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Argonne National Laboratory

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

May 1967

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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

May 1967

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Issued June 30, 1967

FOREWORD

The Reactor Development Program Progress Report, issued monthly, is intended to be a means of reporting those items of significant technical progress which have occurred in both the specific reactor projects and the general engineering research and development programs. The report is organized in a way which, it is hoped, gives the clearest, most logical overall view of progress. The budget classification is followed only in broad outline, and no attempt is made to report separately on each sub-activity number. Further, since the intent is to report only items of significant progress, not all activities are reported each month. In order to issue this report as soon as possible after the end of the month editorial work must necessarily be limited. Also, since this is an informal progress report, the results and data presented should be understood to be preliminary and subject to change unless otherwise stated.

The issuance of these reports is not intended to constitute publication in any sense of the word. Final results either will be submitted for publication in regular professional journals or will be published in the form of ANL topical reports.

The last six reports issued
in this series are:

November 1966	ANL-7279
December 1966	ANL-7286
January 1967	ANL-7302
February 1967	ANL-7308
March 1967	ANL-7317
April 1967	ANL-7329

REACTOR DEVELOPMENT PROGRAM
Highlights of Project Activities for May 1967

EBWR Plutonium Recycle Program

EBWR was shut down in late April to remove special fuel pins and several standard fuel pins for subsequent isotopic analysis at Pacific Northwest Laboratory. Also during the shutdown period cadmium-covered foils were inserted for irradiation at power, as part of an experiment to study the epithermal ratio of capture to fission in Pu^{239} and Pu^{241} , and the turbine was inspected. At the end of the month EBWR will be brought up to operating power.

EBR-II

Significant core-loading changes in EBR-II were made between Runs No. 24 and 25. A reduction in the previously observed power coefficient was observed.

A detailed reactor kinetics analysis measurement program at powers from 500 kW to 45 MWt indicated that the reactivity addition required to increase power over this range had decreased from a previous overall average of 66 lh to the present 39 lh. The overall average is now 0.87 lh/MW, and everywhere the prompt coefficient is strongly negative. Having assured that the reactor can be operated safely, Run No. 25 was begun on May 19. On May 24, while operating at 45 MW, the fission gas monitor indicated an alarm signal and the reactor was shut down. An intensive program of analysis of cover-gas samples and coolant sodium was initiated to determine the extent of release of fission products.

A two-phase plan of action was formulated with the objective of returning the reactor to operation in an expeditious manner. Interlock checks, measurements of rod scram time, nuclear channels tests, and determination of core reactivity indicated that no apparent damage to the reactor existed and no significant fuel movement had taken place.

The next step of the planned investigation involved operation of the reactor at 2.5 MWt for an extended period, followed by stepwise increases in power levels with a hold at each step for about 8 hr. This procedure was to be followed until a level was reached at which a signal from the fission gas monitor would provide a repeated indication of fission gas release, the value of which would be used as a reference in the further identification of the failed element. At month's end, the reactor had operated for 13 hr at a power of 2.5 MWt with no evidence of further fission gas release. The reactor was then shut down pending DRDT concurrence in proceeding with the established plan.

Fuel Cycle Facility

Normal production activities were carried out, with 14 subassemblies being fabricated. Three alloy-preparation runs were made with unirradiated feed stocks to supplement the inventory of irradiated fuel. A materials test subassembly, designated as Surveillance I, was received from the reactor, washed with alcohol, and dismantled. Six irradiated fuel elements were sent to the Idaho Chemical Processing Plant for reprocessing and corrosion studies. Work on the cold line for fabrication of unirradiated fuel is proceeding on schedule.

Hot Fuel Examination Facility (HFEF)

This facility will have the capability of interim and terminal examination of fuels and structural materials that have been irradiated in EBR-II, TREAT, and other facilities in support of the fast reactor and other safety programs. A project organization has been established, and a project staff is being organized. The general technical requirements of the facility have been written and are being reviewed with principal users within the Laboratory.

ZPR-3

Experiments are reported on Assembly 49, a sodium-voided version of Assembly 48, and the second in a series of six physics cores planned for ZPR-3. This assembly has a composition of fuel with a ratio of U^{238} to Pu^{239} of 4.5, graphite, stainless steel, and void. A comparison of data between these two cores will result in information on the effect of sodium on various nuclear parameters. Fission ratios at the core center were measured. Reactivity measurements were also made in the same location for various fissile, fertile, structural, and poison material samples.

ZPPR

Most major concrete pours have been completed, and the contractor is placing earth fill around the reactor cell and workroom-vault area. Interior air-handling equipment and ductwork is being installed. Electrical connections are being made to installed equipment. Roof filter sand is undergoing grading preliminary to removal of fines and a double-water wash. The status of reactor components is reported.

AARR

Title II design is about 30% complete.

Specifications for Construction Package No. 1 (rough grading and excavation) have been sent to prospective bidders, as have the specifications for the semipermanent and removable beryllium.

Preliminary specifications for the reactor-building foundation mat and for the containment liner are being reviewed.

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I. PLUTONIUM UTILIZATION--EBWR

A. Research and Development

1. EBWR Plutonium Recycle Program

The reactor shutdown in late April following the second irradiation step was continued during the first part of this month. During this downtime, a second set of irradiated fuel pins was removed from the plutonium region of the core. This second set of pins will be transported to Battelle Northwest Laboratory following a delay of several months. The first set of fuel pins (removed February 1967) left the Laboratory for Battelle Northwest on May 18, utilizing the "Yankee" cask. These pins arrived at Battelle and were removed from the cask without unusual problems.

During the shutdown, the top casing of the turbine was removed to determine the source of vibration increase noted on the evening of April 27. The cause was found to be due to the loss of three blades and the shroud piece from the eighth Rateau staging. Also some of the remaining blades on this stage were found to be cracked at the base of the shroud rivet, and therefore the entire row of blades was removed. The blading was not replaced. Because the condition of the remaining rows of blades was not inspected, it was decided that the turbine would not be used during the remainder of the irradiation program. The casing and stationary blade diaphragms showed heavy erosion.

The plant was started up May 13 and reached the 40-MW level the following day. Boric acid was removed, and the power was increased to 60 MW. At this level, flux and pressure variations were seen. After some experimentation in varying the modes of control and lowering the power level to 40 MW, a cyclic flux variation of about 25% and a frequency of about 30 cycles per minute developed. The reactor was shut down within two minutes after these variations began.

The bonnet and plug of the pressure control valve for the reboiler facility was removed in order to replace the valve stem packing. It was at this time that a stainless steel blade, $2\frac{1}{2} \times 8\frac{1}{2} \times 1/8$ in., was found lodged in the valve seat. This blade had been used as a guide piece during the assembly of the steam ducting within the reactor vessel and had been spot welded to the inside of the steam duct. Although two other similar plates were found to be securely in place, the weld was reinforced by bolting the blades to the duct. Resumption of power operations began May 29.

Power generated by the plant is as follows:

	<u>Thermal Power (MWh)</u>	<u>Electrical Power (MWh)</u>
During May 1967	6,660.6	-
Total since start of Plutonium Recycle Program	172,193.6	16,671.0

II. LIQUID-METAL FAST BREEDER REACTORS

A. EBR-II

1. Reactor Operations

a. Run 25. Planned experiments were continued in the program to perform reactor kinetics measurements prior to the start of Run No. 25. These measurements were deemed important in view of the significant changes in core loading of the reactor since the last power operation. As reported last month (ANL-7329), measurements made during this cautious approach-to-power program revealed that the incremental power coefficient throughout the range of power up to 45 MWt had decreased from the values experienced in previous runs. The presence of the usual prompt negative power coefficient was verified by rod-drop data taken in increments up to 45 MWt.

Further analysis of the overall data indicates that the reactivity addition required to increase power from 500 kW to 45 MW had decreased from a value of 66 lh for Run No. 24 to 39 lh for Run No. 25. The power coefficient decreases from a value of 1.62 lh/MW at 10 MW to less than 1.0 lh/MW at 25 MW. The overall average is now 0.87 lh/MW.

The isothermal temperature coefficient was remeasured for the new reactor loading configuration. The bulk-sodium temperature was reduced to 651°F to measure the isothermal temperature coefficient. The reactivity gain was observed to be 51 lh, which corresponds to a temperature coefficient of 1.04 lh/°F. This compares favorably with the coefficient determined for a 77-subassembly core of 1.0 lh/°F.

To separate the thermal contributions to the reactor power coefficient, a series of runs was made in which the temperature change at the reactor coolant, ΔT , was increased by decreasing primary flow at some specific power level below 45 MWt. The reactivity change associated with reactor flow change at a constant power was thus determined within the power-to-flow ratio permitted (0.72). At 22.5 MWt the initial flow reduction between 100 and 75% produced a slightly positive change in reactivity. Below 75% flow, the reactivity change was negative. At 40 MWt the reactivity change associated with flow variations appear to be negative in the same flow region.

b. Fission-gas Release on May 24. With the conclusion that operation of the reactor with the lower power coefficient does not constitute a hazard, approval to operate routinely at 45 MWt was received; operation at this power level began on May 19. On May 24, a fission-gas alarm signal from the charged-wire fission-gas monitor was received in the control room at about 1210 hr. No increase was noted on any of the three FERD (delayed-neutron monitor) channels, subassembly outlet temperatures,

reactor outlet temperature, nuclear instrumentation, or reactor power level. A gas sample from the primary tank was immediately taken and reactor shutdown was initiated at 1218 hr, when above normal gas activity was confirmed in the sample. At 1226 hr, air activity above normal was observed on air-activity monitors in the reactor building and the building was immediately evacuated. Gas samples from the primary tank were taken throughout the afternoon and night. Major activity in early samples was Xe^{138} . By 0800 hr, May 25, short-lived activity had decayed, leaving only Xe^{133} and Xe^{135} in the cover gas.

On the morning of May 25 the reactor-building air activity was well below RCG-40. After health physics surveys revealed contamination below acceptable limits, normal access to the building was established.

A two-phase plan of action was formulated with the objective of returning the reactor to operation in the most expeditious manner possible. The overall plan encompasses tests, fuel and sample interchanges, and analysis to determine the source of the fission gas; removal of the defective subassembly (or subassemblies) responsible for the release; and obtaining assurance by means of operating checkouts that the reactor is functioning normally and safely in every respect.

Phase I was approved on May 26 and conducted on May 27, 1967, with the action and results listed below:

(i) The interlock checks were completed. All control and safety rods were first manually driven through their normal operating range to detect any evidence of binding; none was noted.

The scram times of all control and safety rods were measured in accordance with standard procedures. The control-rod scram times varied +12 to -20 msec from the drop times measured on April 24, 1967, at the start of Run No. 25. Maximum time was 302 msec, compared to the maximum allowable of 325 msec. The safety-rod scram times varied from 900 to 594 msec in four measurements, compared to a 1000-msec maximum allowable time. These measurements indicate no significant changes in control- and safety-rod scram times.

All reactor nuclear instrument channels and all area radiation monitoring system channels were checked and found satisfactory. The Fuel Element Rupture Detector (FERD) and the charged-wire Fission Gas Monitor (FGM) were checked out and found satisfactory. All interlock checks were satisfactory.

(ii) The standard approach-to-critical procedure was carried out. The inverse-count-rate plot indicated no significant change in core reactivity conditions.

(iii) The control-rod positions at 50 kW indicated a reactivity gain of 5 lh compared to a prediction based on the critical taken on April 26, 1967. This reactivity prediction included a temperature correction and a correction of approximately 70 lh for burnup. The value 5 lh is within the expected calculational uncertainty, and it is concluded that there has been no significant change in core reactivity.

(iv) The relative reactivity worths of pairs of control rods were determined by an exchange technique. These results were compared with the difference between rod worths measured by periods at the beginning of Run No. 25. The differences were 0 to $4\frac{1}{2}$ lh. This deviation is considered acceptable.

Neither the FERD or the FGM showed any increase in level during the 3 hr of operation at 50 kW as compared to the shutdown condition. Gamma-ray spectroscopy of cover-gas samples taken before and during the 50-kW operation confirm that no measurable amount of fission gas was released.

It is concluded that there has been no significant fuel movement or reactivity change since the start of Run No. 25.

