different neutron sensors. Look for asymptotically-linear approaches to a common limit. Upon request, predict critical fuel loading, critical fuel-loading uncertainty, and next safe fuel-loading increment.

2. Learn Asymptotic Period (BNW Program Learn-II)

Observe successive supercritical doubling (or tripling) times. Look for asymptotic coalescence. Periodically predict asymptotic period and its uncertainty. Upon request, convert accrued knowledge to reactivity and reactivity uncertainty by use of transport perturbation theory.

During the past month, this work progressed through formulation and debugging of the learning logic of the second loop (Learn-II). Conversion to reactivity comes next, and then the first loop (Learn-I). The inverse development sequence here assures accrual of relevant feel for computer reliability limitations prior to formulating the safety-sensitive prediction of critical mass.

PHOENIX FUEL REACTOR PROGRAM - P. L. Hofmann

MTR-Phoenix Experiment

Nuclear Design Calculation

A continuing problem in the design of the MTR-Phoenix experiment is the determinations of detailed three-dimensional power maps for the core. As indicated in previous reports, splicing of 2-D and 1-D calculational results to obtain a 3-D map have not proven very successful.

In the procedure presently used, four 2-D calculations (three horizontal cuts and one "side" view) are being spliced together to obtain a 3-D power representation. Two horizontal 2-D calculations represent the core with the control rods completely inserted and completely withdrawn. The third 2-D horizontal calculation represents the control rod followers below the core. One dimensional calculations, using HFN, are then made on the unrodded core, rodded core, and the fuel region below the core to determine that value of the transverse buckling that would yield the same k_{eff} as the 2-D calculations. In the unrodded core, the values used for the nuclear densities are the volume weighted nuclear concentrations. For the core with control rods inserted, the volume weighted nuclear concentrations are further reduced by flux and volume weighting using the flux values from the 2-D rodded case.

With the three buckling values thus obtained, a 2-D axial computation is then made using material dependent bucklings to represent the three situations of the unrodded core, the rodded core, and the fuel below the core. In the table below, the values of k_{eff} from the 2-D horizontal calculations with a constant buckling value and the 2-D axial calculations with the material dependent buckling are shown. The agreement between the calculations is quite good, which indicates that there are no gross differences between the systems, and that the proposed splicing scheme might be useful.

Two Dimensional Calculations on MTR Phoenix Core

	k_{eff}	
	<u>Horizontal Cut</u>	<u>Axial Cut</u>
Unrodded Core	1.264	1.268
Rodded Core	0.905	0.907

MTR Core Thermal Hydraulics Analysis

The latest power peaking estimate for the partial plutonium core indicates that the maximum overall core peaking factor could be about 5.0. This is without incorporation of flux suppressors or fuel zoning and occurs directly under a control rod about midway axially through the core. It is estimated that the axial peaking contribution is about 3.0 and the horizontal about 1.67. Under these conditions the maximum reactor power, based on a 245 °F (118.3 °C) nominal surface temperature limit would be about 21 MW.

Debugging the modified thermal hydraulics code is underway and should be completed in the near future. This program is a version of the one now in use, but differs in that hydraulics calculations are incorporated with the heat transfer calculations.

Flux Suppressor Design

It was suggested that flux suppressors for the symmetric 3 x 9 core be mounted on the adjacent L-pieces provided necessary modification of these pieces is permissible. This would greatly simplify incorporation of flux suppressors in the core since it could be accomplished simply by machining the L-pieces to accommodate the suppressor plate. Hence, the difficulty of fabricating the suppressors as an integral part of the fuel assembly is eliminated.

PRCF, CAF Experiments

Preliminary work has begun on the calculations for evaluation of the safeguards of the PRCF-Phoenix experiment. Areas under investigation are: heat transfer and moderator void formation models, values of heat transfer and kinetics parameters, the significance of the beryllium reflector and the three-zone core to kinetics calculations. Some simple kinetics calculations were performed using the MIDAS program.

Concerning the CAF experiments, the final draft of the description of these experiments has been completed. It has been submitted to the AEC for information and for approval of necessary revisions of operating limits.

Fuel Fabrication for PRCF

An economical method to produce the 24.625 in. long by 2.845 in. wide by 0.040 in. thick picture frame was developed. The picture frames and clad plates are being fabricated. The welding technique was developed and dummy plates were welded to check dimensional control. Over one-half of the Pu-Al core plates were rolled and prepared for assembly. Additional high exposure plutonium will be required to complete the requested number of plates.

REACTOR FUELS AND MATERIALS

FAST FUELS OXIDES AND NITRIDES - W. E. Roake

FTR Driver Fuel Irradiation Tests in EFFBR

Work was started on the preparation of a test proposal for irradiating candidate FTR driver fuel (SS-PuO₂ cermet and UO₂-PuO₂) in the Enrico Fermi Fast Breeder Reactor (EFFBR). Mixed oxide fuel pins are to be irradiated to investigate: (1) the difference in irradiation behavior between packed-particle fuel and pellet fuel, (2) the effectiveness of mechanical restrictors to reduce or eliminate fuel relocation, and (3) general irradiation performance characteristics. With SS-PuO₂ cermet fuel, under investigation will be (1) matrix stability, (2) pin swelling characteristics, and (3) fuel-matrix-cladding interactions.

Currently, the mixed oxide fuel pins are scheduled to operate with a maximum linear heat generation rate of 13 to 15 kW/ft for 225 full power days. Calculated maximum exposure on these pins will be about 48,000 MWd/ton. The cermet pins will operate with a linear heat generation rate of 8.2 kW/ft to a calculated maximum burnup of about 8.5% of the plutonium atoms.

Pin designs will incorporate features which will permit TREAT experiments to be performed on the mixed oxide and cermet fuel pins irradiated in the EFFBR. Toward this end, a visit was made to Idaho Falls to discuss fuel specimen specifications for TREAT experiments with ANL. A sodium loop facility is currently being installed in TREAT and should be available for use with the mixed oxide and cermet fuel pins.

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

Thermal-Diffusivity Measurements

The thermal diffusivities of several samples were measured with the laser-pulse diffusivity apparatus which was recently constructed. The thermal diffusivity of a single crystal UO_2 sample was measured from 100 to 1400 °C and the results compared favorably with previous measurements on another apparatus. Preliminary data were obtained on pneumatically-impacted PuO_{2-x} . Measurements were obtained from 100 to 1500 °C. Additional experiments are planned to verify the results and to extend the measurement range to 1700 °C.

Irradiation Performance of Fast Reactor Fuel Candidates--UN-20 wt% PuN

Mass spectrometry and α -energy determinations of the plutonium isotopic distribution in samples removed from various radial positions in irradiated UN-20 wt% PuN fuels were begun. Fuel rods with 2.4×14^{20} and 13×10^{20} f/cc, both irradiated at nominally 65 kW/ft, are being examined. Results are needed to determine whether the gross plutonium redistribution indicated by α -count-rate measurements on these samples represents concentration or isotopic gradients across the fuel radius.

Mixed-Nitrides Research

Procedures used in the synthesis of UN and PuN from the metals have been improved and standardized. Batches of 20 g of PuN and 40 g of UN which have well developed crystal structures and no oxide contamination (as indicated by X-ray diffraction analysis) are being routinely produced.

Experiments to identify the conditions which yield well homogenized solid solutions of selected compositions in the UN-PuN system are in progress. Variables being investigated include time, temperature, particle size, and mixing conditions. Experiments on sinterability of the 80% UN-20% PuN mixed system are underway.

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.

Irradiation Program

A controlled temperature (low pressure) irradiation capsule reached its goal exposure of 0.2 at.% burnup and was discharged. The capsule was controlled at 700 °C (beta phase) while the reactor was up and at 500 °C while the reactor was down. A tandem assembly of two capsules continues 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.% BU; the other is operating at low pressure (~30 psi) and 525 °C toward a goal exposure of 0.4 at.% BU. A controlled temperature (low pressure) irradiation capsule designed to operate at 450 °C toward a goal exposure of 0.3 at.% BU has been shipped to a reactor for charging. These capsules, containing enriched and natural uranium specimens, will provide data needed in evaluating the effects of temperature (both alpha and beta phase), pressure, burnup, and minor alloy additives on the irradiation behavior of various fuel compositions.

The possibility of irradiating capsules containing uranium samples of various enrichments which would provide several burnup levels in one capsule is being investigated. A controlled temperature (low pressure) irradiation capsule has been modified to allow relatively easy adjustments of the heat transfer fins. Laboratory experiments are being conducted to determine the heat transfer characteristics of this capsule design.

A paper, "Controlled Temperature-Pressure Irradiation Capsules," BNWL-SA-583, has been written for presentation at the International Symposium on Capsule Irradiation Experiments, Pleasanton, California, May 3-5, 1966.

NONDESTRUCTIVE TESTING - J. C. Spanner

Detection of Irradiation-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. Conceptual design of a test facility for nondestructively examining irradiated specimens is concurrently in progress.

Tests were performed on four irradiated and four unirradiated Charpy samples of A 212 B and A 302 B steel. Three separate tests were conducted, but data from one of the tests were not valid. The two valid tests were performed at 2-1/4 MHz and 5 MHz. A null balance between direct surface reflection components and lateral shifted boundary wave components--as established on unirradiated specimens--was disturbed on irradiated specimens. These disturbances resulted in signal amplitude changes of up to 6:1 at 2-1/4 MHz and 10:1 at 5 MHz. Certain anomalies were also observed in that null regions were found on irradiated specimens. The dominant conditions; however, were a null balance on the unirradiated specimens and an unbalance on the irradiated specimens.

Two special transducers were fabricated and evaluated for use in performing these measurements. These transducers were constructed of 2 MHz X-cut, PZT-5 and were damped with a tungsten loaded epoxy. Both were 1/4 in. in diameter and each had high sensitivity and good beam symmetry.

Ultrasonic Transducer Development

Development and evaluation of new ultrasonic transducer concepts and improved fabrication methods were continued. A multiunit, concave focused transducer was assembled and evaluated. Each of the four crystals in this unit can operate individually or in conjunction with the others on both transmit and receive and they may also be used either broadband or narrowband, or both, by sequential pulsing. This capability is potentially applicable for two-frequency test applications. Nonuniform or oriented polarized shear information can be detected by ratio measurements between diametrically opposite sets of crystals. This ability may also be useful for grain orientation or residual stress measurements. Other measurements, such as thickness, acoustic impedance, attenuation at more than one frequency, and the shear-to-longitudinal velocity ratio, are also potentially obtainable.

An attempt was made to use back-to-back bucking of two piezoelectric elements to effectively phase cancel all but the exposed element's first surface pulse. This technique did not prove feasible because the internal sound waves and the voltages generated could not be adequately disassociated.

A supply of broadband, high sensitivity, high resolution transducers with good beam symmetry is being fabricated to provide versatile transducers for various feasibility studies. The total supply will include transducer diameters from 1/8 through 1 in. and frequencies from 1 through 20 MHz in both flat and lens focused elements. From two to four transducers of each type will be fabricated and all transducers will be individually evaluated and performance records will be maintained.

Electromagnetic Testing

Tests are being implemented to evaluate the use of the maximum value of the defect signal in each signal channel of the multiparameter tubing tester instead of using the total signal, which includes all values of the defect signal locus as the test coil approaches and leaves the defect area. This will result in signals which have less information content than is available. However, the present tester has capabilities of separating only four parameters and cannot ideally handle all of the information available in the total signal. Thus, reducing the amount of information available may result in a signal which may more appropriately characterize the specific parameters of interest.

To do this, three main circuit functions, timing, sampling, and pulse stretching, all new to the multiparameter test, are required. The timing function is needed in gating the signal sampling circuits at the time when the defect signals are maximum. The signal sampling circuits, so timed, will sample the signal in each channel of the multiparameter tester, preserving polarity. These sampled signals, too short in duration (200 μ sec) for proper display, must be stretched to a duration of 0.01 sec while still preserving polarity and original relative amplitude.

The timing, sampling, and pulse stretching circuits have been breadboarded and are being tested for single channel operation. The timing circuit comprises (1) a full wave rectifier which produces signals of single polarity from the defect signals, (2) amplifiers, (3) a differentiating circuit, and (4) a

zero crossing detector which produces a timing trigger at the peak of the defect signal. The sampling circuit uses a shorting type gate in which defect signals from the multiparameter tester are shorted to ground through a transistor, except during a 200 μ sec sampling period. This sampling period is established by a univibrator which is initiated by the timing trigger.

The timing and sampling circuits meet the needs of the evaluation test. The pulse stretcher circuit functions generally as desired, except for some undesired variation in output pulse shape. After this condition has been corrected, the timing circuit and four sampling and pulse stretching circuits will be assembled in more permanent form for evaluation of the new technique with the multiparameter tester.

Investigation of the six-diode gate circuit as a phase detector is nearing completion. The two channels for the detector have been built with modifications in the reference signal squaring circuits to improve the accuracy of the phase angle calibration. Provision has been made to adjust the detector to zero output with zero input signal. Previously, a nonzero output was possible. Circuitry has been included to isolate the low impedance circuits of the detector from the signal and reference sources.

Initial measurements indicate that the instrument will give readouts of amplitude and phase for signals in the range of 25 kHz to 500 kHz. Signals up to 4 V peak-to-peak can be used with amplitude linearity within 2%. Over the same range, phase angles are accurate within 1 degree. Response of the system to phase and amplitude changes is approximately 5 msec.

Infrared and Thermal Research

Development of an instrument that will employ sinusoidal thermal wave methods to rapidly measure high and low temperature thermal properties of materials is continuing. No direct contact between the samples and measuring device will be necessary.

The immediate goal of this work is to improve the present prototype so that accurate thermal properties measurements can be made on highly conductive materials. Two transducers are presently being developed. The first of these, a miniature transducer, was found to be much less sensitive than expected for measurements on highly conductive samples. In addition, input thermal impedance of the transducer was influenced more by thermal diffusivity of the samples than by the thermal conductivity-heat capacity-density product. Subsequent theoretical work has shown that thermal diffusivity should indeed influence the results when the transducer is smaller than about $1.3 (\alpha/f)^{1/2}$, where α is thermal diffusivity and f is the frequency. Hence, small transducers should afford a means of measuring thermal diffusivity and detecting cracks perpendicular to the outer surface. Work to improve the sensitivity of the miniature transducers is continuing.

As predicted, increasing the size of the transducer has improved sensitivity. Confirmation was obtained in a number of experimental measurements using a 3 in. square transducer with helium coupling on aluminum, graphite, lead, stainless steel, and quartz. These measurements on the large transducer also indicated that the Argand plane locus of the transducer

thermal input impedance, generated by changes in the sample to transducer spacing, was different from the locus generated by changes in the sample thermal properties over much of the possible range. (A comparable result was obtained over a limited range last month with the miniature transducer.) Hence, the sample thermal properties can be measured in this range without dependence on transducer to sample coupling, as was predicted theoretically in 1964 in HWIR-1792. Additional work will be required; however, to eliminate coupling effects in measurements on high conductivity materials (such as aluminum and copper). In addition, sensitivity of the large transducer, although much higher than that of the miniature transducer, is not yet as high as desirable for these materials. Theory indicates that further improvement is possible. In general, the higher the "nonloaded" transducer impedance (no sample in place), and the smaller the coupling impedance between the transducer and sample, the greater the influence of sample thermal properties on the "loaded" transducer impedance. However, due to the finiteness of practical heat sinks, high impedance transducers also have smaller possible peak-to-peak temperature variations and signal-to-noise limitations in the temperature measuring equipment become important as the nonloaded impedance is increased. Present transducers may be further improved by reducing their input impedances, since the temperature measurement signal-to-noise ratio is the major present limitation in accuracy. In addition, an experimental quantity of high sensitivity, low mass, solid state temperature sensors, which should improve the signal-to-noise ratio, has been ordered.