Phase II of the plan involves reactor operation at a power level sufficient to provide an increase in the activity of the blanket gas and to obtain an increase of signal of the fission-gas monitor to provide reference levels. This, in turn, will be used as a basis for comparison after replacing subassemblies, one or more at a time, followed by a period of reactor operation at the same power level that produced the increased signal. The program of subassembly replacement is planned to start with the high-burnup experimental subassemblies first and to proceed in a systematic manner to the maximum-burnup driver fuels if necessary. At month's end the implementation of Phase II had proceeded to a maximum power of 2.5 MWt. The reactor power was increased in steps of 0.5 MWt, remaining 30 min at each step. Cover-gas samples were obtained, and the signals on the FERD and FGM systems were carefully monitored at each power level. No increase was noted in any of the above measurements. Finally, the reactor was held at 2.5 MWt for a period of 13 hr without observing any fission gas release. The reactor was shut down at 0800 hr on May 30, pending the formulation of additional plans and the answering of questions raised by RDT. These will be submitted to the AEC for concurrence before proceeding to higher power levels.

The nature of the fission product gas release was diagnosed. From an extrapolation of the record given by the FGM strip-chart recorder and the results of the previously made bare-pin calibration studies, it was concluded that the failure involved the equivalent of 185 Mark-IA fuel elements. The results of similar calculations for a prototypical encapsulated oxide test specimen indicated that the signal recorded by the FGM system could be explained in terms of the failure of a single specimen.

c. Subassembly Transfers. Four core-type and one sixth-row fueled subassembly were received from the Fuel Cycle Facility. Thirteen spent outer blanket subassemblies were transferred to the Fuel Cycle Facility. One materials surveillance subassembly which was removed from the reactor after long-term irradiation was transferred to the Fuel Cycle Facility for disassembly and inspection.

d. Operational Support and Maintenance. The Fuel Unloading Machine (FUM) was disassembled for inspection of the transfer port and internal gas passages. In general, the deposition of sodium oxide in the FUM was fairly thin; all of the lines were cleaned; the port was cleaned and the unit has been reassembled.

The vapor traps of the FUM argon system were replaced with new traps. The gas purifier was replaced with a stainless steel-mesh filler. The outlet nozzle of the primary-tank-argon cooling system was inspected and found to be quite clean. The piping on the outlet of the nozzle to the first vapor trap was disassembled and a considerable amount of sodium was removed. New bonnets and reinforced operator supports were installed on the pneumatic valves on the suction or negative pressure side of the argon system.

New cam rollers were installed in the storage-basket Ferguson (rotational) drive, and realignment of the vertical drive rollers has proceeded.

The startup feedwater pump was operated for 129 hr this month. The cylinders were repacked three times, and two reconditioned plungers were installed.

The overspeed-trip governor for the main turbine was re-adjusted after receipt from the G. E. shop. After adjustments the turbine tripped at about 3920 rpm with close repetition.

The installation of the new secondary-sodium-sample station continued.

e. Operator Training. Current activities in training are involved with the reorganization and rewriting of the reactor-control-console power plant and electrical training material. Three revised training units were completed in the reactor control console area and two in the electrical area. A training unit consists of a study guide, open-book assignment, and closed-book test.

Five operators completed oral examinations prior to qualification. These include: three fuel handling operators, one electrical operator, and a reactor control console operator.

f. Operating Manual. To maintain a usable, well-integrated Operating Manual which is necessary to assure safe and effective operation, a process of periodic review and revision is carried on. Plant modification and operating experience require the preparation of new procedures or extensive revision to existing sections of the Manual.

Issuance of Special Operating Procedures is a continuing effort so that the operations personnel are supplied with the most current procedures possible.

Review and updating of the EBR-II Operating Manual continued. Division VIII-E, "Fuel Handling System - Special Procedures" is being revised to reflect recent modifications and will be issued next month. All sections of Division X, "Miscellaneous Systems" were reviewed. Those sections requiring revision have been rewritten and will also be issued.

Photographs of thirty-eight (38) system and equipment panels which have been modified were retaken. The drafting section has completed preparation of these figures for inclusion in the Operating Manual.

The following Special Operating Procedures were issued:

- (i) Bulk Sodium Level Reactor Safety Trip.
- (ii) Nitrogen Purge to the Small Plug Liquid Metal Seal.
- (iii) FUM Argon Cooling System Flow Indications.

2. Fuel Procurement

Relative to Mark-IA-IB fuel procurement, recommended vendor facilities were visited, and discussions were conducted aimed toward concluding a contract for supply of 34,000 Mark-IA fuel elements. Contractual arrangements should be completed very shortly. Production should be initiated in April 1968, and completion of the contract is expected in February 1969. An option for further procurement must be exercised prior to completion of this contract (proposed to be within 2 weeks after delivery of 20,000 elements). Following tests of Mark-IB and Mark-II fuel elements, decisions will be reached on modification of vendor production.

3. Reactor Improvements

a. Irradiated-fuel Monitor. Hardware fabrication is ~98% complete. In preparation for preliminary tests, the upper end of the thimble will be filled with heavy concrete and the various mechanisms will be assembled.

b. Improved Gripper and Holddown Force-limiting Device. The design has been completed and fabrication and procurement have begun (see Progress Report for February 1967, ANL-7308, pp. 3-4).

c. Sodium Chemistry Technology. In connection with analyses of possible causes of and problems from copper contamination in the primary sodium system (see Progress Report for March 1967, ANL-7317, pp. 4-7), the original plugging-loop (PI-A) economizer is being examined destructively. Plans are being made for a new plugging loop and sampling station.

(i) Examination of Plugging-loop Economizer. The original PI-A plugging-loop economizer is being examined destructively so as to determine the extent and nature of any copper deposits that might be present on the surfaces that had been in contact with sodium. The economizer is being cut into several longitudinal sections; samples of frozen sodium from each section will be analyzed chemically.

The radiographic examination of the entire economizer and the first results of the examinations after slitting open the shell halves were described previously (see Progress Report for April 1967, ANL-7329). Since then, further examination shows that the shellside transition region (between where copper deposited in the lower section but not in the upper section) occurs along a 6-in. section 4 to 10 in. from the bottom of the economizer. Sodium samples and deposits recovered from the surfaces of all the tubing sections are being analyzed to provide a copper concentration profile along the entire flow path through the economizer. The X-ray diffraction powder patterns show copper (face-centered cubic) to be the major component, with possibly Cu_2O (cubic) as the minor component, in the deposited material.

d. Rotating Plug Seal Cleaning System. Development continued of a centrifugal-type separator to remove oxides from the tin-bismuth alloy. A 10,000-rpm separator mockup is being tested with a homogenized oil-water fluid to determine flow characteristics and separation efficiency. If results prove satisfactory, a tin-bismuth and alloy mixture will be tested.

e. Irradiation Subassembly Mark-C19. The entire series of hydraulic flow tests were completed with the water loop to determine pressure drop and flow rates for the 19-element test-section array. The same order and procedures were used for these tests as were used for the Mark-C37 flow tests (see Progress Reports for January 1967, ANL-7302, pp. 13-14, and February 1967, ANL-7308, p. 3). The tests verified preliminary calculations indicating that the lower pressure drop in the 19-element test section would result in a reduced bypass flow of ~10% at maximum bypass flow conditions. No mixing studies were performed because it was evident from the Mark-C37 ink-injection tests that efficient mixing occurs. The fact that efficient mixing occurs with this design also

permits an additional benefit: space at the upper end of the subassembly can be used for a 3-in. length of stainless steel radiation shielding in both the Mark-C37 and Mark-C19 subassemblies. Figure 1 shows the final design arrangement of the Mark-C subassembly.

f. Fuel Subassembly Mark II. Upper and lower radiation-shield sections for the Mark-II subassembly were flow tested and helix-type restrainers for the Mark-II fuel elements were investigated (see Progress Reports for March 1967, ANL-7317, pp. 20-21 and April 1967, ANL-7329).

(i) Model-STS Shield Section. The Model-STS shield section for the Mark-II subassembly is the same as the standard Model-STB (triflute) shield section for the Mark-I subassembly except for having reshaped flow-passage entrances to decrease the pressure drop. In addition, because the Mark-II fuel elements are 6 in. longer than the 18-in. Mark-I fuel elements, the Model-STS upper shield section is only 15.25 in. long (as compared to 19 in. for the Model-STB upper section). Total weights of the Model-STS upper and lower shield sections are 10.0 and 13.6 lb, as compared with 12.8 and 14.2 lb for the corresponding Model-STB sections.

Flow tests in a water loop (139 gal/min) show that the pressure drops in the Model-STS upper and lower sections are 0.5 and 3.7 psi less than in the corresponding Model-STB sections. This total 4.2 psi lesser pressure drop with the Model-STS shield pieces approximately fulfills the requirement for the proposed Mark-II fuel subassembly. Two additional variations in radiation-shield designs for the Mark-II subassembly are being fabricated for similar flow tests.

(ii) Fuel-pin Restrainers. The Mark-II fuel elements have ~9 in. of gas space above the fuel, which is about three times as long as in the Mark-I fuel elements. The column buckling strength of the 40-mil-wire type of restrainers investigated for the Mark-IB fuel elements (see Progress Report for January 1967, ANL-7302, p. 10) is inadequate for 9-in. lengths. Thus a helical restrainer concept is being investigated. The design consists of a 0.010-in.-thick stainless-steel strip that is 0.152 in. wide and is twisted along its ~9-in. length into a helix having an ~3-in. pitch.

Five samples, having various pitches of 2.5 to 3.25 in., were enclosed in standard fuel-element cladding tubes and were compression tested with a test machine using the same fixtures and techniques that were employed in testing the Mark-IB wire-type restrainers (see Progress Report for October 1966, ANL-7267, pp. 8-10). These tests show that this restrainer can support a force of 72-98 lb before the tube wall deforms to the extent of 0.002 in.; the various pitches have no noticeable effect. This range of force, which is equivalent to that for the shorter Mark-IB wire-type restrainers, would adequately prevent fuel-pin displacement within the tube of the Mark-II fuel element. The investigations are continuing.

Fig. 1. EBR-II Irradiation Subassembly Mark-C

g. Primary-tank Sodium Pump and Filter. Design work has been completed on a filter and pump assembly that will provide a method for determining the amount, type, and size of impurities that might exist in the region of the primary-pump intake (see Progress Report for January 1967, ANL-7302, p. 13). Component fabrication has begun.

h. Oscillator Rod Mark II. Design studies have been completed on the changes and modifications to the original Mark-II oscillator rod (see Progress Report for April 1967, ANL-7329). Details of these modifications are being designed.

i. System Design Descriptions. Work is proceeding in the preparation of EBR-II System Design Descriptions. These will make readily available design and technology information on EBR-II that is pertinent to plant and equipment designers and to utility operators for their use as guidelines in the design of safe and reliable future liquid-metal-cooled fast breeder reactor power plants. The descriptions will stress reactor-related equipment and will include unique features affecting safety, reliability, and performance; significant modifications to the "as built" facility also will be described. It is anticipated that the design descriptions will be contained in five volumes with major topics as follows:

Vol. 1 General EBR-II Facility Description

Vol. 2 Primary-system Description

Vol. 3 Secondary-system Description

Vol. 4 Steam-system Description

Vol. 5 Electrical-system Description.

This program has eight major milestones: (1) organization of source data, (2) development of working outlines, (3) preparation of first-draft manuscripts, (4) review of first-draft manuscripts, (5) preparation of second-draft manuscripts, (6) review of second-draft manuscripts, (7) preparation of final copy and illustrations, and (8) preparation of printed publications. Organization and indexing of the significant source data are well under way. Preliminary outlines have been prepared and distributed for review. Detailed working outlines are being prepared.

j. Primary Tank Pressure Relief. The normal control system for the argon cover gas may not be capable of handling unusually rapid vacuum or pressure changes within the primary tank fast enough to prevent damage to the tank. It is necessary that a pressure-vacuum relief system be provided as a backup for the normal system to provide protection for the tank in the event of such a pressure transient. The new pressure-relief system furthermore will protect the primary tank if the normal system malfunctions or if the pipeline connecting this system to the primary tank should become plugged with sodium.

A pressure-vacuum relief device has been fabricated and will be connected to a nozzle in the cover of the EBR-II primary tank. This device will provide a protective oil loop seal between the reactor building environment and the primary-tank argon cover gas while allowing the tank to discharge, either inward or outward, to relieve excessive differential pressures between the primary tank and the reactor building.

This system is presently under final analysis to establish firmly the limits of pressure relief, the physical location of all components and piping, and to determine the best method for providing a discharge from the system.

k. FERD Loop Plugging Meter and Sampler. The standard method of obtaining a sample from the primary-sodium system is at a point in the primary purification system. The plugging meter for the primary system is also located in the primary purification system. During the scheduled replacement of the primary purification system there will be a need for an alternative sampler and plugging meter. Since samples taken from the FERD loop would be very representative of the coolant flowing through the EBR-II reactor core, this sampler and plugging meter will be installed in this loop. Continuing use of these units for analysis of samples and plugging temperatures obtained from the new system will also serve to verify results obtained from the primary purification system.

The operation of the reactor is dependent upon the operation of the FERD loop. Therefore, a careful analysis of any possible interaction with the FERD loop is necessary to prevent possible interference with the operator of the loop.

The design of a plugging meter to be installed in conjunction with the FERD loop has been approved contingent upon the successful proof-test operation of the plugging meter in the secondary purification system. Installation of this plugging meter in the secondary system has been completed.

All material and component parts for the installation of the FERD loop plugging meter are on hand. Certain necessary installation drawings remain to be completed.

The installation of the FERD loop plugging meter and sampler is scheduled following the completion of EBR-II Run No. 25. The installation of the sampler will be subject to approval by the EBR-II Plant Modification Committee.

l. New Plugging Meter for the Primary and Secondary Purification Systems. The original plugging meters in use in the primary and secondary purification systems present operating difficulties. Accurate determination

of the plugging temperature is difficult to obtain. Maintenance of these instruments is frequent and for the most part due to failed valve bellows. There is a need for an improved plugging meter which is simple to operate, comparatively easy to maintain, and which provides consistent information.

A plugging meter design should have the following features:

- (i) an air-sodium heat exchanger to provide a controlled cooling rate;
- (ii) a sequenced heating system to avoid damage to valves; the heating system should be so designed that the lowest temperature in the system during a plugging run is at the plugging valve;
- (iii) a simple operating procedure.

The design of an improved plugging meter for the primary and secondary purification systems is essentially complete. The new plugging loop design is not different in principle from the design of the original plugging loop in use in the secondary purification system. However, attention has been given to provide a greater ease of operation. The new design includes an air-sodium heat exchanger. The flowmeter and throttle valve are located ahead of the economizer in a constant-temperature area to avoid perturbations from temperature variations. The heating system for the plugging loop is designed such that the lowest temperature in the system during a plugging run is located at the plugging valve. The plugging valve consists of a modified 1/2-in. NUPRO 8UW valve. The valve plug was modified by cutting ten 0.040 x 0.040-in. grooves in the face of the plug. The plugs have been returned to the vendor for installation in the valves. Delivery for the completed valves has been quoted at four weeks.

Upon completion of the instrumentation and electrical design, the engineering package will be presented to the Plant Modification Committee for review and approval.

m. Fuel Unloading Machine (FUM) Revisions. Operating experience with the FUM dictates that, although operation has been adequate to date, modification of the system is required for long-term reliability. A large maintenance effort has been necessary to date on this system. With the emphasis on increasing plant factor, the operating reliability of the FUM must be substantially improved to insure its availability for the increased usage which will accompany higher plant factor.

The modification work planned for the FUM consists of two phases: the first is the entire mechanical system and the second, the argon gas system. Work on the mechanical system will include design of a new gripper, a redesign of the entire position-indication system, elevation drive system, and jaw-actuating mechanism.

Two basic problems associated with the FUM argon system are in-leakage of air and entrained sodium and sodium vapor in the gas stream. As a result of air in-leakage, sodium oxide deposits built up in the area of the gripper and also in other parts of the system. These deposits have critically affected the operation of the FUM during past operating periods.

The procedure to eliminate these problems will be to detect all leakage and eliminate its source, to devise a more effective method of separating entrained sodium from the gas stream before its entry into the main system, and to improve filtration.

(i) Mechanical System. Flow testing of a prototype Mark-III FUM gripper was completed, and the unit has been installed in the FUM for final checkout. The new manual jaw actuator was installed and has been operated in air with the new gripper. Several modifications were made in the position-indication system to provide proper operation with the increased jaw travel used in the new gripper. All internal FUM piping has been cleaned, and the machine is ready for the entire system.

(ii) Argon Gas System. The external parts of the argon gas system have been made essentially leaktight. The ball valves have been modified by use of special bonnets and silicone O-rings. As a result, in-leakage through these valves has been essentially eliminated. Continuing attention will be necessary in this area.

Evidence indicated that because of the low pressure drop across the horizontal molecular sieves, the bed had settled and the gas stream was channeling over the top of the bed. To eliminate this problem, a design incorporating baffles has been completed and will be installed at the earliest opportunity.

n. Nuclear Instrumentation Improvements. Improvements will be made in the nuclear instrumentation to reduce the maintenance caused by radiation damage in the thimbles and to improve reliability, accuracy, safety, and plant factor.

Circuit design to integrate the new equipment into the circuit now in use has continued. Design work necessary for the source range channels is in the final stage. Preparation of the material required for safety review is in progress.

The nuclear equipment will be supplied with an IBM Cat. No. 766046 reed relay. This relay has been checked by loading with the normal circuit load of a second relay coil. The relay has performed satisfactorily for 10 million operations.

A second, heavy-duty relay is required to act in conjunction with this. This relay has been tested with the anticipated load and has operated successfully through 10 million operations. We are now attempting to cause this relay to fail by overloading in order to have an available history on the mode of relay failure.

(i) Temperature-monitoring Devices. A program to replace temperature-monitoring switches (shutdown-initiating components), which load the primary sensors, has been started. The new units will provide improved accuracy and reliability.

The engineering details have been completed, and work has started on panel fabrication and circuit interconnections. The equipment delivery from Leeds and Northrup Company has been set back to the second week in July 1967. Work will continue on the panel as far as possible, pending arrival of these components.

(ii) Multipoint Recorders. These units, for which there are no existing standbys, are required for operation of the reactor. A program will provide for scheduled replacement of these units to permit factory rebuilding and upgrading. Extra units will then serve as standbys when the program is completed.

Equipment delivery has been delayed by the Leeds and Northrup Company until the second week in July 1967. No further work is possible until this equipment is received.

(iii) Miniature Recorders. A program to isolate the indication and control function related to several critical parameters to provide improved accessibility for maintenance has been started.

Equipment delivery has been delayed by the Leeds and Northrup Company until the second week in July 1967. No further work is possible until this equipment is received.

(iv) Selected-parameter Data-processing System. This system provides: (a) a 50-point system for alarm and surveillance of selected parameters, such as core subassembly thermocouples for reactor protection; and (b) a 100-point system for routine surveillance of plant parameters and for nonroutine analysis of system dynamics. This will also provide improved data acquisition for reporting plant performance.

(α) 50-point System. Installation of this system was completed during this reporting period. Operational checkout and debugging are in progress.

(β) 100-point System. Factory checkout of the equipment is in progress, but final delivery will be held up two or three weeks pending receipt of the incremental tape recorder. A sample tape of data has been received from the manufacturer. This tape has been used as input for the local computer for checking compatibility.

Interconnecting cable and cable trays between the master patch panel and the test equipment room have been installed. Terminating of these cables will be held up until equipment has been received and checked out by a factory representative.

(v) Pressure and Temperature Sensors. This is a packaged assembly of two pressure transducers and eight thermocouples to monitor temperature and pressure in the upper plenum of the reactor, replacing failed units in that area.

The design of the thimble assembly is essentially complete and is currently being reviewed.

o. Relief Valve for Main Steam Header. The relief valve on the main steam header in the Power Plant is lacking in dependability. Although this valve is not necessary for plant safety, it is nevertheless unsatisfactory. It has never popped or reset at consistently constant pressures and has been a high-maintenance item. It was therefore decided to obtain a suitable replacement, one that would overcome the shortcomings of the existing valve.

From an intensive study, it was concluded that an electrically controlled, pilot-actuated relief valve, such as the Crosby-Ashton Pressur-matic Vent Valve, would be the solution to our problem. This valve is capable of adjustment from 1 to 5% blowdown. It can also be operated manually by a three-position electric panel switch. When operated manually, it can be opened and closed at any time, at any pressure higher than 50 psig, and will remain fully open as long as desired, even at pressures as low as 50 psig. In utility service it has proven to be dependable and relatively free of maintenance.

Normally this will be the "working" valve, that is, it will be set to be open before the drum safety valves release. It will therefore reduce the duty and associated maintenance of these valves and thereby enhance their dependability. It has the further advantage that it can be maintained while the steam system is pressurized and the plant is at power.

This relief valve will be placed upstream of the main block valve in the sodium boiler plant so that it can relieve boiler pressure whether or not the block valve is open.

Design for this installation is complete, the valve is on hand, and work can begin when the steam plant is depressurized for a sufficient length of time in concurrence with other planned work.

4. Reactor Analysis and Testing

→ a. Reactivity Feedback Tests. To provide additional information as regards understanding of the weakened power coefficient in EBR-II, additional tests were carried out. The results of each test were found to be consistent with a feedback model in which power response is sensitive to two opposing effects, viz., fuel expansion and subassembly bowing.

(i) Tests with Reduced Flow. With the flow fixed at 90% of the full rated value of 9000 gpm, the power of the system was raised incrementally from 50 kWt to 22.5 MWt. Values for the reactivity inserted into the system through control-rod action yielded a curve similar in shape to the power-coefficient curve established at 100% flow but lying consistently lower, by approximately 3.2 lh at 22.5 MWt. This implies that over the power range from 0 to 22.5 MWt a reduction in flow results in a slight increase in the reactivity of the system.

In the next series of tests, reactor power was held constant at 22.5 MWt while the flow was decreased from 100, to 90, to 75, to 60, and finally to 54%, of full rated value. At each new condition of flow, extreme care was taken to establish control-rod positions for equilibrium conditions of power. The results demonstrated that reducing the flow from 100 to 75% resulted in a net gain of 2.5 lh, and reducing the flow further from 75 to 54% resulted in a loss of 6.6 lh. Summing the results for the two ranges of reduced flow gives an overall (net) loss of 4.1 lh when the flow is reduced from 100 to 54% of full rated value.

To establish the sign of the flow coefficient at higher levels of power, the flow rate was decreased from 100 to 95% at 42.5 MWt. The results indicated that for this power region reduction of the flow causes a slight reactivity loss, 2.6 lh. The change in sign, negative at 22.5 MWt and positive at 42.5 MWt, is consistent with the model described below.

(ii) Rod-drop Tests. In order to better understand the competition between the negative effect due to fuel expansion and the positive effect due to subassembly bowing, rod-drop tests were conducted for fixed reference conditions (50 kWt, 22.5 MWt, and 41 MWt, all at 100% flow) and for each condition of reduced flow. Although the results of these tests are still being analyzed, preliminary examination of them indicated that the response of the system, even under conditions of reduced flow, is consistent with a feedback dominated by prompt, axial, fuel-expansion effects, but influenced by a competition between more strongly delayed negative and positive components. During the next three months, a concerted effort will be made to translate the results of the rod-drop tests into time constants and power-coefficient components for individual feedback processes.

(iii) Experiments at Constant ΔT . As an integral portion of the tests with reduced flow, a successful attempt was made to establish the reactivity change associated with a significant reduction in reactor power and flow rate. Approximately 8.0 lh were lost in the change from 41.5 MWt and 100% flow to 22.5 MWt and 54% flow (ΔT in both cases, 122°F). An analysis is being made to interpret these results in terms of a power coefficient for the prompt axial expansion of fuel.