It must be emphasized that accuracy of transducer input impedance measurements alone is not sufficient to obtain accurate thermal properties measurements; maximum dependence of the transducer input impedance on sample properties is also important.

Further effort is being applied in both areas to develop a rapid, high accuracy, noncontacting thermal properties measuring system for solid materials at high and low temperatures.

Fundamental Ultrasonic Studies

A new approach to the study of pulse distortion that is experimentally encountered as compared to calculated distortion has been undertaken. The new studies attempt to obtain frequency dependent attenuation curves for unknown materials by analysis of measured pulses. Previously, attempts were made to wave model by assuming that the attenuation curves were known.

To systematically determine the best procedure for obtaining the attenuation curves from measured pulse data, it was necessary to re-examine the methods of Fourier series analysis which had previously been used. Since systematic errors do occur in the experimental data, it was desirable to optimize the mathematical methods for smoothing the data using the methods of least squares. The least square methods that had previously been developed for use with Fourier series were found to provide optimum results with few exceptions. Following these studies, a computer program was written to yield the attenuation curves of unknown materials from measured broad frequency band pulses. The program is currently being debugged, and initial results indicate the approach will be successful.

Further studies with the model for attenuating waves disclosed that transient responses may exist which had previously not been encountered. The transient term has a frequency less than the incident wave frequency by an amount determined by the material constants.

Attempts were made to assess nonlinearities associated with the electronic apparatus used for previous ultrasonic measurements. This was of interest as a possible source of error in the pulse distortion recordings. Though the test equipment selected for these tests was believed to have good, high frequency capabilities, it was found that undesirable loading effects made it difficult to obtain good quantitative results.

The new equipment for grinding spherically focused transducers was received and fabrication is proceeding. A special transducer holder that will accommodate both flat and spherically focused elements has been designed and fabricated. Electrical connections are made by applying silver loaded cement to the transducer faces and allowing the cement to overlap the metal portion of the holder. Concentric metal portions, which are insulated from one another, then become the electrical connections. This arrangement, therefore, does not require localized soldering which has presented fabrication problems in the past.

NUCLEAR CERAMICS - R. E. Nightingale

Stoichiometry Effects on Irradiation Behavior of UO_2

Postirradiation metallography and autoradiography was completed for pneumatically impacted $\text{UO}_{1.99}$ and $\text{UO}_{2.05}$ specimens and their $\text{UO}_{2.00}$ standards. These results, together with previous studies of $\text{UO}_{1.90}$ and $\text{UO}_{2.02}$ and work in progress on $\text{UO}_{2.15}$, will yield a correlation of fuel-temperature profile with stoichiometry in the composition interval $1.90 \leq \text{O/U} \leq 2.15$.

Problems encountered in diamond-core drilling the large samples required for determining radial oxygen redistribution were solved by increasing the drill "chip clearance" and internally cooling the drill. Radial samples were removed from $\text{UO}_{2.00}$ and $\text{UO}_{2.05}$ specimens. Coulometric analyses of the former showed no significant variation in radial oxygen concentration. This agrees with earlier measurements on a sintered $\text{UO}_{2.00}$ fuel core. Analyses of the irradiated $\text{UO}_{2.05}$ samples are in progress.

Pneumatically impacted UO_2 was much less machineable than the high-density sintered pellets initially used in developing drilling techniques. Also, the higher the excess oxygen content of the UO_{2+x} specimens, the easier the material was to drill. Whether this is attributable to gross effects such as laminations and random cracks or to more intrinsic material characteristics such as greater plasticity is not yet known. Hardness measurements and metallography, which are in progress, may provide answers for these questions.

Stoichiometry Effects on Grain Growth of Pneumatically-Impacted UO_2

Measurements of grain-growth kinetics for pneumatically impacted UO_2 as a function of stoichiometry were initiated. These data are needed

to correlate grain structure with temperature in irradiated specimens. The procedure employed is to double-vacuum-encapsulate oxide specimens in platinum and quartz envelopes and then anneal for various times at temperatures between 1200 and 1700 °C. This temperature interval should be adequate for purposes of this experiment since no grain growth occurs below 1200 °C, and a thermal-gradient-dependent mechanism which gives rise to columnar grain structures becomes rate controlling above 1700 °C.

Actinide Oxide Phase Studies

Guinier diffraction patterns of several heat-treated americium oxides with compositions near the sesquioxide exhibited several weak lines in addition to the cubic and hexagonal phases. These lines were indexed satisfactorily on the basis of a monoclinic structure isomorphous with the B-type rare earth sesquioxide structure. This implies that Am_2O_3 exists as all three rare-earth oxide modifications (as does Pm_2O_3). Annealing experiments are in progress in an effort to transform the Am_2O_3 completely to the B-type monoclinic structure.

High-Frequency Induction Heating of Oxides

An oxide-susceptor induction furnace capable of high-temperature operation under large oxygen partial pressures was successfully developed. The furnace comprised a 1-1/2 in. ID, 1/4 in. wall CaO-stabilized ZrO_2 tube imbedded in fused ZrO_2 grain for thermal insulation. This tube was enclosed in a fused silica tube about which a 5 in. diameter, 4 in. long induction coil was wound. The furnace was preheated by coupling to a SiC rod inserted in the ZrO_2 tube. Above approximately 1600 °C, the ZrO_2 was sufficiently conductive to couple the 10^7 cps field and the SiC was withdrawn. The furnace operated at 2050 °C for several hours and survived numerous thermal cycles. A maximum axial temperature gradient of 75 °C/in. was measured in the 3 in. long, 1-1/2 in. diameter hot zone. No temperature stability or control problems were apparent.

The maximum temperature for prolonged operation of this furnace appears to be 2050 °C. At higher hot-zone temperatures, internal melting of the ZrO_2 susceptor occurred. Development of Y_2O_3 -doped ThO_2 susceptors, which should operate at substantially higher temperatures, is in progress.

Optical Properties of Single-Crystal ThO_2

Changes in the optical absorption spectra of single-crystal ThO_2 were observed after heating in hydrogen or air. Hydrogen-annealed ThO_2 exhibited absorption bands at 0.71 and 0.403 μ . Annealing in air removed these bands and produced an intense band at ~0.306 μ . The absorption edge for both specimens was 5.0 eV.

Growth of PuO_2 Single Crystals

A chemical method was worked out to determine the solubility of PuO_2 in sodium silicate glasses from which the controlled nucleation and growth of PuO_2 single crystals will be subsequently investigated. Several additional flux systems have been identified, and solubility measurements will be made to determine their usefulness in crystal growth.

Anion Self-Diffusion Measurements

A small cycloid-tube mass spectrometer is being placed in operation for use in anion-diffusion studies on uranium and plutonium ceramics. The diffusion measurements will employ the stable isotope exchange technique. The instrument will be checked initially for accuracy in measuring the isotope ratio $^{18}\text{O}/^{16}\text{O}$ to determine its suitability for oxygen-ion self-diffusion measurements on UO_{2+x} and PuO_{2-x} .

Plutonium Ceramics Irradiations

Eight of the nine remaining irradiation capsules containing undiluted plutonium compounds were removed from MTR and ETR because they had achieved exposures in excess of those anticipated in fuels thus far. The discharged capsules were stored in the MTR basin and will be returned for examination. Current status of the program is summarized in the table.

Plutonium Ceramics Irradiations

Capsules Discharged April 1, 1966

<u>GEH-14</u>	<u>Material</u>	<u>f/cc x 10⁻²⁰</u>	<u>MWd/ton Fuel</u>
405	PuN, Impacted	113	310,000
407	PuN, Impacted	55	149,000
411	PuO, 95% TD, Sintered	116	417,000
413	PuO, 95% TD, Sintered	90	321,000
538	PuN-50 vol% W, Impacted	77	165,000
540	PuN-50 vol% W, Impacted	43	93,000
549	PuN-24 vol%, Pu, Arc Cast	42	99,000
409	PuC, Arc Cast	92	251,000

Capsule Under Irradiation

<u>GEH-14</u>	<u>Material</u>	<u>Exposure, f/cc x 10⁻²⁰</u>		
		<u>Goal</u>	<u>During 3/66</u>	<u>Cumulative to 4/1/66</u>
551	PuN-15 vol% Pu, Arc Cast	32.0	1.0	20.0

Current plans call for leaving GEH-14-551 in-reactor until mid-1967.

Pneumatic Impaction: Bonding Mechanism Study

Specimens for the second pneumatic impaction series were fabricated and impacted, and examination of the impaction results is under way. These experiments are designed to investigate:

- Extent and character of bonding between 3 mm cubes of UO_2 single crystals, under a variety of conditions;
- Response and behavior of several matrix materials;
- Diffusion and bond influence of high-oxygen compounds at the single crystal interface;
- Fabrication behavior of $\text{W-UO}_{2\pm x}$.

The specimens were impacted without incident. The tungsten carbide punch fractured on first impact at 425,000 psi; ten subsequent impactations at the same pressure were conducted with tool-steel punches (upper punch: Type SKH-9, Vickers hardness ~ 700 ; lower punch: Type SKD-61, Vickers hardness ~ 425). The lower punch was slightly shortened by compression with each impactation, but punch fracture did not occur. Strain-gauge calibrations of the impactation conditions are under way to confirm impact pressures.

The specimens have been examined briefly by radiography and are currently being prepared for detailed study by microscope and microprobe. Radiography showed that the specimen cubes generally remained axially positioned and that only short-range movement of the pressure matrix powder occurred.

Pneumatic Impactation Vacuum System

The combination thermocouple and cold-cathode ionization gauge unit was received and is now being calibrated before installation on the vacuum system. The gauge measures from 5 to 10^{-6} Torr.

Several tests were run on the new vacuum system to determine the efficiency of different size evacuation tubes that are being used on pneumatic impactation billets.

A new approach to cutting off the evacuation tubes just prior to impactation has been tried with very good results on several billets. An evacuation tube is welded to the top lid near the edge at an approximate 30° angle. This allows the billet to be placed in the die with the evacuation tube intact. The tube is then sheared off by the impactation punch. This technique assures a good vacuum on the billet at impactation.

FFTF-Cermet Driver Fuel Evaluation

The mechanical properties of cermet fuel pins were determined for pins of varying UO_2 volume percents. The yield and ultimate strengths of a 20 vol% UO_2 -304 SS cermet were determined as a function of temperature.

The elastic modulus, yield strength, and ultimate strength of previously characterized cermets were determined. The mechanical testing conditions were as follows:

Sample Diameter	0.125 in.
Crosshead Speed	0.020 in./min
Extensometer	B3M microformer
Temperature	Ambient

The data obtained are as follows:

Vol% UO_2	Extruded			Swaged		
	Modulus, psi	Yield, psi	Ultimate, psi	Modulus, psi	Yield, psi	Ultimate, psi
0	21.5×10^6	39,000	92,600	20.8×10^6	92,700	108,000
10	16.1×10^6	28,900	56,500	14.2×10^6	71,500	78,600
20	17.0×10^6	30,000	42,000	15.3×10^6	65,400	79,700
30	15.3×10^6	27,000	27,600	8.74×10^6	40,000	40,650

Previous evaluation of the fuel pins showed excessive carbon pickup from the mild steel extrusion can. Carbon precipitation had occurred in some of the pins. The elimination of carbon from the fuel was achieved by using a copper extrusion can. The mechanical properties of the cermet extruded in this manner are shown below:

Vol% UO ₂	Extruded			Swaged		
	Modulus, psi	Yield, psi	Ultimate, psi	Modulus, psi	Yield, psi	Ultimate, psi
0*	30.6x10 ⁶	44,700	----	29.2x10 ⁶	119,000	----
20	25.8x10 ⁶	38,200	54,500	----	----	----
40	20.8x10 ⁶	25,200	25,600	----	----	----

*Pure 304 SS.

The yield strength and ultimate strength of a 20 vol% UO₂-304 SS cermet (in the as-extruded condition) was studied as a function of temperature. The data are shown below:

Testing Temperature, °C	Yield Strength	Ultimate Strength
22	38,200	54,500
400	24,500	37,000
720	16,500	23,000
800	16,000	18,700
940	6,500	7,300
1000	4,000	5,000
1100	3,000	3,200

Metallographic examination of the tested samples is in progress. Preliminary examination shows the UO₂ to be fractured when tested at temperatures between ambient and 900 °C. At temperatures above 900 °C fracturing of the UO₂ decreases and the stainless steel begins to pull away from the UO₂ particle creating voids.

Materials for Basic Research

Characterized UO₂ single crystal specimens were prepared and/or sent to Research and Development Center, General Electric Company and to the School of Mineral Engineering, Mining, Metallurgical, and Ceramic Engineering, University of Washington, for use in basic properties studies.

Deformation data were received from University of British Columbia on UO₂ single crystals previously supplied to them.

NUCLEAR GRAPHITE - R. E. Nightingale

Nuclear Graphite Irradiations

The long-term irradiation program for conventional nuclear graphite continues to progress satisfactorily. The present capsule, H-3-12, has completed 1-1/2 cycles of reactor operation with all thermocouples functioning properly. The binary-gas system is completely satisfactory and the maximum samples temperature is being maintained at 815 ± 10 °C.

Preliminary comparisons were made with UKAEA data obtained in the Dounreay Fast Reactor (DFR) and good agreement was obtained. The 800 °C CSF data (transverse samples) show nearly the same expansion rate as the PGA data from the DFR at temperatures of 300 to 700 °C. The parallel data are in even closer agreement, with the 800 °C CSF data falling among the PGA data points.

Irradiations of "Dimensionally-Stable" Graphites

The capsule, H-3-22, for the irradiation of "dimensionally-stable" graphites completed four cycles of irradiation in the GETR and was successfully disassembled the week of March 21. All 84 samples were recovered in excellent condition and have been returned for measurement and analyses. Length, coefficient of thermal expansion, and electrical resistivity measurements are presently being made. For all the samples the weight changes observed were less than 0.4%. However, for irradiation temperatures over 1000 °C the weight changes may become a problem so that double encapsulation of the samples is being examined. (See "High Temperature Capsule Modifications," following.)

Twelve of the 84 samples were extremely radioactive (10 to 30 R/hr at the surface) and will be measured at a later date. In addition, two were accidentally broken during measurement. Of the 70 remaining samples, 58 are being prepared for installation in the H-2-23 capsule for reirradiation. The capsule will also contain two new samples of the old geometry and 52 0.250 in. diameter rod samples contained in four "telephone dial" sample holders. Both these holders and the rod samples will be measured so that dimensional-change data can be obtained on specimens up to 1-3/4 in. in diameter. The four holders are made with different gas gaps to provide information on in-reactor heat-transfer characteristics of the capsule geometry. The H-3-23 capsule is scheduled for installation in the reactor late in April.

High-Temperature Capsule Modifications

The proposed design for future high-temperature capsule assemblies resembles a "telephone dial" in cross section. Heat-transfer calculations and laboratory tests indicate a temperature control range of 300 to 500 °C is possible using a binary gas system of helium and nitrogen at a gamma heating of 10 W/g. Elevated sample temperatures should be easily attained with this construction, 1500 °C at 10 W/g gamma heating being a representative value. Four such assemblies in H-3-23 will furnish experimental data on this design. Carbon felt will also be wrapped around two of the holders to test its effectiveness in reducing thermal radiation in a capsule environment.

As higher temperatures are reached in the irradiation capsule, oxidation of the graphite samples from impurities in the temperature-control gases becomes a crucial problem. Consideration is being given to using a doubly-encapsulated system for the irradiation assemblies at very high temperatures. The samples would be sealed in an inert atmosphere in the inner tube, and temperature control would be maintained by varying the heat transfer through the mixture of two gases between the two tube walls.