(iv) Forced Perturbations of Reactor Power by the Trapezoidal Insertion of Reactivity. In these tests the shape and time dependence of the output power wave was studied while reactivity was alternately inserted and withdrawn through the programmed movement of a control rod over a fixed distance in the linear portion of the control rod-reactivity worth curve. A visual examination of the power dependence during the tests revealed that the kinetic behavior of the system, under conditions of both full and reduced flow, is consistent with an overall negative power coefficient dominated by the influence of prompt feedback for the axial expansion of fuel. A substantial effort is being made to interpret the results of trapezoidal reactivity-insertion tests directly in terms of the reactor transfer function.

(v) Coherent Power Perturbations. During the course of the reduced-flow tests, studies of coherent power perturbations were carried out. Upon reaching equilibrium, after each reduction in flow, reactor power was permitted to drift. Shortly following the fixing of control-rod position and the flow rate of the secondary coolant, without any external forcing, the power began to vary sinusoidally with a frequency ranging from 0.031 to 0.066 cycle/min, with a decided tendency for higher frequencies at higher ΔT values. Without exception, the amplitude of the power variation grew to a maximum during the first three cycles with the largest amplitude observed (at 22.5 MWt and 75% flow) varying within the limits of $\pm 6\%$. The results of these studies will be used to substantiate or to reject a postulated mechanism based on a feedback loop between the heat-exchanger discharge and primary-pump inlets.

(b) Model for the Weakened Power Coefficient in Run No. 25. As discussed in the Progress Report for April 1967, ANL-7329, Run No. 25 was noticeably characterized by a strong decrease in power-associated feedback. Although many changes were made in the loading for Run No. 25, the most significant was the replacement of depleted uranium blanket sub-assemblies in rows 7 and 8 with stainless steel-filled reflector units. The considerable amount of test data leads to the conclusion that thermal deformation effects in the reflector region are primarily responsible for the changes in feedback noted. Although subject to future revision and refinement, a physical model consistent with experimental observations has been formulated.

To appreciate the importance of features incorporated into the loading of Run No. 25, it is instructive to review the feedback characteristics of earlier loadings. In these, strong temperature gradients existed across the row 7 and 8 blanket subassemblies, with the sense of the gradient such that inner surfaces of the hexes were hotter than outer edges. As a result, such subassemblies tended to deform in a manner which effectively moved upper ends outward with respect to the core as reactor power increased. In effect, this action provided more freedom for outward motion of core subassemblies at upper ends as they, too, bowed under temperature gradients. Accordingly, the upper ends of core subassemblies were free to move radially until outward bowing effects in blanket subassemblies were completed. From this point on, additional core-deformational effects were manifested in the form of an inward bowing (at the midplane) which was sensed by the reactor as a positive reactivity feedback. Evidence substantiating this mechanism has long been noted as a decrease in the differential power coefficient in the vicinity of 20 MWt. Upon closure of midplane clearances, further power increases were reflected by a contiguous overall radial expansion of the core, an action which tended strongly to increase the differential power coefficient.

For Run No. 25, on the other hand, heat flow from row 9 blanket subassemblies into row 8 stainless steel reflector units causes the outer surfaces of row 8 to be hotter than inner surfaces. A reverse gradient of this nature causes row 8 subassemblies to deform in a manner such that upper ends move towards the core center as the power increases. As an immediate consequence, the upper ends of core subassemblies find constraint at a lower level of power (10 rather than 20 MWt). Since less thermal expansion has occurred in core subassemblies at the lower power level, a proportionately greater decrease in core radius can be accommodated before the core reaches maximum compaction. The additional space available for inward bowing, under these conditions, is apparently not completely occupied until the power level increases beyond 35 MWt. The fact that the power coefficient in the region 35-45 MWt still lies below that for earlier loadings, even though the core may be fully compacted, may be explained in terms of a gap which exists between the tops of row 8 and row 9 subassemblies. In effect, rows 7 and 8, closed tightly at the top by reversed bowing effects, constitute a constraining barrier which prohibits contiguous expansion effects into more outlying regions of the blanket. Accordingly, negative expansion effects which formerly applied over the entire core and blanket regions, are now limited to the core and the innermost row (No. 7) of the stainless steel reflector.

Essentially, then, the shape of the power coefficient curve in Run No. 25 reflects competition between components which may be either positive or negative, with the latter always emerging as the stronger, although to a smaller extent than for previous loadings.

The results of routine tests conducted during the startup of Run No. 25, coupled with the results of more recent supplemental experiments, are qualitatively consistent with this model. The tendency of the power to drift reflects the influence of unavoidable variations in inlet coolant temperature acting through a decreased power coefficient and an unchanged isothermal temperature coefficient. Eventually it may or may not be shown that the coherence noted in power perturbations is the consequence of a transport-type lag between heat-exchanger discharge and primary-pump inlets.

Results of rod-drop test, while still under review and analysis, indicate the existence of decreased negative components in the time region 2-15 sec. This evidence strongly suggests either one or both of the following: greater restraint on overall core and blanket expansion, or an increase in bowing effects which tends to move subassemblies inward at the top after complete contact is effected between row 6 subassemblies and those of the barrier formed by the stainless steel reflector units in rows 7 and 8.

Additional testing, coupled with more rigorous analyses during the next six months, is expected to produce a model, similar to that described above, which is qualitatively consistent with experimental results and quantitatively in accord with actual changes in the magnitude for the reactor power coefficient.

5. Surveillance

a. Effect of Recycling Types 304 and 304L Stainless Steel Tubes.
To evaluate the effect of storing tubing of Types 304 and 304L stainless steel (SS) in air after exposure to sodium, which is important for determining whether irradiated test samples can be returned to the reactor, a four-phase test program is being conducted. Phase-I tests have compared the corrosion and sensitization, as measured by electrical resistivity, of Types 304 and 304L SS that had been exposed to 1100°F sodium in a loop for 7 days, oxidized in air for 1 hr, steam-cleaned for 15 min to remove sodium from inside the tubes, and washed by immersion in demineralized water for 45 min. The grain structures as observed in photomicrographs showed that the temperature and sodium had sensitized the Type 304L SS somewhat, but that the Type 304 SS had been highly sensitized.

The specimens then were subjected to a Strauss Reagent corrosion test and electrical resistivity was measured at intervals, with the results shown in Fig. 2. Within only 4.2 hr, the Type 304 SS samples were in various stages of disintegration, the Type 304L SS samples showed some evidence of attack, and the as-received (had not been exposed to sodium or to 1100°F) Type 304L SS was unaffected. There were eight specimens of each of the four kinds of material. As contrasted with the Type 304 SS specimens, the data spreads within each of the two types of Type 304L SS specimens were quite modest. The lowest curves in the shaded regions

Fig. 2. Rates by Which Strauss Reagent Changes Electrical Resistivity of Types 304L and 304 Stainless Steel, and Electropolished Type 304 Stainless Steel Tubing That Had Been Exposed to 1100°F for 7 days

for the Type 304 SS and electropolished 304 SS are each for one specimen; the curves at the top of the two shaded regions each represent data from five specimens; the topmost curves for these two types of Type 304 SS each represent data from two specimens. The data spread might be caused by the seam weld or by differences in tube fabrication or grain size; this will be further investigated by identifying the location of the specimen in relation to the seam weld.

In the meantime, to prepare for Phase-II tests that will determine whether storage will affect the sensitization and corrosion during subsequent recycling in sodium, Type 304L SS specimens that had been exposed to sodium and washed with the Phase-I batches are being stored separately in air and in a container with CaSO_4 desiccant. The samples to be tested in Phase II will have been stored for 10 days. Phases III and IV will involve longer storage.

b. Materials Surveillance Subassembly. SURV-I, the first of ten reactor-materials surveillance subassemblies placed in the reactor on March 1, 1965 (see Progress Report for March 1965, ANL-7028, p. 1) was partially dismantled in the Fuel Cycle Facility. Sodium trapped within the subassembly was removed by reaction with isoamyl alcohol, and a succeeding wash with ethanol, to preclude high-temperature chemical attack of the metallic surfaces by sodium hydroxide when the standard cleaning procedure using water and steam is used (in fuel-element decanning).

During the dismantling of the assembly, preliminary observations indicated that the various stainless steel parts of the subassembly accumulated, or liberated by corrosion, crud differentially: (i) the upper removal extension shaft was covered with a dark brown and fairly adherent coating, and several large blobs of white material, probably sodium-alcohol reaction products, were attached to the gripper extension; (ii) the outer hexagonal can was almost completely clean, although there were small dark areas at the bottom and the top; (iii) the outer stainless steel cylinder of the stellite bushing was covered with a very dark brown rusty coating, and the lower adapter was differentially filmed over with dark corrosion products; a light film very lightly masked the finely ground surfaces in the two grid-plate bearing regions and the shaft was more heavily coated in between.

When the contents of the hexagonal can were recovered, the metallic surfaces were examined. The stainless steel cans encapsulating the three samples of graphite were heavily coated with a dark gray, strongly adherent product. The tie rods, hex guides, nuts, and Belleville-spring washer of the canned graphite subassembly were similarly coated. The force effects of the Belleville-spring washer were not determinable beyond establishing that the three cans were still tightly held together.

All 19 tubes containing the EBR-II materials of construction were bright and shiny. All 12 filler strips in the 19-tube array were coated with a rusty corrosion product. A disc of newly machined yellow brass was introduced into the cave to permit better judgment of the appearance of the aluminum-bronze labyrinth under the yellow sodium lights illuminating the caves. The exposed labyrinth was coated with a thin film of dark corrosion products through which was seen the characteristic yellow color of the bronze. All the machined edges were sharp and there was no visible pitting. A preliminary micrometer measurement of the diameter indicated little or no change of dimension. Variations in notch ductility were observed when the 19-tube cluster was being disassembled. As few as four complete bends through an angle of 30-45° to as many as 12 bends were required to break the connecting notched pin between the tube and the hexagonal grid plate; about two-thirds of the tubes broke after eight bends.

Tube No. 10, which contained four flux-monitor wires (Cu, Ti, Fe, and Ni) was scanned for gamma activity; a maximum of 4 R/hr was measured at the middle. A similar scan of one of the canned graphite sections showed 25 R/hr. A swab from one of the surfaces (50 mR/hr) was taken for the identification of composition of the deposits on the can surface.

Plans were made for more detailed examination of the surveillance specimens.

6. Driver Fuel Development and Metallurgical Assistance

a. Investigation of Anomalous Swelling of Mark-IA Fuel. The nature of the problem and the general status of the investigation have been described (see Progress Report for March 1967, ANL-7317, pp. 8-12). The entire task has been subdivided into 11 subtasks and progress will be reported in this format.

(i) Irradiation of MC-S Type Fuel. Status unchanged--burnup being accumulated.

(ii) Encapsulated Irradiation of SL-type Fuel. The burnup limit for EBR-II driver fuel is set at such a value that fuel swelling will produce a calculated pressure in the gas plenum of about 330 psi and a jacket hoop stress of 3000 psi. These pressure and stress limits are conservative and safe, as made evident by the complete absence of failed fuel elements in the EBR-II driver-fuel subassemblies. Before we can consider raising the limits, it is necessary to perform capsuled irradiation tests to higher burnups under conditions such that a failed fuel element will not jeopardize continued operation of the reactor. Seven fuel elements of the Mark-IA design were fabricated with pins of the rapid-swelling SL type.

Five elements were loaded into irradiation test subassembly XO17 and two into XO15. Target burnups are 1.5% for XO17 and 3.0% for XO15. Irradiation started with reactor Run No. 23. The elements reached a burnup of only about 0.2% before the reactor was shut down for modification.

(iii) Effect of Design Change from MK-I to MK-IA Status. Status unchanged--burnup being accumulated.

(iv) Effect of Enrichment Change. Status unchanged--burnup being accumulated.

(v) Effect of Cooling Rate. Status unchanged--burnup being accumulated.

(vi) Effect of Trace-element Impurities. Fuel pins from two virgin alloy melts, one with and one without deliberate additions of trace-element impurities, were cast at Lemont and have been shipped to Idaho for fabrication into fuel elements. Identity with respect to position and orientation within the casting array was maintained; hence this experiment supplements Test (v) above. The specifications and available analytical results are tabulated below in Table I.