Effect of Irradiation on Coefficient of Thermal Expansion

Samples of a molded Texas-coke graphite heat treated at different temperatures were irradiated to 16,200 MWd/AT at about 600 °C. The observed length change and coefficient of thermal expansion (CTE) between 25 and 425 °C for transverse samples are shown in the following table. The unirradiated CTE is the average of two samples while the CTE at 16,200 MWd/AT is from one sample. The sample heat treated at 1370 °C reached a maximum contraction of 1.8% at 7000 MWd/AT before expanding to its present dimension. It is seen that higher heat-treatment temperatures cause smaller increases in thermal expansion. The CTE's of the samples heat treated at 2570 and 2700 °C actually decreased upon irradiation. Another effect is that contractions greater for lower heat-treatment temperatures except for the 1370 °C sample which had already undergone "turnaround". The table also shows that the crystallite size is smaller with lower heat-treatment temperatures.

Heat Treatment Temperature, °C	%ΔL/L 16,200 MWd/AT	X-Ray Crystallite ₀ Size (Lc), Å	CTE, 10 ⁻⁶ in./in./°C	
			Original	16,200 MWd/AT
1370	+1.7	45	4.8 ± 0.2	11.7
1800	-2.7	75	3.6 ± 0.2	4.5
2100	-1.9	200	3.6 ± 0.2	4.2
2300	-1.1	300	3.1 ± 0.3	3.6
2520	-0.3	380	4.0 ± 0.4	3.4
2700	-0.4	380	3.3 ± 0.2	2.95

Thermal Expansion in Polycrystalline Graphite

Previous attempts to correlate the thermal expansion of polycrystalline graphite with the single-crystal expansion coefficients have necessitated the introduction of an empirical c-axis accommodation coefficient.^(1,2) A method has been developed to calculate theoretically the thermal expansion of polycrystalline graphite taking into account this c-axis accommodation.

The method was applied specifically to two samples studied by Price and Bokros.⁽²⁾ The results obtained are compared in the following table with previous theoretical and experimental values of the CTE in the longitudinal (α_{11}) and transverse (α_1) directions.

(1) A. L. Sutton and V. C. Howard. J. Nucl. Mater., vol 7, p. 58, 1962.

(2) R. J. Price and J. C. Bokros. J. Appl. Phys., vol 36, p. 1897, 1965.

Experimental and Theoretical Values of the Coefficients
of Thermal Expansion

	CTE of Molded Graphites, 10^{-6} C^{-1}		CTE of Highly-Oriented Molded Graphites, 10^{-6} C^{-1}	
	α_{11}	α_1	α_{11}	α_1
Experimental	7.0	2.7	14.7	1.4
Theoretical (no accommoda- tion)	11.8	8.1	17.4	5.3
Theoretical (accommodation)	6.6	3.6	14.3	1.6

The present analysis is a substantial improvement over previous theoretical values which did not take accommodation into account. The amount of *c*-axis accommodation as a function of the crystallite orientation predicted by this method lies within the experimental limits given by Price and Bokros.⁽¹⁾

Because of the complex nature of polycrystalline graphite, it is unlikely that a single mechanism is responsible for all the variations found in the values of the coefficients of thermal expansion. However, the fact that a model based upon *c*-axis accommodation can give such good agreement with experimental values suggests that microcracks formed in cool-down from graphitization temperature are important in determining the thermal expansion of polycrystalline graphite.

Fracture Mechanisms in Graphite

Selected areas on the inner diameter of a cylindrical graphite sample were studied under compressive stress using a metallographic microscope. Although the compressive stress was concentrated by the introduction of a notch, the sample failed first in regions of tensile stress. The existence of prefailure cracks was not as pronounced as seen previously in regions under tensile stress. However, cracks appear to form parallel to the applied stress in particles aligned with their *c*-axes perpendicular to applied stress.

Electron-Microscopy Studies of Gas-Graphite Reactions

An investigation of the microstructural factors involved in gas oxidation of polycrystalline graphites is about to get under way. The initial phase of this work will be carried out using transmission specimens in the electron microscope. The gas-graphite reactions are to be studied outside the microscope initially. However, the reactions will be observed as they occur, in the microscope, when the appropriate microscope stage accessories have been developed.

(1) R. J. Price and J. C. Bokros. J. Appl. Phys., vol. 36, p. 1897, 1965.

A necessary preliminary to this study is the preparation of suitable transmission specimens of polycrystalline graphite. To be satisfactory for gas-reaction studies, such specimens must meet some rather rigid requirements. It is desirable to be able to take specimens from any specific location and orientation in bulk pieces of graphite. The specimens must be of the order of 1000 Å thick at the point to be examined and must maintain all structural features, including kinds and distributions of impurities, characteristic of the bulk material from which they were taken. Such ideal specimens have never been produced. For this reason, considerable initial effort is being devoted to this subject.

The present procedure being investigated for specimen preparation is a three-stage process involving mechanical machining, spark machining, and chemical machining. For the mechanical machining step, small diameter graphite rods are made from bulk stock on a lathe, employing techniques to minimize the formation of defect structures and microcracks in the material.

Next, a conical depression is spark-machined into the end of the rod employing a gaseous dielectric with appropriate cooling of the specimen. The spark-machining process is carried out with a graphite tool and is believed to produce much less disturbance of the original structure than the equivalent mechanical machining operation. Finally, a thin wafer section containing the spark-machined depression is cut from the graphite rod and subjected to chemical machining to produce the microscope specimen. The chemical machining step is performed by treating the wafer section with oxygen which has been excited with a microwave discharge. The intent of this process is to produce active oxidizing species which are completely nonselective in attacking a graphite surface. This process is used to remove enough material from both top and bottom surfaces of the graphite wafer until a small hole is created at the bottom of the conical depression. The electron microscope will then be used to examine the thin graphite area near the edge of this hole.

The Effect of Oxidation on the Sonic Modulus of TSX Graphite

In determining the effect of the oxidation rate on the decrease in sonic modulus of TSX graphite with extent of oxidation, given samples have been alternately oxidized at two different temperatures in order to avoid differences between samples. With the use of widely different temperatures, however, the possibility arises that changes in the sample resulting from oxidation at one temperature will influence subsequent effects occurring upon oxidation at the alternate temperature. For example, oxidation at temperatures above about 850 °C results in an increased specific surface area over the external regions of the sample, because at these high temperatures, oxidation is confined to the outer portion of the sample. Subsequent oxidation at a lower temperature is then influenced by this externally-located, high specific surface area, and consequently the resultant modulus loss is less than it would have been had the low-temperature oxidation proceeded in its normally homogeneous fashion. Alternately, the low-temperature oxidation probably influences the results of the high-temperature oxidation.

Experimentally, it has been found that when a sample is alternately oxidized at temperatures of 650 and 780 °C, a logarithmic plot of the relative modulus versus the fractional residual weight yields a straight line.

If the upper temperature is above 780 °C, the plot shows curvature. It has now been found that when a sample is oxidized solely at 780 °C some curvature is exhibited, presumably because in the absence of the lower temperature oxidation the internal surface area is not as readily available at 780 °C. In the case of a sample oxidized solely at 750 °C, a linear plot is again obtained, and hence, 750 °C appears to be about the maximum temperature at which the modulus loss is independent of the oxidation rate.

A series of parallel samples have been oxidized in air at 700 °C to varying extents up to about 15% weight loss. Measurements of their sonic moduli have been completed, and measurement of mechanical moduli and ultimate tensile strength is planned.

IRRADIATION DAMAGE TO REACTOR METALS - A. L. Bement

Alloy Selection

The liquid-metal, temperature-controlled capsule, irradiated in the ETR core during Cycle 78, was opened in Radiometallurgy. Examination of capsule components revealed that loss of capsule temperature monitoring midway in the operating cycle resulted from thermal fatigue of the thermocouple cladding.

A program is being initiated to evaluate the ATR Gas Loop specimen container under operating conditions provided by the model demonstration loop. A separate program using the model loop to screen nickel-base and refractory alloys for future irradiation damage studies in the ATR Gas Loop is being planned.

In-Reactor Measurements of Mechanical Properties

A creep specimen of annealed 304 SS is accumulating irradiation prior to starting an in-reactor creep test at 840 °C and 5000 psi. The out-of-reactor control creep test at these conditions ruptured after 926 hr and exhibited a strain of 26% total elongation at rupture. Minimum creep rate during the test was 1.76×10^{-4} in./in./hr.

Out-of-reactor tests at 9000 °C, 5,000 psi and 700 °C, 10,000 psi ruptured after 96 and 10,630 hr, respectively. Rupture strain of both specimens was about 28%. The specimens were submitted for metallography.

In-reactor tests on heat treated Zr-2.5 wt% Nb are in progress at 300, 350, and 400 °C. A creep rate of 6.5×10^{-7} in./in./hr has been established for the 300 °C, 55,000 psi in-reactor creep tests. Creep rates of 3.0×10^{-7} in./in./hr and less than 1×10^{-7} in./in./hr were obtained on this same specimen at previous loads of 45,000 and 36,500 psi, respectively. No creep rate has been obtained for the control test at 55,000 psi. The 350 °C, 45,000 psi test is continuing. The erratic behavior of the extensometer circuit has been corrected and a creep rate of 1.4×10^{-6} in./in./hr determined. The out-of-reactor creep rate at the same conditions was 2×10^{-6} in./in./hr.

The 400 °C test is currently accumulating irradiation at temperature. A stress of 36,500 psi will be applied at a later time. An out-of-reactor control creep test at these conditions ruptured after 4,025 hr and a total

elongation of around 11.5%. Third stage creep began after about 1200 hr at around 1.3% strain. The minimum creep rate was 8.3×10^{-6} in./in./hr.

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 series of tensile tests have been performed on AISI 304 and 348 SS irradiated at an average temperature of 650 °C in an inert gas environment. Tensile tests were conducted at several temperatures between room temperature and 750 °C on both round and sheet type specimens. The samples tested received a total integrated fast neutron exposure of approximately 1.0×10^{20} nvt. Postirradiation annealing treatments were performed on selected samples to study the recovery of damage caused by high temperature irradiation.

The results of these tests show that the strength of the materials at all deformation temperatures is not affected significantly by irradiation at 650 °C. The ductility is reduced, however, particularly at test temperatures of 600 °C or above. These results are in good agreement with tests performed previously on AISI 348 SS irradiated at 750 °C and tested over the same range of test temperatures. The greatest reduction in ductility occurs at test temperatures between 500 and 750 °C and is emphasized by the fact that the tensile specimens fail at points where the extensometer was attached.

The tensile ductility in the intermediate deformation temperature range between 200 and 400 °C is not significantly reduced by these high temperature irradiations. However, previous tests on these materials have verified that irradiations at temperatures below about 400 °C can cause a drastic reduction in ductility at the intermediate test temperatures between 200 and 400 °C.

Nickel-Base Alloys

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

Transmission microscopy of Hastelloy X-280 tensile specimens has continued in an attempt to understand the influence of irradiation damage on microstructural modifications. Comparisons have been made of the stressed sections of control and irradiated specimens which had been given 20% cold work followed by an anneal at 1000 °F (540 °C) for 24 hr before irradiation. The control specimen showed precipitation induced by tensile strain at 1350 °F (730 °C) in the form of rectangular particles which are primarily

$\leq 400 \text{ \AA}$ in size. Particles in the irradiated specimens appeared to be fewer in number and smaller in size. This reduction in number and size of precipitate particles in the irradiated material indicates that less of the strengthening elements is removed from solution. This may account for the increased strength of the irradiated material compared with the control material (68,000 versus 55,000 psi for 0.2% YS, 75,000 versus 58,000 for UTS).

Fracture Mechanics

The fracture toughness of annealed Zircaloy-2 irradiated to an approximate neutron exposure of 1×10^{21} neutrons/cm² in the ETR G-7 loop was determined last month. The fracture toughness, K_{Ic} , of both RD and TD specimens was essentially the same as that determined for exposures of 3×10^{20} and 6×10^{20} neutrons/cm² previously. It appears, therefore, that saturation of neutron damage occurs at an exposure level of approximately 3×10^{20} neutrons/cm². Anisotropy in fracture properties found in unirradiated specimens was observed to be nearly absent in the irradiated material.

In order to measure the ability of a material to arrest a running crack, the stress intensity factor at crack arrest (K_{Ia}) was measured for both irradiated and unirradiated Zircaloy-2. The variation of K_{Ic} - K_{Ia} was determined as a function of both temperature and irradiation. Specimens oriented such that cracking occurred transverse to the rolling direction exhibited no systematic variation of K_{Ic} - K_{Ia} with neutron exposure, but did show a general increase of K_{Ic} - K_{Ia} with temperature for each irradiation level.

Specimens with cracks propagating in the rolling direction exhibited both an increase of K_{Ic} - K_{Ia} with temperature and a trend toward lower values of K_{Ic} - K_{Ia} with increasing neutron exposure. These results indicate that with increasing temperature crack jumps will be longer in both irradiated and unirradiated specimens. Also, at temperatures above 20 °C, cracks will jump further in the rolling direction than the transverse direction after neutron exposures of 6×10^{20} and 1×10^{21} neutrons/cm².

The effect of cold work recovery on the fracture toughness of Zircaloy-2 was determined. Specimens of Zircaloy-2 containing 20 and 40% cold work were annealed in vacuum at 475 °C for 100 and 1000 min and tested at 0 and -40 °C after each anneal. The results of the annealing treatment were to cause an increase in the fracture toughness at both test temperatures followed by a slight decrease in toughness after the 1000 min anneal. This behavior was observed for both 20 and 40% cold worked material along both orientations. Recovery after 100 min at 475 °C was much greater for the TD than for the RD orientation, but complete recovery did not occur for either orientation.

Work on hydrided Zircaloy-2 continued with the testing of specimens charged with 400 ppm hydrogen. Fracture toughness in the transverse direction decreased with increasing test temperature in the range -120 to -20 °C above which it increased. The rolling direction specimens did not exhibit this reversed temperature dependence, but remained nearly constant up to 20 °C and then showed an abrupt increase above this temperature. Low temperature anisotropy in fracture toughness was observed in specimens

containing 400 ppm hydrogen, but not in specimens containing 100 ppm hydrogen. The results of these tests are to be correlated with metallographic and chemical analysis of the material.

Advanced Metallurgy Studies

High Temperature Applications

A literature survey was continued to determine operating temperatures and structural material requirements for reactors presently operating, and under construction, or under study. This information is being used to assess future materials requirements and possible materials problem areas in support of the ATR Gas Loop program.

Fast Neutron Applications

A permeability apparatus that minimizes end-effect errors has been designed and is being assembled for hydrogen effusion studies. The elimination or control of rate affecting reactions at the hydrogen absorption surface is of particular importance for cladding and other relatively thin specimens. Work on this problem has yielded a number of possible solutions. A few, including the photosensitized dissociation of hydrogen and the use of coatings to modify surface catalytic activity, appear quite promising and will be investigated experimentally.

Preliminary studies have been undertaken to develop suitable physical entrapment techniques that can be used to introduce known quantities and distributions of inert gases into alloys for subsequent gas behavior investigations. An alternate approach has also been devised that will use internally precipitated, gaseous, interstitial compounds as inert gas bubble analogues.

High Pressure Studies

The article, "Exploration of the Combined Effects of Ultra-High Pressure and Nuclear Irradiation on Solid-Solid Phase Transformations," has been accepted for publication in the July issue of the Journal of Applied Physics.

Preliminary calculations on the increase in displacement energy due to pressure indicate that barium is the most suitable element to study. Cesium, rubidium, and potassium show larger pressure coefficients but are more difficult to handle experimentally. Using a simple model, the displacement energy of barium is predicted to increase by a factor of 7 under 20,000 atm (300,000 psi). This is mainly due to barium's large initial compressibility and lattice parameters.

An increase in the displacement energy should decrease the total number of atoms displaced under irradiation. Counteracting this decrease in number of defects produced is the increased retention of existing defects due to their decreased diffusion coefficient under pressure. This means that a solid which is irradiated and compressed at the same time should retain more damage after irradiation than one which is irradiated at 1 atm. The result is analogous to irradiation at cryogenic temperatures, because in both cases the lattice parameters are reduced and the diffusion coefficient lowered.

The pressure "healing" of vacancy clusters and voids has been investigated. It is concluded that rather low pressure (<1000 atm) should "heal" microscopic clusters down to a few angstroms across and prohibit the formation of vacancy clusters in solids which are irradiated and compressed at the same time. This means that if two foils are irradiated to the same exposure but under different pressure (i. e., 1 and 1000 atm), electrical resistivity experiments would indicate more damage in the specimen under high pressure, but electron microscopy would suggest the reverse situation. In one case, under high pressure, a finely dispersed distribution of defects will be produced. In the other case, under low pressure, the same number of defects could be produced but allowed to collect in complex configurations. Experiments are being devised to test these predictions in molybdenum.

Hydrogen Production by (n, p) Reactions in AISI 304 SS

A study has been initiated to determine the method and rate of hydrogen generation by (n, p) reactions in AISI 304 SS for a variety of reactors. This hydrogen gas may have an adverse effect upon the mechanical properties of the metal. A knowledge of the expected rates and levels of hydrogen generation will permit a more thorough analysis of the behavior of hydrogen in the 304 SS.

The reactors to be analyzed are the ATR, ATR (with converter), EOGR, and the FFTF. These four reactors will provide representative hydrogen-production values for thermal, mixed-spectrum, and fast-reactor systems.

The constituents of the 304 SS that are believed to be important sources of hydrogen due to their abundance and cross section are Fe, Ni, Cr, N, S, and Si.

The methods employed for the determination of the hydrogen generation will be similar to those used for the calculation of helium production by (n, α) reactions. The results of the hydrogen (n, p) study should provide a more accurate account of the gas production than did the helium (n, α) study because there are more energy-dependent cross section data available for the (n, p) reactions.

Fast-Neutron Dosimetry Study

A joint experiment has been initiated between Atomics International (AI) and Battelle-Northwest to test methods of neutron-spectra determination and fast-neutron dosimetry. This experiment is intended to test and improve the dosimetry and damage correlation methods used in material irradiations. Correlation of results of different neutron-damage studies depends on obtaining a reliable description of the neutron flux spectrum in different reactor environments.

This joint study consists of: (1) calculating the neutron spectra in a graphite-lattice test position in a Hanford reactor by current neutronics codes; and (2) using the derived spectral form to calculate values of saturated

activities (or average cross-sections) on a number of foil materials that were irradiated in the position. The calculated activities are then compared with the measured foil activities.

Advanced foil-activation methods are being developed for experimental determination of neutron-flux spectra. McElroy, Barrall, and Ewing (AFWL-TR-65-34 vol 1) have developed a method that will provide continuous information over the entire neutron-energy range. It has been anticipated that the experimental results will be accurate to within ± 10 to 30%; therefore, the integral neutron flux will be specified at all energy points over the range of 4×10^{-7} to 18 MeV to within an accuracy of ± 10 to 30%. The theoretical results are expected to agree with a similar accuracy.

Another important aspect of this cooperative effort will be to provide data on foil-activation counting procedures. In addition to work at AI and BNW, R. Armani at ANL and J. Walker at Sandia have supplied and will count the foils they have provided for this test.

Progress of the cooperative project to date is as follows. All foil packs have been irradiated and sent to the respective counting sites for activity measurements. The foil activation measurements are completed at all sites, and the data have been forwarded to W. N. McElroy for assimilation and reduction.

The spectral determination of the irradiation facility has been completed using a 21-energy-group transport code with fine energy structure in the fast-neutron region (16 quarter-lethargy groups above 0.183 MeV). This calculation has been made with varying degrees of angular complexity. The first case was calculated using two azimuthal angles and two polar angles (S_2P_2). The angular complexity was increased to S_4P_2 for the succeeding calculation with an increase of 26% for the flux (Φ) $E > 1$ MeV, 21% for the flux (Φ) $E > 0.5$ MeV and an 18% increase in the flux (Φ) $E > 0.18$ MeV. Thus, the increase in integral flux and the spectral shift (indicated by the varying increases in the integral flux from S_2P_2 to S_4P_2) are a function of the angular complexity of the transport calculation. The angular detail of the calculation will be increased until the differences between the runs become negligible. The calculated average cross sections for the $Fe^{54}(n,p)Mn^{54}$ reaction for $E > 1$ MeV, $E > 0.5$ MeV, and $E > 0.18$ MeV for the S_2P_2 case was 92.1 mb, 61.0 mb, and 44.9 mb, respectively; whereas, the same values for the S_4P_2 cases were 111.6 mb, 76.7 mb, and 58.0 mb.

Comparison between calculated and measured foil activities have not been completed; however, W. N. McElroy has indicated that preliminary data reduction indicate that results look promising. Complete information will be forthcoming in a paper to be published jointly by BNW and AI concerning this project.

ATR GAS LOOP OPERATION AND MAINTENANCE - G. A. Last

ATR Gas Cooled Loop

The contract submittals on the ATR Gas Loop received through April 15, 1966 have been reviewed, and comments transmitted to Ebasco Services. Among the major equipment items that have been reviewed are the components for the in-reactor tube, circulator control, and the gas purification system.

A design review meeting was held at Ebasco Offices in New York on April 4, 5, and 6, 1966. This meeting was attended by PNL, PPCO, AEC-IDO and AEC Headquarters personnel. Discussions included loop design, procurement, and research and development programs supporting the Gas Loop construction. The remaining Ebasco specifications and drawings were reviewed and approved. In addition, a meeting was scheduled at PNL for the week of April 25, 1966, to be attended by PPCO, AEC-IDO, and Ebasco Services, to incorporate all necessary phases of construction and postinstallation equipment tests into Ebasco's System Operational Test Specifications.

Model Gas Loop

The new scale model heater for the gas loop has been installed and operated under varying flow, pressure and temperature in order to establish some basic operating characteristics and to test out the new heater operation. The new heater has operated a total of 67 hr with the maximum loop conditions being 2100 °F (1150 °C) heater outlet at 300 lb/hr helium flow rate. The heater has been through ten thermal cycles from room temperature to varying heater outlet temperatures. Power inputs of 60 kW on the Nichrome heater section and 50 kW on the molybdenum section were used to attain the 2100 °F heater outlet temperature. One full flow run of 500 lb/hr with a heater outlet of 2000 °F (1100 °C) was made. The transformer taps for the moly heater section will be changed to allow a heater outlet of 2100 °F to be obtained at the 500 lb/hr flows. Additional testing is scheduled to determine the basic characteristics of the new heater and the metallic foil insulation.

METALLIC FUELS DEVELOPMENT - J. E. Minor

Chemonuclear Fuel Development

The chemonuclear fuel concept requires a fuel in the form of a very thin honeycomb whose web thickness is 0.000120 in. (3.0 μ) thick, and a core of 20% U-80% Pd, 2.5 μ thick. These honeycombs will be fabricated by electron beam vapor depositing the cladding and core alloy onto hexagonal rods, then extruding the bundled rods together and finally etching out the sacrificial rods to leave the honeycomb.

Since the honeycomb will require the coating of more than 300 rods, some method of coating which will finish the maximum number of rods per batch is necessary to ensure good uniformity in coatings and to put the time and effort required into the realm of practicality. Toward this end a device has been designed and built which operates inside the vacuum system and will rotate up to 48 rods at a time through the vapor stream while at the same time rotating each rod on its own axis. Special connectors were designed to allow quick changing of rods and a heater was designed for baking the rods at temperatures up to 400 °C during plating. Special bearings and new low vapor pressure solid lubricants, such as the recently developed tungsten diselenide, allow the device to operate completely oil-free and to be vacuum baked for maximum cleanliness.

Irradiation of Thorium-Uranium Fuel Elements

The irradiation of six tubular Zircaloy-2 clad Th-2.5 wt% U, 1.0 wt% Zr fuel elements continued successfully in the ETR-P7 and M-3 loops. The fuel element having the highest exposure, 13,425 MWd/ton, has shown a fuel volume increase of 2.4% and is currently operating at a maximum core temperature of 530 °C. This volume increase is no greater than that required to accommodate the solid fission product atoms.

One fuel element which was damaged in handling was returned to Battelle-Northwest for examination. This element has an exposure of 11,300 MWd/ton or approximately 1.3 at.% burnup. Fuel swelling determined by bulk density measurements is 1.4%. Postirradiation measurements show that a 0.007 in. increase in diameter and a 0.084 in. increase in length occurred during irradiation. Except for the minor mechanical damage at one end cap, the fuel element appeared to be in excellent condition. This element will be subjected to defect testing in a pressurized water loop and destructively examined in Radiometallurgy.

ENGINEERING DEVELOPMENT

UPSTREAM BOILING BURNOUT - D. R. Dickinson

Two groups of laboratory experiments were performed on a uniformly heated test section 0.435 in. ID and 12 ft long electrically heated. The first group of tests was run with water at a mass velocity of 3×10^6 lb/hr-ft² and a downstream pressure of 1500 psig. As the inlet quality was increased from 0 to 50%, the burnout heat flux decreased from 360,000 to 250,000 Btu/hr-ft². The burnout location moved progressively from the downstream end to a point 3 ft from the upstream end as the inlet quality was increased to 30%, but did not move any closer to the upstream end with further increases in quality. This behavior is generally similar to that previously observed with the 6 ft long test section. Of particular significance, upstream burnout was not accompanied by a rise in burnout heat flux with increasing quality as had been found at higher mass velocities.

Upon attainment of boiling burnout conditions, further increases in heat flux resulted in only moderate wall temperature rises; therefore, it was possible to continue the test beyond the first indication of burnout. When the heat flux was increased over the burnout value, the region of burnout spread until, if the test were continued far enough, it extended over most of the length of the tube; in several cases, the point of strongest burnout indication (highest wall temperature) shifted upstream. The test section had rings of thermocouples 90° apart at the midpoint and at the downstream end. These showed a very large circumferential variation in the burnout heat flux (as great as a factor of 1.7). Earlier, similar variations had been found at the downstream end of the 6 ft test section. This is a major unexpected and unexplained phenomenon encountered in these tests. The last test (50% inlet quality) was continued to a heat flux of 700,000 Btu/hr-ft², or 2.8 times that at the first burnout indication. In this test, 100% outlet quality was attained. At the conclusion of this test, the inlet quality was gradually reduced from 50 to 10% with the heat flux held constant at 540,000 Btu/hr-ft². Initially, almost the entire test section was in a postburnout state. In general, the downstream wall temperatures fell somewhat, while very large increases were noted near the upstream end.

The second group of tests was run with water at a mass velocity of 7×10^6 lb/hr-ft², downstream pressure 1500 psig, and inlet conditions ranging from slightly subcooled to 15% quality. In these tests, the burnout heat flux remained constant with increasing enthalpy. The burnout location moved upstream until it reached a point 3 ft from the upstream end. Then, on further increases in enthalpy, the burnout location remained stationary, but the burnout heat flux rose from 1.07 to 1.28×10^6 Btu/hr-ft². This behavior is in general agreement with previous results with a 6 ft heated length.

NEUTRON FLUX MONITORS - G. C. Fullmer

Regenerating Detectors

Continued operational success of the regenerating concept is being demonstrated by the two experimental (U^{234} - U^{235}) detectors installed in a test facility of the KW-Reactor. The concept is based upon the replenishment of fissionable material (U^{235}) through neutron absorption in fertile material (U^{234}). Fabrication is planned of two advanced-model chambers to further demonstrate the concept and to establish final-form configuration.

Beta-Current Generator Detectors

Following gamma irradiation tests of the assembled dual-chamber, twin-lead detector, for which a gamma sensitivity of about 5×10^{-17} A R⁻¹ hr⁻¹ was measured for both the B^{11} chamber and the carbon chamber, the complete experimental assembly was carefully checked and then installed in a test facility of the KW Reactor. Prior to the reactor installation, several minor air leakage problems were noted and corrected; this permitted successful evacuation of the assembly. Initial data obtained from the reactor test indicated that the low thermal neutron cross-section of the B^{11} , estimated to be about 5 mb, will definitely pose considerable difficulty in achieving successful operation of the self-powered chamber using this isotope. The dual-chamber concept, where carefully calculated amounts of B^{11} and pure carbon are used to obtain an equal background response, (0.979 and 0.890 g respectively in the experimental assembly), appears to be necessary if B^{11} is to be useful as a long-life, in-core, neutron flux detector. Careful selection of the assembly materials (pure iron shell and quartz capsules) was used to minimize buildup of reaction products.

The initial in-core tests at the KW Reactor resulted in measured currents in the nanoampere range for both the B^{11} and the carbon chambers; however, more complete testing was precluded by a scheduled outage of the reactor. Considerable further testing and development will be required before it can be ascertained if the B^{11} concept has sufficient merit for the intended use.

Microwave Detectors

Developmental progress was achieved on the microwave concept for measurement of in-core neutron flux, where a special resonant cavity containing a neutron-absorbing gas is employed along with appropriate waveguides to permit measurement of microwave frequency changes related to the surrounding neutron flux density. Because of a marked reduction in the frequency shift when the neutron-absorbing gas is contained in a quartz capsule inside the

resonant cavity, development was aimed toward the design of specially-sealed cavity to eliminate the need for the capsule. The required design, employing dielectric sealing, was established, and fabrication bids are being solicited from microwave component manufacturers. Plans were made to acquire four cavities; two will be filled with He^4 gas and two will utilize other absorbers to provide enhancement of the plasma-forming process and hence a greater shift in frequency.

CRITICAL FLOW AT HIGH TEMPERATURES AND PRESSURES -

F. R. Zaloudek

Experiments were performed to study critical flow from a 3 ft long section of 2 in. schedule 80 pipe. This piece of pipe was provided with wall pressure taps at 0.025, 0.060, 0.090, 0.125, 0.250, 0.500, 1.00, 2.00, and 4.00 in. from the discharge end. The experiments were performed by the discharge of high pressure water from a 1000 gal, 3000 psi pressure vessel through the test section to atmosphere. During the blowdown, the pressure in the vessel was maintained at a constant value by the introduction of air into the tank. The pressures along the test section was measured at the wall, and these pressure indications were extrapolated to the discharge end to find the "critical pressure." Preliminary analysis of the data revealed that at a water enthalpy of 505 Btu/lb, critical pressures from 270 to 320 psia were obtained for total discharge rates of 3700 to 4300 lb/min. These data appear to agree well with earlier results obtained with smaller test sections.

MICROWAVE DETECTION OF COOLANT IMPURITIES AND MEASURE-

MENT OF IN-REACTOR TEMPERATURES - G. C. Fullmer

Microwave Techniques

A frequency-stabilizing circuit was developed for use with the commercial frequency generator in order to improve the sensitivity of the microwave technique for measuring moisture in reactor coolant gas. Initial tests indicated a long-term drift stability limit of about 50 kHz; this improves the moisture detection level to about 700 ppm, a factor of 10 improvement. Theoretically, the detection sensitivity is limited only by the frequency stability and by the frequency-change measurement ability. Thus, the present effort is directed toward improving the stability and the measurement technique.

Infrared Techniques

Successful laboratory tests were achieved with the assembled infrared analyzer system, which employs a quadrant filter mechanism to obtain selective attenuation of specific infrared frequencies. An invention report was submitted regarding the concept.

Further calibration work demonstrated that proper operation was achieved with the experimental apparatus for producing gases with known moisture concentrations. The apparatus will be employed in both infrared and microwave moisture concentration measurement experiments.

PLUTONIUM RECYCLE PROGRAM - F. G. Dawson

FUELS DEVELOPMENT

PRTR Fuel Element Performance

Postirradiation examination of fuel rods from FE-6504, which ruptured during irradiation in the FERTF, is essentially complete. Fuel Rod FB-55, which was two positions distant from the ruptured rod and which exhibited the most severe external evidence of localized overheating in the top end region of the four overheated rods, was examined.