TABLE I. Trace-element Impurity Melts

Melt No.	Type	Impurity Elements (ppm)					
		Carbon	Silicon	Aluminum	Iron	Oxygen	Nitrogen
409	SL--no additions	141	-	20	100	20	17
410	With additions	238/244	779/790	1790/1870	100	29/199	5/10
410	Specifications	200/300	350/500	600/900	<100	-	-

The specified silicon and aluminum contents were missed rather badly in Melt 410. Microprobe analysis shows that the aluminum is uniformly distributed throughout the alloy. A high-silicon phase associated with precipitated U_2Ru appears at the grain boundaries. The response of this phase to heat treatment is being studied. The assembly of these pins into fuel elements is being held up pending the results of the heat-treatment study and the results of isotopic analysis.

Specifications for three more melts containing deliberate additions of trace-element impurities have been drawn up. These castings are to be prepared at Idaho from virgin melt stock, using the FCF ingot-preparation and pin casting procedures. The specifications are tabulated below in Table II.

TABLE II. Trace-element Melts

Melt No.	Trace-element Additions (ppm)			
	Carbon	Silicon	Aluminum	Iron
A	200/300	350/500	<100	<100
B	200/300	350/500	250	750
C	200/300	350/500	750	250

(vii) Effect of Pressure in Swelling of Mark-IA Fuel. The Mark-IA fuel loading of EBR-II was noted to have a higher swelling rate than the Mark-I loading, attributable either to differences in the fuel material or to the lower pressure in the longer plenums in the Mark-IA fuel. The two variables of pressure and heat treatment are being investigated separately in special capsule tests in the CP-5 reactor.

Inch-long sections of fuel castings of known thermal history are being enclosed in strong jackets under varying pressures, and will be exposed to 1.2 to 1.5 a/o burnup to see the part pressure plays in inhibiting swelling. A range of pressures ranging from slightly above atmospheric to about 2000 psi will be studied. Complete metallographic records are being kept.

On May 22, 1967, the first capsule containing six such specimens was placed in CP-5. An irradiation period of five to six months will be needed. The initial six fuel specimens are SL-type material and were selected on the basis of high-swelling characteristics, as noted by past performance. These will be followed by fuels that have demonstrated somewhat lower swelling, and next by a confirmatory capsule for the initial series.

The third capsule and the first will be loaded with alternate sections from the same fuel pin, with test pressures running in opposite directions to cancel out the effect of graduation in casting heat treatment along the pin.

(viii) Effect of Heat Treatment. A variety of heat treatments are being investigated for use in the CP-5 irradiations. These include primarily solution heat treatments and quenched and aged structures.

(ix) Hot-lab Examination of Irradiated Fuel. There are some 10 different items in this subtask. We will report this month only on the status of the chemical analyses at Lemont. The results are shown in Table III.

TABLE III. Chemical Analyses of Irradiated Fuel Pins

Casting Batch	Pin No.	Melt Refining No.	Subassembly	Grid Position	Chemical Analysis (ppm)	
					Carbon	Silicon
4001	1	22	C-184	87	106	88
4002	9	23	C-180	60	189 ^a	260 ^a
4002	14	23	C-180	57	121	
4002	24	23	C-184	32	139	57
4002	82	23	C-184	60	146	90
4003	3	24	C-186	53	200	199
4003	5	24	C-186	52	199	212
4003	19	24	C-186	46	198	223
4003	48	24	C-184	57	192	217
4003	71	24	C-180	32	193	
4003	86	24	C-179	82	214	181
4003	90	24	C-179	81	211	204
4003	92	24	C-179	79	132 ^a	86 ^a
4004	9	25	C-184	7	108	134
SL-20	40	-	C-185	2	149	
SL-20	52	-	C-184	36	143	74
SL-21	20	-	C-186	60	153	
SL-29	51	-	C-184	43	157	102
SL-29	68	-	C-179	52	138	
SL-30	8	-	C-184	37	217	23

^aSpecimens appear to have been interchanged.

Each reported carbon analysis is an average of six separate determinations (duplicate analyses on samples taken from the top, center and bottom of the fuel pin). There appears to be no axial segregation.

Some of the silicon results are still somewhat tentative in that they represent averages of only two or four determinations. The final results should be available next month. Again no certain segregation was noted, although there was some tendency for the silicon to run lower at the midlength position.

The available results also show that different pins from the same casting batch have the same composition and that there is no pin-by-pin segregation in a given casting.

Specimens (top, middle, and bottom) from 20 fuel pins have been shipped to Oak Ridge for aluminum and iron analyses. No results are available as yet.

(x) Characterization of Unirradiated Fuel Pins. No new results this month.

(xi) Analysis of Fuel-pin Swelling. Approximately 450 data sets of fuel-pin swelling versus maximum average fuel-pin burnup have been transmitted to the Applied Math Division for programming. The burnup is that calculated by an Idaho program that takes account of the flux gradient across a given subassembly. Hence, we cannot consider the swelling versus burnup relationship pin by pin rather than by the subassembly-averaging procedure hitherto used. The effect is relatively small for subassemblies in the first few rows of the reactor core, but becomes very appreciable in the outer rows of the core. The magnitude of the effect is illustrated by the swelling results from subassembly B-331 irradiated in reactor grid position 6-A-3. The results are tabulated in Table IV and plotted in Fig. 3. The long tail toward low swelling fits in quite well with swelling values obtained from subassemblies irradiated

TABLE IV. Fuel-pin Swelling versus Burnup in Subassembly B-331 Irradiated in the Sixth Row of EBR-II Core

Grid Position	Pin No.	Swelling (% $\Delta V/V$)	Calculated Burnup	
			Max	Avg
55	28	14.49	1.048	0.936
44	40	13.86	1.069	0.954
54	31	13.79	1.089	0.972
43	41	13.56	1.109	0.990
56	30	13.47	1.008	0.900
45	39	13.19	1.028	0.918
52	33	12.98	1.170	1.044
47	38	12.36	0.947	0.846
53	32	12.32	1.129	1.008
41	95	11.26	1.190	1.063
48	37	9.93	0.907	0.810
49	36	7.36	0.866	0.774
39	100	7.14	0.846	0.755
50	35	6.31	0.826	0.737
40	98	5.35	0.806	0.719
51	34	4.36	0.785	0.701
Average		10.73	[1.190]	0.883

Fig. 3. Fuel-pin Swelling vs. Burnup for Subassembly B-331; Reactor Grid Position 6-A-3.

nearer the center of the core in which the spread in swelling values and in burnup is much smaller. Much of the wide spread in swelling values can be accounted for by variations in burnup. The results suggest that pin position in the casting array may not be as important a variable as was previously thought.

b. Mark-II Driver-fuel Development. A total of 45 Mark-II encapsulated fuel elements (26- and 24-in. lengths) have been fabricated, and the fuel pins and jackets were sodium-bonded.

Six of the 45 elements are now awaiting both encapsulation in the outer protective capsule tubing and sodium bonding. They have had measurements and tests as follows: lengths, diameters, weights, densities, X-ray for sodium height records, additional eddy-current testing for sodium bonding, and leak testing.

Fourteen of the remainder of the elements are in various stages of the above series of tests, and 25 are awaiting initiation of testing.

Metallography of the encapsulating tubes, which are of Type 304L stainless steel, has been initiated to determine the effect of reactor residence time at temperature, i.e., the effect of times approximately equivalent to 1 and 2% burnup. Exposure will be at 600°C.

7. Vertical Assembler-Dismantler

The construction phase of work on the Vertical Assembler-Dismantler has been completed. Performance tests of all systems as well as remote assembly and operations tests have been completed. A final test and rehearsal of operating and assembly procedures was conducted. Motion pictures of fuel assembly-disassembly operations and of remote installation and maintenance of the machine were made for use in training operators. The equipment, control console, and all accessories are expected to be delivered to the Fuel Cycle Facility in Idaho about June 5, 1967.

Performance tests of the vertical drive units and control system indicate that all performance estimates and specifications have been met

or exceeded. The variable-speed servomotor drive, used for normal operations, provides the low speeds and the short stopping distances required to control the assembly of the hexagon tube properly without exceeding desired force limits or damaging the force-measuring load cell. The high speed permits rapid repositioning of machine elements to bring and keep the work within the viewing and manipulation range. The 2-speed drive, which is to be used as a substitute during maintenance of the servo drive boxes, also has satisfactory response for safe control of forces during assembly operations. It does not, however, provide the wide range of speeds required for optimum control of the machine. Speed and stopping distances for both drives are tabulated below:

	Servomotor Drive		2-speed Drive	
	1 in./min and 30 in./min		3 in./min and 24 in./min	
	at 1 in./min	at 30 in./min	3 in./min	24 in./min
No-load stopping distance (in.)	0.001	0.020	0.005	0.050

Additional tests were performed to determine the effectiveness of the automatic force-limiting feature built into the machine. With a hex tube in the force boom and a lower adaptor in the positional slide, the condition of final hex tube seating was simulated to determine maximum forces produced at various slide approach speeds. The force limit was set at 400 lb (the proposed low force limit). The results are:

	Maximum Force Applied to Subassembly (limit set at 400 lb)	
	Force Boom Driven Down 2-speed Drive	Positional Slide Driven Up Servo Drive
Minimum slide speed (lb)	410	410
Maximum slide speed (lb)	870-930	640-700

8. Quality Monitoring and Control of Sodium Coolant

a. Sampling and Analysis of Primary Sodium. The sampling system has operated satisfactorily. However, direct manual operations are required to obtain samples. Hence, sufficient time must elapse after sampling radioactive sodium for decay to levels such that direct handling is permissible.

The results of analysis are given in Table V.

TABLE V. Primary Sodium Samples--May 1967

Date	Type of Sample and Intended Analysis	Remarks
5/1/67	Extrusion-type sample vessel, for total carbon determination	Sample contained shrinkage voids penetrating sodium mass; hence, considered unsuitable for carbon determination; discarded
5/4/67	Same	Same
5/5/67	10-ml Pyrex beaker for cop- per analysis	Sample was stored pending collection of a group to be sent to ICPP laboratory for anal- ysis; sample was not analyzed for copper, but was used for fission product analyses after EBR-II fission product release
5/8/67	Same	Same
5/11/67	Aluminum tube; carburization potential	Stored for historical purposes
5/17/67	10-ml Pyrex beaker, for cop- per analysis and gross gamma-activity measurement	See note for 5/5/67 sample
5/19/67	Same	Same
5/24/67	10-ml Pyrex beaker, for fission product analysis	
5/25/67	Same	
5/26/67	Aluminum tube	Stored for historical purposes (long-lived fission product activity)
5/27/67	Series of 10-ml Pyrex beaker samples from PI-A and PI-B sample points, for copper analysis	

b. Carbon in Primary Sodium. A sample of primary sodium taken January 4, 1967, was analyzed for carbon with the following results on six aliquots:

Aliquot	Carbon Conc (ppm)	Aliquot	Carbon Conc (ppm)
1	21.8	4	22.4
2	5.5	5	8.6
3	19.6	6	10.2

The sample had been taken in the extruder-type sample vessel by improvisation. No means currently exists for taking "flow-through" samples from the primary system. Although this sample was taken under far-from-optimum conditions, it was analyzed to establish a probable range of total carbon content in the primary sodium. As noted in Table V, two unsuccessful attempts were made this month to obtain primary sodium samples for carbon determinations. The samples were unsuitable for this analysis because shrinkage voids had penetrated the bulk of the solid sodium. Since these samples must be handled in the atmosphere, contamination would certainly have occurred, giving erroneously high values.

Work is in progress to provide for filling the extruder vessels by a "flow-through" freeze-line technique.

The very pure atmosphere attained in the glovebox used for carbon analytical work resulted in sodium slugs so clean that they adhered to the inside of the reaction bottles. The resulting necessary extra handling is expected to have contributed to the scatter noted above. Measures were taken to add moisture to the atmosphere in the glovebox. This reduced the "stickiness" of the sodium slugs and reduced the handling problems.

During the month, the analytical equipment for carbon in sodium suffered some damage when the gas-purification section spontaneously ruptured due to exhaustion and consequent volume change of the BaO absorbent. The section was replaced and blank runs made to establish the effectiveness of the replacement.

c. Primary Sodium Plugging Temperatures. Plugging temperature determinations made during each shift of operation in May gave values that varied from 225 to 250°F.

d. Primary Argon. The values recorded by the continuous gas chromatograph are given in Table VI.