Due to the manner in which Rod FB-55 was removed from the element, it was not possible to determine the postirradiation balance point. Measurements along the length of the rod, however, show that the diameter has not changed as a result of irradiation. The postirradiation fuel density was determined for three axial positions along the length of the fuel rod. The fuel density was determined in rod segments approximately 2 in. long at axial positions 20 in. from the top, 30 in. from the top, and 20 in. from the bottom of the rod. The amount of fuel in each segment indicates that the fuel density in the upper segment of the rod has decreased approximately 12% whereas the fuel in the lower two segments has increased approximately 2 and 1%. The apparent mass imbalance is not completely understood; however, the results indicate that axial fuel relocation occurred in this rod which experienced considerable fuel melting during operation.

Examination of the transverse section taken 30 in. from the top of this rod, which is in the same plane as the defect in the ruptured rod, shows that fuel melting occurred to as much as 62% of the radius. A small central void at this axial position extends to approximately 17% of the radius. The large degree of melting and the high fuel temperatures that existed in this rod during irradiation are also reflected by the high fission gas release value of 88%.

Excluding the ruptured rod, postirradiation examination results show that melting at the plane 30 in. down from the top of the four nondefected rods irradiated under presumably the same conditions, ranges from 46 to 64% of the radius. These observations suggest that the physical behavior of vibrationally-compacted fuel operating in long fuel rods under high power generating conditions varies even in like rods exposed to the same irradiation conditions. Such behavior complicates the prediction of fuel structures formed in vibrationally-compacted rods during irradiation.

Examination of the fuel in the top of Rod FB-55 shows that the UO_2 -1 wt% PuO_2 graded enrichment became mixed with the UO_2 -4 wt% PuO_2 fuel material. The depleted UO_2 pellet is relatively unaffected although interdiffusion has occurred at the pellet-powder fuel interface. No central void formed in the top end region of this rod although fuel melting occurred to within 0.1 in. of the pellet. Unexpectedly high fuel temperatures in the top end region resulted in sintering and grain growth to the outer surface of the fuel. Molten fuel apparently flowed outward along radial cracks to possibly contact the cladding without causing any noticeable core-cladding reaction. Although the external appearance of the locally overheated cladding indicated a more severe condition in this rod, metallographic examination indicates that the degree of overheating is comparable to that observed in the other three

affected rods. The cladding was overheated to temperatures above 850 °C since beta transformed grains are present in the Zircaloy microstructure. Measurements indicate that the fuel column in this rod moved the depleted UO₂ pellet upward approximately 0.54 in. against the downward force of the plenum spring.

Molten Fuel Defect Test

Postirradiation examination of a transverse cross-section of an irradiated, intentionally defected UO₂ fuel rod indicated that the fuel was molten to approximately 20% of the radius.

Test GEH-12-39, conducted in the P-7 Loop in ETR, was designed to study the behavior of an intentionally defected, vibrationally-compacted, UO₂ fuel rod operating under molten-core conditions. The defect was 0.06 in. diameter hole through the PRTR-size Zircaloy-2 cladding.

The loop outlet water coolant temperature was 430 °F at 1600 psi. During the 35 day irradiation period the test was subjected to 13 reactor scrams from full power. Gamma activity bursts in the loop piping cubicle were greater than 100 R/hr; the initial steady-state gamma activity was 12 R/hr and decreased to 10 R/hr during the latter portion of the irradiation period. The test was designed to produce a maximum linear rod power of 26 KW/ft but generated only about 22 KW/ft due to lower than expected neutron flux in the loop.

FERTF Test Element Design

A special FERTF test element has been 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 UO₂-PuO₂ fuel rods. A prototype basket was fabricated and the following table outlines the schedule of planned tests with the prototype assembly.

<u>Date</u>	<u>Facility</u>	<u>Type of Rods</u>	<u>Test, Experiment or Study</u>
April 25-29	PRCF	UO ₂ -2% PuO ₂	Radial and Axial Flux Patterns, Reactivity Measurements
May 2-6	Thermal Hydraulics Laboratory	UO ₂	ΔP Measurements, Flow Splits
May-July	Mechanical Development Loop	UO ₂	Charge-Discharge Mechanical Stability, Fretting Corrosion

Cycle Analysis

Volume I of the study Comparative Costs of Oxide Fuel Elements has been completed. The title of this volume is "Computed Results for Fuel Elements Fabricated by Vibrational-Compaction or Pelleting with Uranium, Low Exposure, or High Exposure Plutonium Enrichment." The second volume on plant, equipment, and shielding design will be completed shortly. Work has started on Volume III, which is a parametric study of major cost variables.

PRTR Fuel Fabrication

The fabrication of twelve 2 wt% PuO_2 - UO_2 Mark I-R₂ PRTR fuel elements and twelve spare replaceable fuel rods was completed. Fifteen additional UO_2 fuel rods were completed. Twelve SS clad, aluminum-cobalt flux wires were assembled for use with the completed fuel elements. This work completes the fuel requirements for the High Power Density Batch Core Loading in the PRTR.

PCTR Particle Size Studies

The equipment for rounding PuO_2 particles has been installed in a hood having a dry air atmosphere.

A mechanism for separating round from nonround particles was designed, built, and tested. Results of these tests indicate that actual separation of the entire quantity of material would require an excessive amount of time. Consequently, we now plan to mill all particles until they are round before removing the size fractions desired.

The initial batch of PuO_2 (1 kg) that was bonded with polyvinyl alcohol has been used in an attempt to produce spheres. Less than 50 g of particles having the required size and acceptable sphericity were obtained. The remaining material was recovered as dust and will be reprocessed after low temperature calcination. Continued efforts are being made to provide a better yield in a shorter period of time.

FUELS REPROCESSING DEVELOPMENT

Characterization of High Exposure Plutonium

Recovery and decontamination of plutonium from three samples (Bundles 6, 14, and 45) of Shippingport Reactor Blanket fuels has been completed and the plutonium sent to Dosimetry Technology for beta-gamma emission measurements. Plutonium recovered was 5.0, 4.3, and 5.0 g for Bundles 14, 6, and 45, respectively. Fission product contamination, largely Ru-Rh^{106} , was less than 20 $\mu\text{c/g}$ plutonium for all three samples. Uranium contamination was less than 0.1 g/g plutonium for all three samples.

Documentation of the nonradioactive engineering development of the Salt Cycle process is nearing completion.

REACTOR PHYSICS

D₂O Moderated Systems

PRCF Experiments with 2 wt% Mixed Oxide Fuel

The PRCF core size was increased from 13 to 28 clusters of mixed oxide fuel. These two cores are the most nearly cylindrical when loaded into a triangular array. The boron concentration was increased from 4.8 to 11 wppm to compensate for the increased reactivity as the core size was increased. With 11 wppm the moderator level for critical was $\sim 1/2$ in. above the fuel meat.

The rod drop method was used to measure the combined reactivity worths of three safety rods. One safety rod and one control rod were measured separately.

To measure the variation of prompt neutron lifetime with increased boron concentration a recording of reactor noise was made on magnetic tape. The analysis of the noise will give a value of β/ℓ to compare to the same ratio measured with 4.8 wppm B¹⁰ in the D₂O moderator.

The reactivity temperature coefficient has been measured from 23 to 71 °C for the increased boron concentration. The coefficient varied from $+0.09 \pm 0.01$ mk/°C at 23 °C to 0.0 at about 64 °C.

Moderator level sensitivity measurements have been completed for the region above the fuel but additional excess reactivity is needed for measurements at lower moderator levels. The change in thickness of an infinite reflector from that in the 13 cluster core is ≈ 2 in.

Lutetium Activations and Spectral Indices

Spectral indices T and r, which characterize respectively the thermal component and the epithermal component of the neutron spectrum, were measured in the PRCF in a cell consisting of a Pu-Al 19-rod cluster moderated with D₂O. The indices were deduced from gamma-ray activities of lutetium foils. Similar measurements were made earlier in the PRTR. The results of these two experiments are compared in the following table.

Spectral Indices in a Pu-Al Cell as Measured
in the PRCF and PRTR

Position of Foil	$\overline{T}$ (°K) ^a	PRCF		$\overline{T}$ (°K) ^b	PRTR	
		r			r	
Center Rod	321 ± 3	0.0656 ± 0.0006		336 ± 13	0.069 ± 0.004	
Intermediate Rod	328 ± 3	0.0633 ± 0.0003		335 ± 12	0.067 ± 0.004	
Outer Rod	319 ± 3	0.0588 ± 0.0004		313 ± 12	0.058 ± 0.004	
Moderator	307 ± 7	0.046 ± 0.001		340 ± 12	0.052 ± 0.004	

^a Temperature of moderator was 297 °K

^b Temperature of moderator was 300 °K

The spectral indices in the cluster, as deduced from the two experiments, agree within their errors. This is not true of the values of T and r in the moderator. The value of T deduced for the moderator in the PRTR is larger than the values of T in the cluster. This casts some doubt on the validity of the measurement of T in the moderator, for the spectrum is known to be harder in the cluster than in the moderator. The PRCF results appear more reasonable.

The spatial distribution of thermal flux and lutetium activations were calculated with the one-dimensional transport theory code THERMOS. The calculations over-estimate the flux peaking in the moderator. It is suggested that this may be due to an inadequate scattering model for D_2O . This conclusion is supported by the recent work of Worden, et al⁽¹⁾ which points out the disagreement in D_2O -moderated systems between calculated and measured values of the disadvantage factor, thermal utilization, and thermal reproduction factor.

A Quarterly Report contribution has been written describing the experiment in the PRCF and the THERMOS calculations.

Batch Core Experiment in the PRTR

Calculations have been made of the effect of using a Hastelloy-C process tube in the Plutonium Recycle Test Reactor (PRTR). This process tube would be in position 1550, the channel occupied by the Fuel Element Rupture Test Facility (FERTF). It is concluded from the results of the calculations that the use of a Hastelloy-C process tube was feasible from a reactor physics standpoint. The use of the Hastelloy allows the desired power densities to be obtained within the FERTF, and contributes to flux flattening over the entire batch core. Other considerations, however, may negate the use of Hastelloy-C in this experiment.

Other calculations have continued, to determine power density in the FERTF as a function of fuel enrichment and basket material. This set of calculations is now essentially complete.

The PRTR critical test procedures, as written by the Engineering Services Department, have been reviewed. Certain modifications in these procedures have been recommended. Methods of analyzing these experiments are being developed.

An experiment was performed in the Thermal Test Reactor (TTR) to determine the feasibility of using this facility for the measurement of boron content in D_2O samples taken from the PRTR moderator. Nominal 1 liter samples were used, containing pure D_2O , 0.75 ppm B^{10} , and 4.89 ppm B^{10} . These samples were placed at the center of the TTR to determine their reactivity worth. The worth of B^{10} was measured to be 1.34 ϵ /mg. The suitability of this method for measuring B^{10} concentration in the PRTR moderator was demonstrated in the experiments.

(1) J. R. Worden, W. L. Purcell, and R. C. Liikala. "Sensitivity of Thermal Reactor Parameters to Scattering Model," Proceedings of Conference on Neutron Cross Section Technology, March, 1966.

H₂O Moderated Systems

CAF Experiments with Plutonium Fuel in H₂O Moderator

Experiments to measure critical masses, bucklings, and reflector savings of 2 wt% PuO₂-UO₂ fuel rods, 1/2 in. in diameter and 36 in. long, moderated with unpoisoned H₂O have been completed in the Critical Approach Facility. A paper describing the measurements and analytical correlations of the results has been accepted by the American Nuclear Society for presentation at the June meeting.

Experiments with Control Rod Materials

A paper describing measurements and calculations of the worth of neutron absorbing rods in plutonium-fueled light water lattices has been accepted by the American Nuclear Society for presentation at the June meeting.

Mixed Oxide Reactor Survey Calculations

Analysis of calculations of burnup of a mixed oxide fuel in various light water environments has continued. The worth of the fission products excluding xenon and samarium was calculated to be between -35 and -37 mk for all systems at 10,000 MWd/ton exposure. If -30 mk is added for xenon and samarium, the total worth of fission products is about -65 mk or 6-1/2% in reactivity at 10,000 MWd/ton exposure.

Conversion ratios, defined as the number of fissile atoms produced per fissile atom lost, were calculated for each system. The conversion ratio for the lowest moderator-to-fuel ratio is about twice as much as for the highest moderator to fuel ratio.

The parameter, $\alpha^{239} = \sigma_c^{239} / \sigma_f^{239}$, strongly affects the concentrations of Pu²⁴⁰ and Pu²⁴¹. The value of α^{239} decreases by 10% from the lowest to the highest moderator-to-fuel ratio. The decrease is due to the decreased absorption rate at nonthermal energies.

Industrial Application

Plutonium and Uranium Enriched Experiments

Program schedules and shipping requirements and costs have been estimated to allow experiments to be conducted at the Westinghouse Reactor Evaluation Center (WREC) using 2 wt% PuO₂-UO₂ fuel rods. These fuel rods are being used at Pacific Northwest Laboratory in an extensive set of subcritical and critical experiments. At the WREC the fuel will be used in zoned loading studies using UO₂ rods enriched to 2.7% U²³⁵.

University Participation

Beta and Gamma Ray Spectroscopy

The requirements of the Doctor of Philosophy degree from Washington State University have been completed by J. Riehl.* His thesis work included results of the investigation of the decay schemes of Lu^{176m} and Zr-Nb⁹⁰ which were obtained at Pacific Northwest Laboratory.

* AEC Research Fellow

Neutron Physics

Slow-Neutron Scattering

The performance of the modified Rotating-Crystal Spectrometer (RXS) was compared to the old configuration. The chopper transmission has been increased from ~ 50 to $\sim 70\%$ despite intensity losses due to operation at speeds up to 50% faster. Signal-to-noise ratio was improved in the monochromated incident beam by a factor of about 2 at 0.15 eV. This is due to the better attenuation of the new borated fiberglass chopper shims and the shorter open time. Room background count rates increased less than a factor of 2 for the new high-pressure He^3 detector-banks, although these have a total sensitivity eight times as large as the old BF_3 banks for room background neutrons. The new aluminum crystal produces time bursts of neutrons near the TOF counters which are 10 to 20% narrower than the old copper crystal. The 50% increase in flight path combined with the slightly narrower initial pulses gives approximately two times better time resolution at the TOF counters at high energies. The results of this modification can be summed up by noting that the initial pulsed beam has been increased in intensity by 50% for fixed ΔE , and the efficiency to detect scattered neutrons has been increased by 3.5 times while the time resolution, angular resolution, and signal-to-noise ratios have been substantially improved.

Data for double differential cross sections of H_2O were obtained with the RXS using a target with 93% transmission and an initial neutron energy of 304 meV. The experiment verified the existence of a small vibrational peak corresponding to the 200 meV scissor vibration of the H_2O molecule. With the higher count rates and better resolution now available, as mentioned above, this peak was clearly resolved. Additional measurements were made with water samples having transmissions of 0.7, 0.8, and 0.95. The thicker samples produced enough multiple scattering to wash out the structure of the vibration line. No difference in spectral shape was found for the experiments with sample transmissions of 0.93 and 0.95. An average resolution unfolded width of 24.5 meV was obtained from the data. Experiments are being planned to provide additional insight into the mechanisms which produce the observed line width. The theoretical interpretation will include the possibility of anharmonic terms in the hydrogen potential as well as multiphonon coupling of the vibration phonon with the known low frequency modes of liquid H_2O . A short paper reporting this first measurement of a resolved high energy vibration in liquid water has been prepared for submission to Physics Letters.

A chronological review of all previous work with the triple-axis spectrometer has been made to place in perspective the review of the quality and problems of past water scattering data. Specific review of the data is not complete. The measurement of analyzer sensitivity under low resolution conditions at 0.114 eV was completed. Under the variety of monochromator conditions used to check for systematic errors the sensitivity was quite reproducible. Five separate measurements were all within a range of 5% of the mean value. The spectrometer was not operated 100% of the time available in order to provide time for review of previous data.

Debugging has been completed on the 1107 program SCATXS which is the main program supervising LEAP, ADDELT, and SCS in separate or overlay operation. Because of budget stringency, extensive checkout has not been done.

Fast-Neutron Cross Sections

Analysis of the data and results from the December total cross-section measurements were continued. The raw open, fitted open, and raw sample data for fluorine, silicon, and indium were plotted. Analysis of the raw sample plots indicated that most of the large expectation values of $(R-\bar{R})^T V^{-1}(R-\bar{R})$, the statistical test on the validity of σ_t per channel, were due to previously undiscovered memory drops in the 400 channel analyzer. The data are undergoing corrections. The fitted open plots show structure that occurs at the right channel numbers and of such magnitude as to indicate correlation with the small structure in the total cross section. This analysis is continuing.

The σ_t results for H^1 were compared to the theoretical and other experimental results with excellent agreement. The σ_t results for H^2 were compared to the curve in BNL-325. Our σ_t results were consistently 4 to 5% lower than the curve. However, our results agreed very well with recent Oak Ridge and Livermore results. The results for B indicated a reversal in the magnitude of the two resonances at 2.45 and 2.58 MeV when compared to the work of Bockelman, Miller, Adair, and Barschall (1951). Analysis of the raw data indicates no reason for the amplitude reversal. However, our statistical deviations are of the order of 4% in this energy range.

The σ_t results for N were sent on request to J. D. Anderson at Livermore. He has measured $\sigma(\theta)_{el}$ for 10 energies between 6.8 and 14.0 MeV and wants to compare the integrated results with our values of σ_t .

The high-speed self-checking Tally paper-tape perforator has arrived, and the conceptual design of its interface to the PDP-8 has been discussed with the design engineer for the interface. Purchase requisitions have been submitted for all of the new electronic equipment planned for the coming fiscal year, and orders have actually been placed for more than half of it.

The revision of the family of data-handling codes to include a complete treatment of covariance is nearly completed. Approximately 20 cases have been run with BIG NED, although not all of these are on the intermediate-library tape yet. This version of BIG NED uses the "search read" feature available in NTRAN on the 1107, so that the data for one case can be located and read from the raw-data tape while the previous case is being processed. Except at a very few isolated points, the (I,I+2) covariance elements are systematically negative for all of the cases processed to date.

LI'L NED (in the simpler NED-2 version) is almost ready for production use. The matrix calculations in this program are done in a synthetic fashion which avoids calculating terms which will be zero or will not later be used, and some features of the logic still must be checked with hand calculations. The (I,I+2) and (I,I+3) covariance elements in the composite results prepared by LI'L NED are frequently positive. TAPEWORM/RAW has now

been brought up to date, and the raw-data tape is now written entirely in NTRAN records. TAPEWORM/NORM has been converted to handle the covariance data, and the old and new intermediate libraries have been merged in a single file. Both BIG NED and LI'L NED are equipped to update this library, and LI'L NED also uses it for input.

Theory Development

Effect of PuO₂ Particle Size on Reactivity

A project has begun to theoretically predict the reactivity effects of the size of PuO₂ grains randomly imbedded in a natural uranium oxide rod in a graphite lattice. While the fast and thermal neutron effects will be significant, the 1 to 2000 eV resonance range is of present concern, since the amount of resonance interference and shielding contribute directly to reactivity changes.

The Nordheim⁽¹⁾ integral technique is being extended to obtain approximate energy dependent collision densities in the PuO₂, UO₂, and graphite. At present, space dependence is ignored as is customary in these calculations. The usual assumption that resonances are narrow with respect to energy losses in moderator collisions is not applied, however.

A set of three equations describing the collision density for each material has been written to take into account the moderating effect of oxygen. These equations are being coded for the 1107 to give point by point collision densities. It has been found that 7200 points of lethargy interval about 0.001 adequately describe the 1 to 2000 eV range. Point cross sections of the isotopes U²³⁵, U²³⁸, Pu²³⁹, and Pu²⁴⁰ are being calculated using the resonance parameter option of the BNW Cross Section Library and are then written to a tape for reuse.

Preliminary work, ignoring oxygen entirely and using single level Breit-Wigner resonances shows that for the dimensions of the cell to be run experimentally in the PCTR, the total reactivity contribution from this energy range is nearly linear for the 0 to 700 μ particle sizes (experimental range with 2% grain concentration in the uranium rod), but reaches a maximum at about 2400 μ .

Code Development

NIT*

The 1107 version of NIT, a code for calculating the resonance integrals of individual resolved resonances according to the Nordheim treatment of ZUT, has been modified to read resonance parameters from an HRG data tape. This feature simplifies the input requirements in many cases of interest. NIT does the same numerical calculation as ZUT, but eliminates the extensive data sorting and library tape maintenance of ZUT. The practical result is that NIT is usually several times faster than ZUT.

Another modification has been made to NIT which attempts to salvage runs which would otherwise be lost because too many mesh points were requested for the resonance integral calculation for any resonance. This number of points is calculated internally as a rather complex function of the

(1) "The Theory of Resonance Absorption," L. W. Nordheim, Proc. Symp. Appl. Math., XI, 1959, p. 58.

resonance parameters and two input parameters, called r and s by Nordheim. These numbers are respectively related to the number of points per energy interval and the number of energy intervals in each integration. Heretofore, resonances with too many mesh points requested have been rejected, with an appropriate output message, and in the 1107 version of NIT this has the ultimate effect of terminating a run. Under the recent modification, when such a situation arises, either r or s is appropriately modified to bring the requested number of mesh points within the required limit so that the calculation can proceed.

BNW Master Library*

The naturally occurring isotopes of iron, nickel and chromium were updated using ORNL capture and total neutron cross section data. The old library records did not have capture widths for the resonances and were updated to include the new information. Because of the new neutron capture cross section information the new library records represent quite an improvement over the older versions. The capture widths were not previously available and have only recently been reported.

The neutron cross section data for Np^{237} was updated using the resonance parameters published in BNL-325, Supplement 2, Volume 3, February 1965. The new library record uses a resonance parameter table for increased calculation accuracy. The neutron fission cross sections and the values of $\bar{\nu}$ were the only fast cross section data that were updated. The fission cross sections are the only data available above 10 keV. Since consistent measurements of $\bar{\nu}$ are not available for Np^{237} , a theoretical value was calculated using the equation proposed by Gordeeva and Smirenkin (AERE-trans.-956, 1965) for all heavy elements of Z less than 96. The resulting thermal value for $\bar{\nu}$ of 2.67 was used with an assumed energy dependence of $0.12E$, where E is the energy of the neutron in MeV. A comparison of the calculated values with the measured values of $\bar{\nu}$ is as follows.

Energy, MeV	$\bar{\nu}$, Exp.	$\bar{\nu}$, Calc.	Differences in Percent	Source	
Thermal	2.91 ± 0.05	2.67	-8.25	AE 10 371	USSR
1.4	2.81	2.84	1.07	2nd Geneva Conf.	LASL
1.67	2.90	2.87	1.03	2nd Geneva Conf.	LASL
2.5	2.72 ± 0.15	3.00	10.29	JNE 9 153	USSR

Additional records of U^{233} , U^{235} , Pu^{239} , and Pu^{241} were added to the library to make complete sets using the respective 2200 meter/second values for the neutron cross sections recommended by Leonard, Westcott, and Sher, respectively. Both of Sher's sets were included to permit comparisons with computations that have used the older values.

ENDF Library*

The ENDF Library (BNL-8381) was imported by Theoretical Physics personnel and made available for general use. The data file service routines (DFSR's) were written for the IBM-7090 computer; however, DFSR4, which

*Partially funded by 02.

is used to print, punch, and copy the library, has been converted for use on the Univac-1107 computer along with a copy of the master library. The ENDF library currently contains the complete Aldermaston/ Winfrith neutron cross section library and a copy of the KAPL oxygen 16 data (KAPL-M-6452).

The copy option of DFSR4 is not recommended for use because this option requires almost 30 min of 1107 time to completely copy the current library.

A conversion program has been written to convert an IBM-7090 library binary tape to the proper BCD character set for the 1107. The conversion requires approximately 3.5 min of 1107 time.

Because of the form of the ENDF Library, it is anticipated that a computer program can be written that will accept the ENDF Library cards and modify the data into a form acceptable to the BNW Master Library.

REACTOR ENGINEERING DEVELOPMENT

Heat Transfer Calculations Pertaining to the FERTF Incident

Calculations were performed to determine the time required to develop the hot spots detected by metallographic examination on the fuel element which failed in the FERTF. These calculations showed that a temperature rise of 334°F/sec (183°C/sec) would occur following the occurrence of boiling burnout on the fuel element heat transfer surfaces. As a result of metallographic examination of the failed fuel, maximum temperature increases of about 800°F (425°C) were estimated. Therefore, an interval of about 2-1/2 sec would be required to cause the estimated temperature rises.

The mathematical model used to obtain these results was somewhat approximate in two respects. It was assumed that the heat transfer from the fuel to the coolant was suddenly stopped by boiling burnout. Actually, boiling burnout does not completely stop heat transfer, but rather it decreases it significantly. If the small but finite heat transfer after boiling burnout were accounted for, somewhat longer times would be calculated. Second, the model does not account for the spatial temperature variations. If these were accounted for, somewhat shorter times would be calculated. Therefore, these two errors would tend to offset each other, but it is believed that the second error is the dominant one in the early stages of the temperature rise. The simple model is, therefore, believed to be reasonably accurate, and the temperatures indicated by the metallographic structures of the hot spots would be reached 2 to 3 sec after disruption of the heat transfer.

Reactor Components Development

Chemical Shim

The dump valve test continued during the month. The valve has now operated a total of 120 days in an aqueous solution containing 160 ppm boron (boric acid). No detrimental corrosion is apparent.

On April 14, the shim rod was removed from the environmental test facility, and the shim well inspected with a boroscope. Minor corrosion was

evenly spread over the inspected areas. Accumulated test time at this inspection was 50 days. Nominal boron concentration was 50 ppm (boric acid). Coupons were removed from the storage tank (124 °F) at the same time and inspected. No unpredicted results were noted. The shim rod and balance of coupons were reinstalled and testing resumed.

CRITICAL FACILITY OPERATION

The modified PRTR Oscillator is to be used in the PRCF. The additional hardware changes needed to accomplish this were determined and a cost estimate provided.

TEST REACTOR OPERATION

Technical Guidance—Project CBB-001—Increased Power Level

Construction of the boron shim system (Phase II) is estimated at 80% complete. The ATP on this system is 95% complete. Completion of testing awaits receipt and installation of filters for the ion exchanger manifold and boric acid injection system. Difficulty has been encountered in obtaining consistent metering performance from the boron injection pump; the pump vendor is being contacted for consultation. The boron analyzer has been received.

Scheduled Phase I items are complete with the exception of the power calculator modification. ATP's were successfully completed on MCC No. 5 and MCC No. 12 EE.

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

During the month the storage basin and loadout area and soiled clothing area were accepted for occupancy with minor exceptions. Testing and balancing of the ventilation systems is presently underway.

The building portion of the work is 100% complete. Minor exceptions and some requested modifications remain to be done, however.

AMF is proceeding on design of the underwater equipment.

Technical Guidance—Project BCP-007—PRTR Waste Handling

This project is to provide improved waste handling capability and control of reactor systems during accident situations. The AEC directive for this work was issued during the month. Battelle-Northwest is performing management services and engineering for the project.

MATERIALS DEVELOPMENT

Ceramic Fuel Dissolution Studies

Ceramic fuel dissolution work during the past month has been directed toward determining structural material corrosion in concentrated acid solvents.

The corrosion characteristics of A 245 carbon steel in 90 vol% concentrated H_2SO_4 —5 vol% concentrated HNO_3 —5 vol% concentrated H_3PO_4

were studied in the laboratory. Average corrosion losses under quiescent conditions at 100 °C in 24 and 72 hr were 0.12 and 0.28 mil. Losses in like periods at 130 °C were 0.25 and 0.64 mil.

The behavior of pump bearing and valve packing materials in concentrated H_3PO_4 and 90 vol% concentrated H_2SO_4 -5 vol% concentrated HNO_3 -5 vol% concentrated H_3PO_4 was examined to determine if effects on these materials might limit the usefulness of these acidic solvents. No weight losses or dimensional changes were found on Graphitar 14 after a 24 hr treatment in concentrated H_3PO_4 at 155 °C. Asbestos and Teflon impregnated asbestos packing materials were partially degraded by H_3PO_4 in 24 hr. However, these materials after treatment could have been used effectively as valve packing. The 90 vol% concentrated H_2SO_4 -5 vol% concentrated HNO_3 -5 vol% concentrated H_3PO_4 solution at 140 °C destroyed the Graphitar 14 pump bearing within 4 min. The attack on valve packing was much more severe than that encountered in H_3PO_4 . The Teflon impregnated asbestos, although partially degraded, could have been used as valve packing after 4 hr of treatment. Increased exposure to the acid solution led to increased degradation. The behavior of the asbestos packing was inferior to the Teflon impregnated material.

Corrosion and Water Quality Studies

H_3BO_3 Analysis

Present plans for monitoring the boron concentration present in the PRTR moderator during operation with chemical shim control include the use of a continuous wet chemical analyzer employing neutralized carminic acid as the indicator. A detailed analysis procedure based on this process has been developed and was described in BNWL-SA-167.

Recently, additional studies have been conducted to determine whether the sensitivity and reproducibility of the procedure can be improved by optimizing the carminic acid neutralization step. Initial results indicate that significant improvement can be obtained by increasing the carminic acid and decreasing the alkaline buffer concentrations. This work is continuing.

FERTF Shroud Tube Decontamination

The interior surface of the aluminum shroud tube surrounding the FERTF pressure tube was contaminated locally by fuel material and fission-product debris in September, 1965. Approximately 2 g of fertile material impregnated the surface of the tube to such an extent that it could not be removed by flushing operations. Mechanical scrubbing and brushing operations reduced the amount of material to about 1 g. Further efforts of this type were unsuccessful.

It was decided to try to chemically remove this fuel contamination using the mixture of hydrogen peroxide with oxalic and gluconic acid (plus other additives) that was so successful in removing loose fuel debris from the out-of-reactor portions of the dry gas system. A two-step cleaning operation was conducted in the shroud tube-pressure tube annulus after carefully isolating this portion of the system. The amount of residual fuel decreased to about 0.75 g after the first chemical cleaning step and to about 0.45 g after

the second step. Cleaning operations were then terminated. No further cleaning operations are proposed at this time.

Primary Coolant System Prefilming

The PRTR primary coolant system will likely operate at a pH near neutral during the high power density core experiment. This system was recently chemically decontaminated and at present does not contain a protective corrosion product film on the stainless steel piping surface. Provisions have been made to prefilm these surfaces by operation at 350 to 400 °F (177 to 205 °C) with pH 10 lithiated coolant before nuclear service is resumed. The objective of this operation is to minimize the amount of corrosion product material that will be present in the recirculating coolant when nuclear operation resumes, and therefore, to minimize the rate at which irradiated corrosion products will accumulate in the out-of-reactor piping system.

The prefilming operation is scheduled to begin almost immediately and to last for 2 to 3 weeks depending on the coolant temperature obtainable during this period. Coupon samples have been inserted in the primary system in-line corrosion loop to permit evaluation of the effectiveness of the prefilming operation. Coolant samples will be analyzed frequently to determine the particulate solids concentration present during this period. An intensive program will then be followed to evaluate radiation buildup rates in the out-of-reactor piping system when nuclear service resumes to determine the effects of prefilming.