TABLE VI. Primary Argon Analysis by Continuous Gas Chromatograph

	Constituent (ppm v/v)		
	Hydrogen	Oxygen	Nitrogen
High	400 ^a	300 ^b	1100
Low	5	0	500
Average	20	0	600

^aSudden increases in hydrogen concentration have been observed followed by slow decrease. The cause is not known, but fuel-handling operations are suspected. Investigation is proceeding.

^bThe indicated oxygen concentration is believed to be spurious because it occurred during a period of instrument malfunction.

Measurements of primary argon gas activity were made during the month and resulted in maximum readings as follows, up to the time of the major release of fission product gas on May 24:

Ar ⁴¹	$\sim 2 \times 10^{-3} \mu\text{Ci/ml}$
Xe ¹³³	$\sim 2 \times 10^{-3} \mu\text{Ci/ml}$
Xe ¹³⁵	$\sim 6 \times 10^{-3} \mu\text{Ci/ml}$

e. EBR-II Fission Product Release. During the morning of May 24, 1967, routine power operation at 45 MWt was in progress. At 1210 hr an alarm was received in the control room from the charged-wire fission-gas monitor (FGM), and the recorder went to a full-scale reading. It was noted that the FGM signal had begun to increase about 5 min prior to the alarm. Meanwhile the FERD (Fuel Element Rupture Detector) loop monitor (delayed neutron) did not evidence activity in the primary coolant. A cover gas sample taken to confirm the charged-wire (FGM) monitor indicated greater than 10 times the normal (normal is 0-5 mR/hr) reading on a portable instrument. The reactor power was reduced to 1 MW.

At 1226 hr, an increase in the air activity in the reactor building was noted on the control-room (remote monitoring stations) (RMS) recorder. The reactor was shut down. A complete reactor building evacuation by all personnel followed. A subbasement air-activity alarm occurred at 1230 hr.

At 1245 hr entry was made into the reactor building with Scott Air Paks and protective clothing. The following measurements were made:

Main Floor Air Monitor	5 K c/m (normal is 50-150)
Main Floor, Average Level	5 mR/hr
Depressed ^a Area, Maximum Level	250 mR/hr
Argon Vapor Trap	125 mR/hr
Effluent Discharge Monitor	4 K c/m (normal is 1-3 K)
Main Floor Collected High-vol Air Sample	35 mR/hr
Depressed ^a Area Collected High-vol Air Sample	60 mR/hr (normal is <1)
Cover Gas Sample	200 mR/hr (normal is <1)
Spot Contamination of Shoe Covers and Glove Measured	0.5 to 2 mR/hr

^aMezzanine level just below the main floor.

At 1250 hr, the gamma monitor in the Argon Purification Cell alarmed at 20 mR/hr. The packing gland on the storage basket shaft, which had been open during maintenance of equipment, was closed. Operation of the primary pumps was continued after reactor shutdown. During the above

time period the site stack radiation monitor indicated no measurable activity release to the atmosphere and the reactor containment building ventilation was continued as normal. The reactor building was not isolated. Radiation and air particulate activity levels within the reactor building continued to rise until about 1600 hr, and then began to reduce. During all of this time, the delayed-neutron monitors (FERD) continued to operate normally and did not indicate the presence of exposed fuel. The normal background neutron count rate decreased as expected with reactor shutdown. The three channels in this system had been calibrated with a neutron source and found to be working properly the morning of May 24 before the FGM alarm occurred. At 1900 hr the FERD was again calibrated with a neutron source. All channels indicated correctly.

Health Physics survey of six persons who had been in the reactor building at the time of the alarm indicated no contamination on them. As a precaution they were given whole-body counts and, again, no activity above normal was detected. At 1115 hr on May 25, air samples indicated airborne concentration to be below the maximum permissible concentration for long-lived, unidentified, gamma-emitting isotopes. The counting of smear samples indicated that contamination was reduced to background levels. Entry restrictions to the reactor building were thereby removed.

There is no evidence of exposure of primary-system sodium to molten fuel. From the time the charged-wire fission-gas monitor indicated the start of an increase in radioactivity, at about 1205 hr, to the time the reactor was shut down at 1218 hr, the reactor operated at 45 MW. During this time, and later, in addition to no indication of increased signal on the FERD system, there was no evidence of increase in any subassembly outlet temperature, and there was no change in reactivity or power level. The evidence to date indicates a release of accumulated gaseous fission products. All fission products detected in the cover gas samples are either gases or have gaseous precursors. This information, coupled with the knowledge that no FERD annunciation was given, indicates very strongly that significant amounts of fuel material were not exposed to the primary coolant before plant shutdown.

In the first cover gas sample taken at 1218 hr on May 24 and counted shortly afterwards, Cs^{138} and Xe^{138} were identified. Counts of the same samples taken at about 0800 hr on May 25 identified mainly Xe^{133} and Xe^{135} . Counting of a second cover gas sample taken at 1300 hr indicated Xe^{135} activity about 450 times normal, corrected to the time of the alarm.

Samples of primary-tank sodium are routinely analyzed for radioactive isotopes during reactor operation by gamma ray spectroscopy. Results are summarized in Table VII for the times of interest.

TABLE VII. Fission Products and Primary Sodium

Sample Date	Integrated Power on Sample Date (starting 4/19/67) (MWd)	Analysis Date	Isotopes	Cs ¹³⁷ Activity (μCi/g)
4/24/67	10	5/23/67	Na ²²	-
5/8/67	167	5/18/67	Cs ¹³⁷ (Zr ⁹⁵ -Nb ⁹⁵) ^a	5.2 x 10 ^{-3c}
5/19/67	306	5/24/67	Cs ¹³⁷ (Zr ⁹⁵ -Nb ⁹⁵) ^a	3.3 x 10 ^{-3b}
5/24/67 (1615)	544	5/25/67	Cs ¹³⁷ (Zr ⁹⁵ -Nb ⁹⁵) ^a	6.5 x 10 ^{-3b}

^aThe level of Zr-Nb activity was very low and does not appear to have changed between May 19 and May 24.

^bValue being verified.

^cContamination was suspected and a new measurement instituted.

Perhaps due to the scatter and noise levels prevalent in FGM and cover gas sample measurements, it appears that no definite evidence of a fission product leak was given by these sources prior to May 24. FERD did not give any signs of delayed neutrons.

f. Sampling and Analysis of Secondary Sodium. No secondary sodium samples or plugging temperature measurements were taken because the sampling system was secured to permit installation of new equipment, comprising a "flow-through" sampler for filling extrusion sample vessels, two UNC electrochemical oxygen cells, and a plugging loop which is to be proof-tested in the secondary system prior to installation in the FERD system. The secondary cold trap remained in operation.

It is expected that the new sampling equipment will be put into operation early in June.

g. Secondary Argon. The values recorded by the continuous gas chromatograph are given in Table VIII.

TABLE VIII. Secondary Argon Analysis by Continuous Gas Chromatograph

	Constituent (ppm v/o)	
	Hydrogen	Nitrogen
High	12	1600
Low	5	1000
Average	10	1500

h. Expanded Sampling Capabilities for Primary and Secondary Sodium Systems. Work is in progress to restore the distillation apparatus to service in both systems. These had been taken out of service because of

their previous questionable performance. Subsequently, test work performed by the Reactor Engineering Division at ANL-Illinois has given improved understanding of sources of error and effects of operating variables. Design modifications have been made as a result of this work. As feasible, these modifications will be incorporated into the existing equipment at EBR-II.

Work is also in progress to provide "flow-through" freeze-line samples for both systems. These will be designed to accommodate the extrusion sampler or simply lengths of tubing. This will permit improved sampling of sodium for analysis at EBR-II or elsewhere.

i. FERD Plugging Indicator and Freeze-line Sampler. As noted in Sect. f above, work is in progress to proof-test the plugging loop in the secondary system prior to installation in the FERD system. Procurement of equipment for a freeze-line sampler to be installed in FERD has been initiated.

The plugging indicator and sampler will permit surveillance of primary sodium quality during replacement of the primary cold trap.

j. Other Work. Other AEC contractor personnel working on solubility measurements for copper in sodium, and those working on mass transport, deposition, and corrosion processes involving copper in sodium, have been contacted.

Information in the literature pertinent to the compatibility of nickel and high-nickel alloys with sodium, as influenced by temperature, coolant velocity, and coolant oxygen content, is being sought. The Liquid Metals Information Center (LMIC), Canoga Park, California, was contacted for information from their sources. Other AEC contractors were contacted also. This work is related to the proposed study of a nickel reflector in EBR-II.

Other AEC contractors were contacted for assistance in establishing functional requirements for the projected EBR-II Radioactive Sodium Chemistry Loop. Receipt of information by June 15 was requested to permit delineation of function requirements to AEC-RDT by the middle of July.

9. Water Treatment Monitoring and Control

a. Treatment Data. The data for the power cycle streams are presented in Table IX.

TABLE IX. Water Treatment Data for Power Cycle, May 1967

	High	Low	Average
Condensate			
pH	9.7	8.8	9.4
Deaerator Effluent			
pH	9.8	9.2	9.6
ppm N_2H_4	>0.3	0.01	0.02
Dissolved O_2	15	>5	5
Boiler Feedwater			
pH	9.7	8.5	9.6
ppm N_2H_4	>0.3	0.01	0.2
ppb Dissolved O_2	15	>5	5
Boiler Blowdown			
pH	9.3	8.8	9.2
Steam			
pH	9.7	9.2	9.5
Condenser Cooling Water			
pH	8.5	5.9	6.4
ppm CrO_4	20	10	12

b. Treatment Equipment

(i) Acid Injection. The normal pH control system was in-operative during most of the month because the injection line, which runs between the acid pump in the power plant and the dilution trough in the cooling-tower basin, developed a leak in the cooling-tower basin. This line runs through the main cooling-water line between the power plant and the tower (it was installed thus to prevent freezing in cold weather). Because the acid line is submerged under water, repair cannot be effected during operation of the cooling system.

Acid is being added manually to the cooling tower basin from 200-lb drums. Design of an above-ground acid-injection line, with trace-heating and insulation, will be initiated in June. This line will be accessible for repair.

The existing line will be repaired as soon as plant operation permits. After repair, it will continue in service until the above-ground line can be completed.

(ii) Chromate Reduction Equipment. Equipment for reduction of chromate ion (CrO_4^{--}) to chromic ion (Cr^{++}) by sulfur dioxide is being tested on the cooling-water blowdown stream. The purpose of this chromate reduction is to eliminate or reduce the discharge of chromate to the soil.

Proper operating conditions are still being established for the equipment and the data are somewhat erratic. Data for controlled test runs will be reported when available.

10. Experimental Irradiations

a. Status of Experiments in EBR-II. The status of all experimental irradiations in EBR-II at the end of May is shown in Table X.

TABLE X. Status of EBR-II Experimental Irradiations, May 31, 1967

Sub-assembly	Date Loaded	Capsule Content and Number of Capsules()	Experimenter	Approximate Accumulated Exposure (MWd)	Goal Exposure (MWd)
XG02	7/16/65	UO ₂ -20 w/o PuO ₂ (1) Stainless Dummies (18)	GE	10,270	13,600
XG03	7/16/65	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	10,270	19,450
XG04	7/16/65	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	10,270	39,000
XG05	9/3/65	UO ₂ -20 w/o PuO ₂ (9) U-15 w/o Pu-10 w/o Zr (1) U-15 w/o Pu-10 w/o Ti (1) UC-20 w/o PuC (3) Structural (5)	GE ANL ANL ANL GE	9,833	14,750
XA08	12/13/65	UC-20 w/o PuC (8) Structural (11)	ANL	8,071	19,800
XO10	3/24/66	UO ₂ -20 w/o PuO ₂ (4) Structural (11) Structural (4)	GE ANL PNWL	7,201	19,600
XO11	5/9/66	UO ₂ -20 w/o PuO ₂ (7) UO ₂ -20 w/o PuO ₂ (9) 304 SS-30 w/o PuO ₂ (1) 304 SS-20 w/o PuO ₂ (1) 304 SS-20 w/o UO ₂ (1)	ANL GE PNWL PNWL PNWL	5,630	8,300
XO12	8/10/66	UO ₂ -20 w/o PuO ₂ (19)	NUMEC	3,501	20,600
XO15	11/15/66	PuO ₂ -UO ₂ (11)	NUMEC	1,846	11,000
XO16	1/13/67	Structural (9) Structural (10)	ANL GE	526	3,000
XO17	11/15/66	PuO ₂ -UO ₂ (11) (Pu,U)C (3) MK-IA (Metal) (5)	NUMEC UNC ANL	1,846	6,500
XO18	12/6/66	Structural (3) Structural (1) Structural (2) Structural and Heavy Metal Fission Yield Samples (1)	GE PNWL ANL ANL	1,156	21,300
XO19	1/13/67	UO ₂ -20 w/o PuO ₂ (7) (U _{0.8} Pu _{0.2})C (3) Structural (8) Graphite (1)	GE UNC PNWL PNWL	526	7,500

TABLE X. (Contd.)