Corrosion Evaluation in Citrate-Oxalic Acid Mixtures

The effect of velocity on the corrosion of carbon steel in citrate-oxalic acid mixtures were studied in the laboratory. The mixtures were similar in composition to those used to successfully decontaminate the PRTR Primary System. The velocity tests were performed with spinning disks; coupons were mounted at selected distances from the center of the disk. The revolutions of the disk were measured with a tachometer. Testing was conducted at room temperature. The disk velocity ranged from 1 to 10 ft/sec. Actually the fluid tends to move with the disk, therefore, the calculated velocities are not absolute. The corrosion rate increased linearly with the calculated velocity, doubling with an increase of 6 ft/sec.

Chemical Shim Corrosion Studies

A PRTR dump valve is being tested to assure operation of the 17-4 PH stainless steel actuator spring during use of chemical shim in the PRTR moderator system. Failure of the spring during reactor operation could affect reactor safety. The valve has continued to operate normally during 4 months testing in the helium atmosphere over a 160 ppm boron solution. The valve springs are periodically wetted with the boric acid solution and actuated. A visual examination during this reporting period has revealed no localized corrosion areas on the actuator spring.

Pilot plant corrosion testing in the EDU Shim Test Loop located in the 314 Building has continued. These tests will determine if the localized aluminum corrosion observed during stagnant conditions in the shim well

will also occur with refreshment flow to the shim well. Loop alterations made at the conclusion of the previous test have made it possible to deoxygenate and then deionize both make-up water and a loop clean-up stream. A PRTR mechanical shim rod was placed inside and forced against the wall of a new 6061 aluminum shim well. During this test the shim well had a 2 gal/hr refreshment flow to the well bottom and a 120 gal/hr flow entering at a point 3 ft from the well bottom. All flow exited near the top of the shim rod. The loop coolant quality was maintained at approximately 50 °C, 6.0 pH, 1.0 μ mhos/cm conductivity and 0.1 to 0.8 ppm O₂. A coupon charge was inserted in the loop storage tank. Inconel 600, 304 SS, aluminized Inconel 600, and aluminum were galvanically coupled to aluminum and were designed to have crevices of varying width. Individual coupons of 17-4 PH SS, carbon steel and 6061 aluminum were also present. Examination of these coupons after 1 months testing revealed only minor aluminum corrosion in the coupon crevice areas. Carbon steel coupons placed in the vapor space of the loop storage tank contained pitting as deep as 2.5 mils while similar coupons immersed in the loop coolant were pitted as deep as 0.5 mils. The 17-4 PH SS coupons located in vapor and liquid spaces of the storage tank indicated no signs of localized corrosion. A boroscope examination of the shim well revealed only minor localized corrosion areas which will be followed as the test progresses.

PRTR Corrosion Monitoring

Corrosion probes located in the PRTR are being used to monitor material corrosion on a continuous basis. Aluminum probes in the moderator and reflector as well as carbon steel probes in the primary shield indicate negligible corrosion rates during the past month.

PRTR Primary Shield Leak Corrosion Tests

The leak in the PRTR bottom shield poses potential corrosion problems in the carbon steel primary shield system and the aluminum calandria. Corrosion rates in the reactor carbon steel primary shield system are presently being controlled to less than 0.01 mils/month by adjusting the coolant pH between 9.5 and 10.0 with NH₄OH. Testing to date has indicated that there would be no corrosion problems to the PRTR process tube or copper O-ring gaskets located in the process tube bellows assembly. Testing presently being conducted will be used to evaluate aluminum corrosion resulting from the shield coolant contacting the moderator system shim wells and piping.

Cleaning Process Water Piping

The removal of low temperature hard water scale and corrosion products from carbon steel pipe in 10% Diammonium citrate with 0.1% of a proprietary inhibitor was examined. Sections of a typical scaled pipe were completely cleaned in 8 hr at 85 °C and 16 hr at 60 °C. Corrosion loss as measured by a blank was less than 1/4 mil. The deposits (100 mg/cm²) appeared to dissolve rather than undercut and slough off.

CYCLE ANALYSIS

Fuel Rejuvenation

Calculations for the cost of fuel rejuvenation using U^{235} to "spike" the fuel were completed and evaluation is in progress.

Kinetics Studies

Work is continuing upon the addition of a package for generation of short term kinetic information to the Fuel Cycle Analysis chain. In particular, the problem areas in the current neutron lifetime calculation have been found and corrections are being developed.

TEST REACTOR OPERATION

Operating Experience

Work in April was directed toward reactor startup with the batch core loading. Progress included the following:

1. The primary seals on the three primary pumps were replaced with units made of tungsten-carbide. The tungsten-carbide seal face replaces a Stellite surface that was believed to have released cobalt to the primary system when worn. All primary pumps have been operated through a 3 to 5 day period, with satisfactory primary seal performance.
2. The primary system was pressurized to 1,500 psi with hydrostatic pressure being supplied with the makeup system injection pumps. The hydrostatic test revealed a few leaks which were subsequently repaired.
3. Flow restriction orifices were removed from 22 process tubes in April. There is now a central core of 55 process tubes with unrestricted flow. Flow monitor changes required to monitor the higher flows have also been made.
4. The core blanket system was reconnected to the drying towers and the system returned to operation, with the core discharge flow being routed through a temporary filter. The moisture detection system sample lines were reconnected between the lower access space and the sample flow rotameter rack in -21' C-cell. Dry gas sample flow was re-established through the individual sample lines.
5. The impact area of the 1550 shroud tube was pressure tested to confirm no penetration of the shroud tube prior to chemical cleaning. A hydrostatic test at 5 psig, followed by a pneumatic test at 5 psig, showed the shroud tube was intact.
6. The light water was drained from the moderator system, the system air-dried, flushed with degraded heavy water, and filled with heavy water from the reflector system. A bypass filter was installed across the moderator pumps in April. The filter is intended to remove particulate fuel material from the moderator coolant.
7. The remaining process tubes were inspected in April. All process tubes now in the reactor have been inspected during the current outage.

8. The process tube temperature sensing element modification program was completed in April.
9. Testing of the boron addition and removal system in the mockup pit has continued throughout the month. Remaining installation work includes moving the equipment from the mockup pit to the process area.

Maintenance Experience

Significant maintenance items for the month included primary pump repairs and boron shim control system mockup. A total of 3,630 craft man-hours were utilized during the April outage as follows:

FERTF Fuel Failure	776
Decontamination	197
Repair	920
Modification and Improvement (mainly capital work)	1079
Operating Assistance	240
Preventive Maintenance	418
	<u>3630</u> manhours

Improvement Work Status

Work Completed

Moisture Detection System
 Calandria Measurements
 Moderator Filter
 Primary System Filters
 Pressure Testing of 1550 Shroud Tube
 Primary Pump HX Piping
 Unit Motion Readings

Work Partially Completed

Hydraulic Thermal Barrier Lift
 New Instrument Power Supply
 Traveling Flux Wire Probes
 Reactor Hall Crane Work Platform
 Effluent Monitor System Calibration Modifications
 Diesel Well Pump—Remote Start
 Instrumented Fuel Element Installation
 Core Blanket Filters
 PRTR Critical Test Instrumentation
 Relocate IX-4
 New Sulphite Tank

Design Work Completed

Reflector Change to Light Water
 Core Blanket Purge Line Removal
 New Sulphite Tank

Design Work Partially Completed

Fuel Transfer System Improvements
Analysis of HX-1 and Primary Pump Supports
Rupture Loop Decontamination Equipment

Process Engineering and Reactor Physics

The final report of PRTR Test 115, "Use of Soluble Poison for Reactivity Compensation in PRTR," has been issued. A document, BNWL-CC-608, "Planned Mode of Operation—PRTR Chemical Shim," was prepared and issued. This document describes the way in which the chemical shim system is intended to be operated.

The PRTR primary system is being prefilled with pump heating prior to nuclear operation. Crud removal and measurement is being performed during this filming. Sintered stainless steel filters, 5 μ pore size, were put in two process channels for continuous removal of crud. Daily water samples filtered through 0.45 μ filters show continuing reduction in amounts of crud being transported in the system. Hot, in-situ, crud samples are also being taken to provide quantitative measurements and analysis of crud materials. New filters were installed in selected lines of the Fuel Element Rupture Detection System to follow crud buildup by increase in radioactivity; however, the activity increase is not measurable because there is so little radioactive material in the primary system.

Alpha radioautographs were made of the impacted area of shroud tube 1550. They indicate 0.4 g of fuel material remained after chemical cleaning.

A series of THERMOS cases were run for the Fuel Element Design and Evaluation Unit in order to compare flux depression in a single Zirconium-clad fuel rod surrounded by H₂O coolant, a zirconium process tube, and D₂O moderator borated to 81 ppm at various PuO₂ enrichments and theoretical densities.

EBWR DEMONSTRATION PROGRAM

Irradiation Testing of EBWR Prototype Fuel Rods

Three capsules (GEH-14-523, -524, and -529 in the table below) continue their irradiation in the MTR. The capsules were assembled with short sections of Zircaloy clad EBWR rods containing vibrationally-compacted, pneumatically-impacted, UO₂-1.5 wt% PuO₂. The maximum burnup achieved with the three capsules, the last of a group of 32, is 5.8×10^{20} fissions/cm³ (23,200 MWd/MT of fuel). Goal burnups to 27,500 MWd/MT of fuel are presently scheduled.

Production-run EBWR rods, 1.07 cm OD by 148 cm long (includes 14.6 cm-long plenum) and with bulk fuel densities of 86 to 89% TD, are being irradiated. The 21-rod bundle (Fuel Element No. 6501) in the PRTR will continue its irradiation when the reactor resumes operation. Present bundle burnup is 0.7×10^{20} fissions/cm³. An additional 24 rods are being irradiated tested elsewhere in three-rod clusters. Cluster goal burnups to 27,500 MWd/MT of fuel are planned. Of 58 full-size rods being tested, 13 have been discharged and of these one has been examined (see table below).

Irradiation Testing of Vibrationally-Compacted
Impacted UO₂-1.5 wt% PuO₂ Capsules^(a)

GEH-14 No.	Bulk Fuel Density, % TD	Heat Flux ^(b) , W/cm ²	Rod Power, W/cm	Rod Power, ^(b) kw/ft	Exposure ^(b)		Fission Gas Release, %
					fiss/cm ³ x 10 ⁻²⁰	MWd/MT of UO ₂ -PuO ₂	
517	82	53	180	5.49	0.44	1,840	5
518	82	113	380	11.6	0.40	1,660	14
519	83	340	1,160	35.2	0.61	2,480	lost
520	82	126	427	13.0	1.73	7,070	38
521	83	139	472	14.4	3.07	14,160	37
522	82	127	431	13.1	3.15	12,930	39
523	84	(158)	(535)	(16.3)	(6.8)	(27,500)	
524	83	(158)	(535)	(16.3)	(6.8)	(27,500)	
525	80	155	522	15.8	1.25	4,570	54
526	82	125	425	13.0	1.1	4,820	?
527	79	129	436	13.3	2.36	10,050	32
528	84	170	577	17.6	2.26	9,050	41
529	84	(158)	(535)	(16.3)	(5.5)	(22,000)	
646(c)	80	147	501	15.3	0.24	1,010	30
647(c)	77	147	500	15.2	0.72	3,140	22
648(c)	81	190	635	19.4	3.69	15,400	30
649(c)	79	188	641	19.5	1.55	6,640	33
650(c)	80	304	1,037	31.6	0.46	1,950	81
651(c)	79	374	1,276	38.6	1.6	6,860	84
706	81	298	1,013	30.6	0.55	2,260	51
707	83	298	1,010	30.8	2.25	9,150	94
708	81	125	424	12.9	0.71	2,950	18
709	81	138	465	14.2	1.3	5,440	27
710	81	313	1,070	32.4	1.3	5,180	~100
711	81	219	745	22.7	2.54	10,500	41
712	84	(236)	(803)	(24.5)	(4.1)	(16,500)	
713	85	267	905	27.6	2.95	11,800	?
714	77	96	327	9.97	0.51	2,220	10
715	78	100	341	10.4	0.98	4,240	?
716	78	(95)	(323)	(9.8)	(1.3)	(5,500)	
717	80	132	448	13.7	0.48	2,010	14
718(c)	79	218	745	22.7	0.40	1,730	52
EI36(d)	86-89	61	208	6.35	0.52	2,010	6

(a) All capsules contain impacted fuel. Tabular values in parenthesis are proposed or design values.

(b) Values based on flux-monitoring wire data and/or burnup analyses. Rod powers are initial values. Zircaloy cladding, 1.08 cm OD by 0.064 cm wall.

(c) Plenum-equipped capsules.

(d) Full-size, production-run EBWR rod irradiated in the PRTR (Fuel Element No. 6501).

ADVANCED SYSTEMS

U²³³ FUELING OF A FAST COMPACT REACTOR - J. C. Fox

The thermal and hydraulic code is debugged and sufficient data obtained to complete the study for an internally liquid metal-cooled core and pin core. Graphs for interpreting data are complete for the internally-cooled model with graphs for the pin model well underway. A rough draft for the appendix part of the report is nearing completion.

To refine the sizing study performed during the previous year, the DTF-IV transport theory code was used to do a size survey for a wide range of void fractions and fuel compositions. Reactivity lifetime calculations for selected reactors were done using FCC, a fundamental mode code. Various kinetics parameters of the three fuels are currently being computed.

NUCLEAR SAFETY

CONTAINMENT SYSTEMS EXPERIMENT - J. M. Batch

Experimental Facilities

Fabrication activities continued on the liquid and aerosol sampling systems. Construction was started on the dock for high pressure gases and on the 220 psig steam supply line for aerosol injection.

A revised schedule for the delivery of the reactor simulator vessel was received from the vendor. This schedule calls for a delay of shipping date to September 9 because of the vendor's difficulty to supply nozzles according to the ASME Nuclear Code, Section III.

Aerosol Development

Testing of injection methods for introducing fission product aerosols into containment vessels was continued. The use of a steam jet as a compressor gave indication of liquid droplet formation during operation of the jet. Slight superheating of the jet motive steam improved the delivery to near normal behavior, but further improvement must be shown before a jet can be used in the CSE tests.

Pyrex windows were installed on nozzles of the ADF containment vessel so that visual observations of the steam-air environment could be made. Rapid injection of steam into room temperature air caused dense, billowing fog. After sufficient steam was added to raise the temperature to steady-state conditions, the fog rapidly dispelled, leaving a very clear atmosphere. Occasional mist and fog were observed later in the run but could be cleared by changing the pressure and temperature conditions in the vessel. When the aerosol was injected, vision was obscured again by the smoke of UO₂ and SS cladding particles. A distinct Tyndall beam was observed which persisted for 60 min.

Steam-air ratios calculated from temperature and pressure measurements agreed well with measurements of air volume and condensed steam.

Decontamination factors of about 30 for Cs¹³⁷ and 6 for Ru¹⁰⁶ were obtained for 304 SS coupons contaminated by aerosols and treated in an ultrasonic bath of detergent solution. Hard manual scrubbing gave a DF of only 3 for Ru¹⁰⁶.

Iodine Release

Evaluation of methods for equilibrating 10 to 100 g of stable iodine with tracer iodine is in progress. For generating quantities of iodine on the order of 1 g or less, the dichromate fusion method is adequate. On a larger scale, difficulties ensue from an increase in fusion temperature of the dichromate iodide mass due to the presence of large quantities of reduced chromium oxides. Investigation of palladous iodide decompositions has ceased since a large number of material transfers are required to obtain a pure iodine product.

Equilibration of solid traced potassium iodide with liquid bulk iodine at 130 to 150 °C appears promising. Preliminary results show an exchange half-life less than 1 hr. It may be possible to obtain a one-step equilibration and release by this method. If highly volatile compounds are present, they will come off early in the release.

One other method of iodine generation which is simple in nature is the decomposition of equilibrated potassium or sodium iodides. So far, this has not been considered seriously due to the possibility of particle release at the high temperatures necessary to decompose the iodides. Recondensation prior to the release to containment could allay this problem.

Multichannel Analyzer Code

Debugging of the multichannel analyzer codes has been completed. In addition to the data reported for a sample analysis as was discussed last month, the goodness of fit parameter is now also reported. This parameter will be used to determine whether all isotopes which are present in the sample have been analyzed for. The Relative Standard Error gives an indication whether an isotope is really present or not in the sample.

The four analyzer and detector systems have been calibrated for a gamma energy range of 1.8 meV per 256 channels and storage of standard spectra has commenced.

Containment Vessel Leakage Tests

A new reference vessel system has been fabricated and installed in the containment vessel. All tubing external to the vessel represents 0.8% of the total volume of the new system. The new reference system has been helium leak tested, and the largest leak found read 20% of a scale that reads 100% deflection when exposed to a 2×10^{-6} std cm³/sec calibrated leak. This leak was repaired.

All of the instrumentation of the last series of tests has been reinstalled in the vessel. Six of 25 of the new RTD's have been added to the system, and four more humidity sensors installed. The nickel wire was restrung and will be used for this series of tests. The vessel was pressurized April 21 for further exploratory leakage testing.

The penetration design for the instrumentation feedthroughs has been drafted and check prints issued.

PDP-7 Digital Programming for the Containment Experiment

A standard statistical analysis routine was written as a link program and has been debugged. A complete test case for the entire system which involves the sequential calling of six link programs was successfully executed. Work began on writing and documenting all aspects of the CSE programs.

CSE Instrumentation

Specifications were prepared for pressure transducers to be used on the reactor simulator. An unusual requirement in this application is the need for temperature compensation over the range from ambient to 600 °F (316 °C). Other special features are that they should be flush diaphragm type with bonded strain gages for high shock resistance. The bid package for all dynamic thermocouples and accessories was received and evaluated.

The first four load cells for use on SM-2 and CSE were received. The influence of temperature on these 10,000 lb load cells will be determined by exposing them to a high temperature test prior to installing them on the SM-2 vessel. Certified load calibration data received with the transducers obviates field load calibrations by BNW personnel.

Dummy Hydraulic Core

Work has begun on the design of a hydraulic core and the associated instrumentation necessary to measure and record all experimental data obtained during CSE blowdown tests. Initially, several existing power reactor cores are being compared to determine design parameters for the CSE hydraulic core.

PRESSURE VESSEL CRACK MONITORING - J. C. Spanner

Detection of Metal Overstress by Acoustic Emission

Development efforts continued toward providing a system to monitor for yielding and/or flaw propagation in reactor pressure piping by sensing the accompanying acoustic emissions. Analysis of data, derived from testing a large, 304-L SS tensile specimen, has been completed. As previously reported, two distinct types of signals were observed--random bursts of relatively high amplitude, and what appears to be a continuous frequency signal whose amplitude varies directly with specimen loading.

Burst signals, chosen arbitrarily from the vicinity of yield point loading, fit very well into a triangulation pattern including sensing points and the point of metal deformation. The primary mode of signal transmission in the metal appears to be shear wave. Velocities, calculated using the known distances and the measured time values, agreed with each other and with theoretical shear wave velocities to within 1%. Earlier problems in time resolution were alleviated by using a dual beam oscilloscope and a camera instead of an oscillograph for this analysis.

Thorough examination shows that what appears to be a continuous frequency signal is actually a random signal occurring on a much shorter

time interval and at a lower amplitude than the burst signals. The average frequency increases from about 2 kHz, at specimen loading below yield, to about 5 kHz, at specimen loading of 85% of ultimate. Amplitude varies directly with specimen loading. A pulse rate counter scan of selectively filtered test data produces a pattern which follows specimen loading quite closely. It was also found that the high sensitivity transducers used in this test were picking up some airborne sound but this does not appear to be a serious problem. Metallurgical examination of the deformed section of the test specimen is in progress for the purpose of comparing metallurgical events with the observed acoustic emissions.

Type 304 SS was chosen for these initial tests because it is used in the primary piping systems of a number of reactors. These data provide a basis to proceed with examinations of crack propagation per se, different loading conditions, etc., to elucidate their unique characteristics.

Surveillance of a hydrostatic test on the secondary side of an N-Reactor heat exchanger indicated that complex geometry will not be a serious problem in signal detection, but it will provide an added challenge in the triangulation of signal sources. Bandpass filters were quite effective in controlling background noise although the data from this test have not yet been thoroughly analyzed.

Evaluation of a broadband, point source, omnidirectional transducer was started. The bandpass of this transducer and its associated amplifiers appeared to be from 900 Hz to about 10 to 15 MHz. However, the development of a readout device to cover single occurring signals throughout this frequency range will present a challenge. This type of transducer will be further evaluated as the potential basis for a multiunit, directional system.

Work has also been started on the evaluation of a mass loaded capacitor receiver for use in the detection of acoustic emission. A new approach will be used to measure capacity changes in the capacity transducer. These capacity changes will be used to frequency modulate a very sensitive oscillator circuit which may be detected and converted to analog instead of using the conventional dc biased capacity coupled technique employed in the electrostatic transducer concept.

Trial adaptation of standard materials for producing a paste transducer which would be easy to apply, polarize, and instrument in hope that it would alleviate the coupling problems and be omnidirectional for shear, longitudinal, and surface waves gave promising results. PZT-5 powder was mixed in an air curing epoxy and molded in a disc form. Silver electrodes were then painted on and the polarization voltage was applied at room temperature.

There is evidence that increasing procurement time for supplies and equipment may become a significant factor in maintaining the time schedule of this program.

RADIOACTIVE RESIDUE PROCESS DEVELOPMENTS - A. M. Platt

High Level Waste

Waste Solidification Engineering Prototypes (WSEP)

Installation of WSEP is greater than 95% complete. Preliminary functional testing has indicated the necessity to rework several electrical items before operational tests. Other mechanical adjustments and corrections are being completed prior to the first operational tests next month. The GE-412 computer installation is 90% complete and programming and debugging is 50% complete.

Two casks of waste (~300 gal each) were received from Isochem and have been stored in the Load-Out Station of the 324 Building. This constitutes the first radioactive material introduced into 324 Building. Safe-guard controls and radiation protection procedures have been initiated.

Melt Compositions

Melt compositions which permit satisfactory processing of PW-1 in the spray calciner-melter system have been developed but variation in permissible concentrations of feed and additives is quite limited. In an effort to find compositions with a broader satisfactory operating range, lead and boron were studied as additives to the $\text{Na}_2\text{O}:\text{P}_2\text{O}_5:\text{PW-1}$ system. Tribasic lead silicate ($\text{Pb}\cdot 0.33 \text{ SiO}_2$) was chosen as the additive for introducing lead; 16 melts were prepared. When compared on a molar basis, tribasic lead silicate was more effective than Na_2O in lowering the melting point but only in the glass forming range, i.e., when the M/P ratio was between 1.0 and 2.0. When the M/P ratio was greater than 2, Na_2O was more effective in lowering the melting point. It appears there is no particular advantage in adding lead to the melts.

A similar study involving B_2O_3 as additives is not yet complete. Several compositions that melt at around 900 °C and have "slump points" of about 700 °C have been found. These may permit the use of mild steel melt receivers; further study is planned.

Glass Studies

Work continued on interpretation of the data from the statistical-factorial study of the phosphate melt system. The first set of contour plots of the drip temperatures have been completed. A formal topical report will be prepared as soon as these correlations are complete.

The study of the mixed phosphate-borate-silicate system is about 60% completed.

Continued adjustment work on the Thermal Analysis System has brought the thermal gradient for dilatometer samples within allowable limits. The thermobalance feature is also now operable to 1500 °C.

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

Experiments were undertaken to determine whether there would be an advantage in ferrocyanide over ferricyanide precipitates in the removal of cesium in conjunction with mixed oxide scavenging. In the pH range 4 to 12, ferricyanide removed from 5 to 9% more cesium than a corresponding amount of the ferrocyanide.

Experiments were performed with carrier calcium and/or strontium to study removals of strontium from the glass melter condensate. Only samples having $\text{Sr}(\text{NO}_3)_2$ additions of 0.04M or greater resulted in increased strontium removal. Using ca. 0.08M $\text{Sr}(\text{NO}_3)_2$, a removal increase of about 5%, i.e., 99.7% total removal, was obtained.

In the second experiment, $\text{Ca}(\text{OH})_2$ was added to the simulated glass melter condensate with other reagents normally used in the mixed oxide-ferricyanide process. Then the pH was adjusted to 10 by NaOH. Rather large concentrations of Ca^{++} are required to increase strontium removals. In this case, addition of 0.4M Ca^{++} improved strontium removal from 99.2% with no Ca^{++} added to 98.4%. No increase was observed at 0.01M or 0.04M Ca^{++} .

Crystallization of sodium sulfate at temperatures near 100 °C from partially neutralized, simulated glass melter condensate was investigated. A 50% solution of sodium hydroxide was added to the simulated condensate in the stoichiometric amount to react with the sulfuric acid. A portion of the partially neutralized condensate was evaporated by boiling to half its original volume. The sodium sulfate crystals were removed by filtration and washed with a portion of the partially neutralized condensate. The filtrate and washes were recycled. This crystallization and wash procedure was repeated six times. Analytical results with Sr^{85} tracer indicated that only half of the strontium is separated from the sodium sulfate. Substitution of small volume water washes in place of the condensate washes did not improve strontium separation.

RADIOACTIVE SHIPMENT HAZARDS - L. C. Schwendiman

Strontium-90 released from glass prepared from actual high level waste was found to range from 5×10^{-7} to 5×10^{-6} per hour when the molten glass was heated at 1100 and 1200 °C. The samples consisted of an actual high level waste glass blended with similarly prepared nonradioactive calcined material. Analyses made so far have shown Ce^{144} to be released at a level below the detection limit, estimated to be around 10^{-7} per hour. Both SS and nickel crucibles were used. Release from other calcine compositions will also be studied.

FISSION PRODUCT AEROSOL CONTAINMENT - L. C. Schwendiman

Investigations of the behavior of the gas phase system $\text{CH}_3\text{I}-\text{N}_2\text{H}_4$ continued using spectrophotometric and gas chromatographic techniques. The data suggest that two processes are involved in the removal of methyl iodide from the gas phase. Furthermore, the residual methyl iodide concentration is apparently 20% of the initial concentration when the hydrazine

should be present in a 10 fold or greater excess of the stoichiometric amount. This conclusion is supported by a similar behavior of the ethyl iodide-hydrazine system. The presence of carbon dioxide greatly affects the removal rate of methyl iodide. The principal mechanism of this effect is the removal of hydrazine from the gas phase probably by the formation of carbazic acid or the hydrazine salt of hydrazinocarbonic acid. Addition of ammonia or some other base to the system should overcome the problems caused by the presence of carbon dioxide in "real" systems.

Shakedown runs were completed with the 114-liter aerosol chamber which is to be used to measure CH_3I removal with sprays containing hydrazine. Wall temperature can now be brought up to greater than 95°C , which may be an important factor in the methyl iodide reaction. The 114 liter chamber will be used until the larger chamber is made available. Scouting tests showed that the wall temperature did not drop rapidly when the spray system was in operation. Methyl iodide tagging and generation apparatus was assembled.

The study of the interaction of iodine with paint and paint constituents continued using a sensitive gas chromatograph as an analytical tool. Considerable difficulty was experienced in separating I_2 and HI from CH_3I . The sensitivity of the gas chromatograph for CH_3I using the electron capture detector was established at 3×10^{-12} g CH_3I in $25 \mu\text{l}$, which gave one-tenth of full scale deflection on the instrument.

It was also found that samples containing Phenoline[®] thinner vapor and I_2 which were exposed to ultra violet radiation at ambient temperature yielded ethyl iodide and either CH_3I or HI or both, while an identical sample shielded from such radiation produced no detectable amount of volatile iodides. Preliminary investigation of the reaction under ultra violet radiation at ambient temperatures of I_2 with coupons painted with Phenoline 302[®] and 300[®] primer showed that ethyl iodide is produced and probably methyl iodide also.

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.

During this period two more A212B subsize-vessel specimens were inserted in the M3 position of the ETR. This makes a total of four irradiated A212B specimens that will be ready for destructive testing in the first quarter of FY-1967. Part of the program involves pressurizing specimens containing a crack through the wall. Of the several patching techniques that have been tried, the most promising is a thin-walled metal sleeve that acts as a bladder. Tests are in progress to evaluate the effect of the bladder on the effective pressure on the vessel.

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

The fatigue growth of cracks in subsize-vessel specimens is being studied. The radial growth of the crack is being monitored by nondestructive testing techniques and the acoustic emission of the growing crack is also being monitored. The starting crack is a longitudinal slot 0.008 in. wide on the specimen's inside surface. It tapers to a sharp notch with a root radius of 0.001 in. or less. The slot is formed in the specimens by a electrojet machining process. The first of these crack growth tests at a temperature of -50 °F is in progress. The crack will be grown to a depth of approximately one-half the specimen wall and then the specimen will be pressurized to failure.

NUCLEAR FACILITY MONITORING GUIDE - I. C. Roberts

Final editing continued on the power reactor effluent and environmental monitoring guide. Most effort during the month was directed toward preparing material recommended for inclusion by representatives of the U.S. AEC regulatory division. A rewritten version of the first chapter was distributed to those representatives and to appropriate persons in the Reactor Development and Technology Division.

CUSTOMER

ASSISTANCE TO PHILLIPS PETROLEUM COMPANY - W. L. Hampson, Jr. Enriched UO₂ Fuel Pins for Contamination-Decontamination Experiment

Sixteen additional fuel element pins, loaded with 8 wt% enriched UO₂ pellets, were fabricated and shipped to the customer. This shipment, plus four pins previously shipped, constitute completion of this contract. A letter, summarizing fabrication data, was also forwarded to the customer during the month. The pins will be used in the Contamination-Decontamination experiment at the National Reactor Testing Station (NRTS), Idaho Falls, Idaho.

SAP END CLOSURES--ATOMICS INTERNATIONAL

A trip was made to Atomics International, Canoga Park, California, to aid them in setting up a process for producing large numbers of end closures in SAP fuel tubes. Assistance was given in optimizing parameters for the process.

Additional work was performed at Battelle-Northwest to provide information on the absolute tensile strength of the welds. This work consisted of making and testing SAP tube-to-tube joints with parameters determined on a statistical basis. The weld samples were machined into tensile specimens and tested at room temperature. Tensile strengths varied from 29,000 to 35,000 psi, or 80 to 96% of the parent material strength.

A final report on the program to furnish SAP experimental end closures and information to Atomics International was completed.

ASSISTANCE TO BROOKHAVEN NATIONAL LABORATORY - S. Goldsmith
Postirradiation Examination of Cermet Pellets

Two capsules containing SS-60 vol% UO_2 cermet pellets were received for postirradiation examination. These cermets were prepared by Brookhaven National Laboratory and irradiated in the ETR. The cermet pellets (34 in each capsule) have hemispherical ends and are approximately 0.160 in. in diameter and 0.180 in. long. The pellets were removed from their sample holders and photographed in the as-received condition. The diameter, length, and density of each pellet was measured.

Selected pellets (about six from each capsule) are to be metallographically examined and autoradiographed. Other pellets are to be selected for burnup and chemical analysis and crush strength measurements.

WESTINGHOUSE DNB STUDIES

Installation of Rod 226-02, Phase II of the 25-rod DNB study was completed. The section was moved and installed in the vertical heat transfer facility without incident. After all connecting piping was installed, the installation was subjected to a hydrostatic test at 2300 psig. No leaks were observed from the heater rod instrument lines. Following completion of the hydrostatic test, the remaining hook-up work began.

Temperatures at the end of the active section for each of the inner 9 high flux rods will be monitored individually while 3 thermopiles will be used to monitor the EOAS temperatures for the 16 outer rods. As before, the 50-rod thermocouples on either side of the flow mixing grid will be grouped into 10 thermopiles. Run plans and data sheets have been prepared for the experiments.

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