Sub-assembly	Date Loaded	Capsule Content and Number of Capsules ()		Experimenter	Approximate Accumulated Exposure (MWd)	Goal Exposure (MWd)
XO20	1/13/67	Mixed Oxides	(9)	GE	526	25,000
		Mixed Carbides	(3)	UNC		
		Structural	(4)	PNWL		
		Structural	(2)	ANL		
		Graphite	(1)	PNWL		
XO21	2/27/67	Structural	(7)	PNWL	526	21,500
XO22	2/27/67	Structural	(7)	PNWL	526	8,000

b. Procurement and Inventory Control of Experimental Hardware.

The existing requirement that ANL supply all standard stainless steel components for experimental capsules and subassemblies imposes an obligation to maintain adequate inventories of these components so as to avoid delays in the irradiations programs. Presently, the hardware inventories are maintained at ANL-Illinois. However, in view of the fact that all coordination of activities in irradiations programs is effected in Idaho, it appears that the functions of hardware procurement, inspection, inventory maintenance, and distribution to experimenters should be carried out in Idaho, in the interest of efficiency and close control. Planning was initiated in May to make this transfer of responsibility. The procurement, inspection, and inventory control of experimental subassembly hardware proper will be handled by the Fuel Cycle Facility, since analogous functions are in existence for the driver fuel subassemblies. This FCF activity will be guided by the demand forecasts.

11. Nondestructive Testing

a. Antimony-Beryllium Neutron Source. A high-intensity Sb-Be neutron source is to be constructed and installed in FCF. The source will be used for neutron-radiographic examination of irradiated experimental fuel specimens without having to remove them from the FCF and transport them to a reactor source of neutrons. The capability of performing neutron-radiographic inspections coupled with the ability to reinsert experimental capsules into the reactor will greatly extend the usefulness of EBR-II as an irradiation facility.

A computer study to optimize thermal-neutron intensity from an Sb-Be source was conducted during FY 1967. The study was completed in March 1967, and a draft of a report covering the Sb-Be source investigation was completed in May 1967.

The design of the Sb-Be source is complete, and procurement of materials is underway. The beryllium has been shipped from the manufacturer and is expected in Idaho by June 2, 1967. Materials for the steel

source container have been procured, and the container is presently being assembled. Completion of the source container is scheduled for June 9, 1967.

Preliminary design of the hardware required for neutron radiography in FCF has been completed, and checkprints have been made. This includes a foil-handling device, fixtures for holding experimental capsules, and equipment for positioning both the capsules and the neutron source. The electrical diagram has not yet been completed. Construction of the equipment will be started in June. The projected date for completion of the installation in FCF is October 1, 1967.

Five dysprosium foils to be used for neutron image detection have been ordered. The expected delivery date is June 23, 1967.

The antimony gamma source will be activated in EBR-II. This will require the use of stainless steel-clad antimony tetroxide sources rather than tantalum-clad antimony metal sources, so that the sources can be washed and reused. There is presently one Sb_2O_4 source available at EBR-II, but more than one source will be required.

Components for four additional Sb_2O_4 sources have been fabricated. Also, 5 lb of Sb_2O_4 were ordered. The receiving inspection revealed that the material was a pale yellow powder rather than a white powder, and the density was considerably below what Sb_2O_4 should exhibit. Apparently, antimony pentoxide (Sb_2O_5) had been substituted for Sb_2O_4 . Since Sb_2O_5 would lose oxygen at the operating temperature of the source during activation in the reactor, the material was returned to the supplier. Delivery of a replacement order is expected in June 1967.

b. Neutron-sensitive Image Intensifier. A neutron-sensitive image intensifier tube developed by ANL and the Rauland Corporation has been ordered. This tube has good neutron sensitivity, but its resolution capability is below that of the foil-converter technique of image detection. Despite limited capability of resolution, the tube will serve a useful purpose for rapid screening of inspection objects. Suspects can then be examined in more detail with the foil technique.

The tube is also sensitive to gamma rays. This background can be overpowered with the neutrons if the neutron intensity is high. Irradiated fuel can be inspected with a reactor source of neutrons, but it is questionable whether an Sb-Be source has sufficient neutron intensity.

It is unlikely that the direct imaging technique, using the image-intensifier tube, can be adapted for use with the FCF neutron source. However, it can be used in the projected neutron-radiography facility at TREAT.

Although the vendor has not given a delivery date, the tube is expected after August 1967.

c. Neutron-sensitive Vidicon Development. An instantaneous imaging device which works on the vidicon principle should have high resolution capabilities and good gamma-ray discrimination. Pickup tubes could probably be made available at a low cost. However, the vidicon appears to have low sensitivity, and special slow-scan techniques may be necessary.

Several materials have been tested which showed a change in electrical conductivity under neutron bombardment. However, no material tested has sufficient sensitivity to operate a vidicon tube at the normal TV scan rate.

Boron nitride was substituted for normal photoconductive target in a vidicon pickup tube, and a camera has been modified to control the integration time between target beam scans. This will allow accumulation of information for times which are orders of magnitude longer than the normal 1/30 sec, and the results of each readout will be recorded on film. This modified vidicon tube will be tested in June 1967.

A recording storage tube (Raytheon 7571/QK685) with a neutron-sensitive target material holds promise as an instantaneous imaging device. This possible application was discussed with Raytheon Corporation in May. They have agreed to furnish a tube for testing in June 1967.

d. Development of Isotopic Neutron Source. A computer study of isotopic neutron sources is being made to find a radiographic source with a higher neutron yield and lower gamma-ray background than the Sb-Be system. The sources presently under study are Am-Cm-Be and spontaneous-fissioning Cf²⁵². The parameters being investigated are moderator composition and moderator thickness. Computer problems for ordinary water, carbon, and heavy water have been run. Other moderators being considered are beryllium and zirconium hydride.

Future work will include an investigation of the sources mentioned alone in a multiplying subcritical assembly. Computational work on the subcritical assembly problem will probably begin in July 1967.

e. Other Techniques of Nondestructive Testing.

(i) X Ray. A 250-kV X-ray machine was procured from government surplus. The X-ray tube, high-voltage cables, and some minor items were replaced to put the unit into operation. The X-ray machine will be used primarily for preirradiation inspection of experimental capsules, but other inspections, e.g., pipe welds, can be made.

The X-ray unit is presently located in an open pit in the Inspection and Test Facility (ITF). This requires special administrative controls while the machine is in operation, and as activity increases in the ITF this location will no longer be acceptable. Construction (GPP) funds were made available in FY 1967 for the addition of a shielded radiography room to ITF. This room will be completed in the first half of FY 1968.

Five zirconium oxide crucibles of the type used in melt refining of fuel in the FCF were radiographed. Several cracks as well as density striations were identified on the radiographs. FCF personnel are currently evaluating the results to determine whether or not radiography can be used for reliable quality control of the crucibles.

Several samples containing temperature monitors being considered for use in EBR-II were radiographed.

f. Eddy-current Inspection Technique. The capability to perform eddy-current inspections is being developed. Inspection of tubing for experimental capsules, inspection of the sodium bond in sodium-bonded experimental capsules, and possibly inspection of evaporator tubes in the EBR-II steam generator are inspection problems to which the eddy-current technique is applicable.

A pulsed eddy-current system is being procured for the inspection of experimental capsules. All components except the oscillator and the conveyor have been received. The oscillator is 20% complete and will be furnished by July 1, 1967.

The conveyor will be essentially the same as the one used by N. Beyer at ANL-Illinois. Drawings have been obtained and bids will be solicited in early June. Expected completion date of the conveyor is in early August 1967.

Three samples of evaporator tubing have been prepared with various sizes of defects. These samples will be used to evaluate testing methods and equipment being considered for evaporator tube inspection.

12. Handling and Examination

a. Tantalum-clad Antimony Source Rods. Two tantalum-clad antimony source rods were removed from the reactor for examination. The approximate accumulated nvt on each source was:

SO-1911 (Row 13):	2.6×10^{20} nvt
SO-1912 (Row 7).	5.3×10^{22} nvt

The source rods were visually inspected in the FCF Air Cell. The tantalum surfaces appeared bright and shiny where there was no sodium oxide. Since the sources were not washed, a considerable portion of the surface was covered with oxide and could not be inspected.

The outside diameters of these two sources were measured by means of a micrometer in the Air Cell. The measurements were taken by setting the micrometer gap smaller than the source and then opening the micrometer until it would pass over the source. Two measurements, 90° apart, were made at 5 points approximately 3 in. apart along the length of each source. An average of the two measurements at each point is given in Table XI.

TABLE XI. Diameters of Antimony Source Rods

<u>Source SO-1912</u>					
Distance from Top of Source (in.)	1	4	7	10	13
Average Diameter (in.)	0.916	0.915	0.915	0.915	0.913
<u>Source SO-1911</u>					
Distance from Top of Source (in.)	1	4	7	10	13
Average Diameter (in.)	0.914	0.916	0.916	0.915	0.913
Diameters of Antimony Slug from SO-1912 (in.)				0.788	0.789
Diameters of Antimony Slug from SO-1911 (in.)				0.780	0.778

After measurement of each source, a small section of cladding was removed for hardness testing and wall thickness measurements. The diameters of the antimony slugs from these sections were measured by the same method as used for the source OD measurements. Two measurements 90° apart were taken on each sample and are also reported in Table XI. The cladding OD, antimony slug OD, and cladding thickness were all within the tolerance of the construction drawing dimensions.

A cross section of the tantalum cladding was metallurgically prepared and examined. There was essentially no change in the microstructure from the center of the wall to either the inside or the outside surface. There was no evidence of corrosion or erosion.

Source Rod SO-1912 was supported at each end and struck in the middle as hard as possible with manipulators. There was elastic deformation, but the source did not fracture.

Hardness measurements were made of a tantalum sample from SO-1912. The readings were 277 ± 5 on a diamond pyramid hardness test with a 100-g load (DPH-100).

The hardness of an unirradiated sample is to be determined in the near future for comparative purposes.

13. Fuel Cycle Facility (FCF)

a. Operations

(i) Hot-line Operation. Normal production activities were carried out throughout the month. Three alloy-preparation runs (total of 31 kg) were made with unirradiated feed stocks to supplement the irradiated fuel inventory. Process operations are summarized in Table XII.

TABLE XII. Production Summary

	<u>5/1/67 through 5/31/67</u>	<u>Total This Year</u>
1. Subassemblies received:		
Core Control, Safety (for processing)	0	39
Other	15	108
2. Subassemblies dismantled (for processing)	3	43
3. Subassemblies dismantled (for exami- nation, etc.); includes 2 refabricated subassemblies with bad hardware	6	51
4. Subassemblies fabricated (includes those made with unirradiated fuel)	14	74
5. Subassemblies transferred to reactor	2	52
6. Elements decanned:		
From irradiated subassemblies	714	
Rejects	85	
Other	-	
Total Decanned	<u>799</u>	

Melt Refining

	<u>Irradiated Fuel</u>	<u>Recycle Material</u>	<u>New Fuel</u>
7. No. of runs	5	10	3
8. Average pour yield, %	92.4	94.0	97.0

TABLE XII. (Contd.)

	<u>Processing</u>	<u>5/1/67 through 5/31/67</u>	<u>Total This Year</u>
9. Injection-casting runs (total number)		21	
10. Elements processed:			
Accepted		1729	
Rejects		263	
11. Elements welded:		1308	
Rewelded:		0	
12. Elements leak tested:			
Accepted		1314	
Rejected		34	
13. Elements bonded (including recycle)		1440	
14. Elements bond tested:			
Accepted		1164	
Rejected		52	
15. Elements to surveillance		42 from 2 subassemblies	
<u>Waste Shipments</u>			
16. Cans to burial ground		14	
17. Oxide and glass scrap to ICPP		2	

The materials test subassembly designed as "Surveillance I" was received and dismantled (see Sect. II.A.5.b). Special handling procedures, including an alcohol wash step to remove the sodium, were utilized for this subassembly.

Some cracking has been observed recently in melt-refining crucibles prior to their use. Of the original batch of 250 crucibles (made of zirconium oxide), all were used successfully without breaking. Eight out of approximately 50 of a recent order of 150 crucibles have broken, primarily from handling prior to the melting operation. The newer crucibles are, on the average, about 5% lighter than the original crucibles, and consideration is being given to specifying a minimum weight.

With the procurement of EBR-II Driver Fuel Elements from either the cold line or from commercial vendors, the inventory of driver-fuel elements is expected to increase beyond that which can be stored or processed in the FCF. Excess irradiated driver-fuel elements will be reprocessed at the Idaho Chemical Processing Plant (ICPP). Until sufficient irradiated driver fuel has been accumulated for a reprocessing run, ICPP plans to store the fuel in a water basin.

Because its design is relatively simple and it can be quickly fabricated, a top-loading and -discharging cask is being designed and built. It will handle one 15-day-cooled subassembly and provide natural circulating water for cooling the contained subassembly. Detailed design of the top loading-top unloading cask is complete. After final checking, the drawings will be submitted for fabrication of the cask.

Six irradiated fuel elements were sent to the ICPP for in-house process and corrosion studies. Five of the elements are being utilized for the corrosion studies of the effect of underwater storage on the cladding. The remaining one is being used for comparison and process studies.

To keep a continuing flow of subassemblies from the Fuel Cycle Facility to the reactor, a steady supply of stainless steel hardware is made available. The present procurement for such hardware is programmed to supply 45,000 fuel-element jackets and parts for 345 subassemblies (200 core, 85 sixth-row core, 50 control rods, and 10 safety rods).

The fuel-element jackets are assembled in Central Shops (Illinois) from purchased components. During the past few months, this fabrication has been hampered due to problems in the manufacture of fuel-element tips. During 1966, the company that has cast these tips successfully on previous occasions was unable to provide acceptable parts. To alleviate the casting problems, the design was changed slightly to allow manufacture from wrought stock. Two vendors have since been making the tips. Bids are being obtained for 25,000 tips to complete the required jackets. The total jacket production for May was about 3000.

Hardware for the control and safety rods was manufactured and assembled by a vendor. These preassemblies, upon addition of the proper code number, are ready to be loaded with fuel elements. All lower preassemblies are being held in Illinois pending a possible change in bearing material.

Components for the core and sixth-row core are being purchased from vendors or being manufactured in Central Shops, and are being assembled in Idaho as needed by FCF. Availability of the parts is summarized in Table XIII.

An order of 6000 Mark-IA restrainers was received this month. The first welds made with these restrainers were unacceptable, and metallographic inspection indicated the presence of slag inclusions in the metal. An order to replace these parts has been initiated.

TABLE XIII. Hardware for EBR-II Core and Sixth-row Core Subassemblies

Part	Quantity Ordered	Quantity Received	Comments
Core lower adapters	240 ^a	123	Balance rejected and returned to vendor
Sixth-row core lower adapters	85	37	Same as above
Hex tubes	500 ^a	487	Complete
Top end fixtures	485 ^a	483	Complete
Shields	600	154	
Flow distributors	300	315	Complete
Grids	300	198	

^aSome extra parts were ordered for irradiation subassemblies.

The first batch of Mark-IB restrainers was received in Idaho. Inspection of the precision weld used to fasten the 0.040-in.-dia wire to the plug indicated that it was not satisfactory. The weld procedure is being revised, and a tensile test of the welds will be added to the regular inspection of these parts.

(ii) Cold-line Operation. A cold or unirradiated fuel-fabrication line is being installed outside the Air and Argon Cells of the FCF. This line will be used to augment the fuel-production capacity of the Fuel Cycle Facility and to serve as backup to it. It is expected to be in operation during the first half of fiscal year 1968, with a production rate of about eight subassemblies per month. With initiation of the cold fuel production, transfer of spent subassemblies in excess of those necessary for remote production will be made to the Idaho Chemical Processing Plant (ICPP) for storage and processing.

The following is the status of the various components that will be used in the cold line for production of unirradiated fuel.

(α) Alloy Preparation Furnace. Installation was completed in April, but a problem developed when the power line to the un-cooled induction coil overheated. The lines have been changed to water-cooled copper tubing. The console for this unit has been completed and checked.

(β) Injection-casting Furnace. The vacuum-pressure tank was completed and tested by Central Shops, and has been shipped and received in Idaho. The internal components have been completed, and installation into the tank has begun. Almost all valves and other accessories have been received. Construction of consoles and power supply has been started.

(γ) Pin-process Equipment. Length-cutting equipment is 75% complete. Measuring stations are complete and are waiting to be checked out with the electronic equipment which is over 90% complete.

(δ) Glovebox Equipment for Element Assembly. The glovebox has been moved adjacent to existing sodium-handling facilities, and windows, gloves, and exterior equipment are being installed by a subcontractor. The sodium extruder, settling furnace, and restrainer welder have arrived in Idaho and are ready to be installed in the glovebox. The purification system is complete and ready to be installed. Miscellaneous small items for sodium handling are being fabricated.

(ε) Fuel-element-processing Equipment. The leak detector and valve cabinet are over 95% complete. The bonding furnace is ready for operation, and the bonding magazines have been delivered to Idaho. The bond tester has been received in Idaho. Consoles for the bonder and bond tester are over 95% complete.

(ζ) Assembly Machine. Remote Control Division has completed the design of the assembly machine and detailing is over 90% complete. The frame has been started in Central Shops and the drive motor ordered.

(η) Installation. A new addition to the Inspection Test Facility (ITF) for the degreaser is well under way. The degreaser has been moved from the FCF and modifications have been started.

(iii) Maintenance and Repair. Routine maintenance and repair is required to keep the existing equipment in the Air and Argon Cells in operating condition.

During the past month, a manipulator carriage failed because of a sheared pin in the carriage-drive linkage. The carriage was removed to the wind-and-weather enclosure and repaired. A universal joint pin in the bridge-drive linkage of another manipulator also had to be replaced during the past month. The pin was broken; however, it was replaced with a new pin before it had worked completely out of the joint. This repair was done remotely in the Argon Cell.

(iv) Operator Training. The form of training utilized by FCF operations is a combination of on-the-job training augmented by a series of lectures. These lectures include safety, accountability, criticality control, and general operating procedures in addition to specifics for cell processes and equipment. The lecture series of the training program for 21 new employees was completed last month. This consisted of a total of 42 individual lectures.

A proposal for an operator-qualification program has been submitted and is being evaluated. Under this program, operators will be tested and qualified for specific equipment operation by the responsible chief technician and by the responsible engineer. Periodic requalifications would be required. Only currently qualified personnel would be allowed to operate the equipment.

(v) Operating Manuals and Reports. As a part of the overall operating and training program, manuals are being prepared on all phases of the operation. In general, emphasis or priority is being given to those sections of operations which are related to personnel safety, cause damage to the facility, and affect only individual components of equipment in this order.

In the past month, 14 sections of the operating manual have been completed and submitted for review and acceptance.

(vi) Product Analysis. Two basic types of analytical effort are included in this area. The first is the chemical analysis required for quality assurance of all reactor fuel. This includes analysis for specific trace elements which build into the fuel during recycle, analysis of cell-atmosphere samples, equipment contamination samples, etc. The second effort is that of postirradiation analysis of the fuel used in the reactor. This is a continuing effort to correlate process and product variations with postirradiation fuel behavior.

(α) Chemical Analysis. Five pins from each of 3 surveillance subassemblies were gamma-scanned with the 256-channel analyzer to determine the radial and axial burnup profile. The numbers of fuel production and surveillance analyses accomplished are tabulated below. No significant deviations from the normal were observed.

<u>Analyzed for</u>	<u>Number of Surveillance Subassemblies</u>	<u>Number of Production Subassemblies</u>
U (total and isotopic)	-	31
Pu (total and isotopic)	-	18
Si	2	23
Al	-	26
Fe	2	33
Mo	2	21
Ru	2	20
Tc	1	
Cu	-	1
Zr	-	1
Rh	-	2
Total	9	176

(β) Postirradiation Analysis of EBR-II Fuel. A continuing program to evaluate the irradiation performance of Fuel Cycle Facility-produced, EBR-II driver-fuel alloy consists of determining the irradiation swelling that has occurred in U-5 w/o Fs alloy of EBR-II driver-fuel elements as a result of service in the reactor.

The program consists of postirradiation sampling and testing of elements manufactured from each casting batch of fuel. Normally, 3 to 10 elements from any one fuel batch are selected for testing. Fuel swelling for elements is determined in the FCF Air Cell by utilizing the fuel-element bond-testing machine. Since the sodium levels for all FCF-produced, driver-fuel elements are determined in the same machine prior to insertion in the EBR-II core, the postirradiation measurement of sodium levels in a fuel element readily allows the determination of the change in sodium level resulting from reactor service. This displaced sodium is a direct measure of the total volume swelling of the U-5 w/o Fs fuel pin contained within the fuel-element-jacket tubing, providing that jacket yielding has not occurred. The displaced sodium volume can then be related to the volume of the as-cast fuel pin to determine a percentage volume of swelling for the fuel.

Each of the 46 batches of reprocessed Mark-I fuel ingots produced in the FCF was evaluated. A total of 245 elements from 19 different irradiated subassemblies was investigated. A total of 62 reprocessed Mark-IA fuel ingots (the type currently being fabricated in the FCF) has been evaluated to date. From these batches, 613 elements from 42 different subassemblies were subjected to testing.

Reprocessed EBR-II fuel as fabricated in the FCF and as represented in this investigation consisted of two types, namely, "MR" and "MRC." The U-5 w/o Fs alloy ingots termed "MR" normally utilized varying proportions of freshly irradiated fuel pins, fuel pins from refabricated fuel elements rejected for failure to meet specifications (leak-testing and bond-testing rejects), and re-enrichment uranium for the melt-refining charge. The refined alloy ingot from this operation was then used for injection casting of new fuel pins. Casting heels from previous injection-casting batches are normally added to the furnace charge to bring it to the desired weight level.

Alloy batches designated "MRC" utilized varying proportions of as-cast fuel pins rejected at the pin-processing station, cropped ends of pins accumulated at this same station, and injection-casting heels derived from previous casting operations as the charge material for consolidation into a refined alloy ingot at the melt refining furnace. No unprocessed irradiated fuel was utilized in these melt-refining charges.

Figures 4 through 6 present the results obtained for FCF-reprocessed Mark-I fuel elements, and Figs. 7 through 9 illustrate data obtained for reprocessed Mark-IA fuel. Figure 10 indicates information obtained for nonirradiated (virgin alloy) Mark-IA fuel pins which had been fabricated by the ANL Metallurgy Division in Building 350 at the Illinois Site. These pins were assembled into fuel elements in the FCF. Figure 11 presents a comparison of results obtained for the three types ("MR" and "MRC" reprocessed alloy, and "SL" virgin alloy) of Mark-IA fuel alloy investigated.

Fig. 4. EBR-II, FCF-reprocessed, Mark-I Fuel, "MR" Type: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 5. EBR-II, FCF-reprocessed, Mark-I Fuel, "MRC" Type: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 6. EBR-II, FCF-reprocessed, Mark-I Fuel: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 7. EBR-II, FCF-reprocessed, Mark-IA Fuel, "MR" Type: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 8. EBR-II, FCF-reprocessed, Mark-IA Fuel, "MRC" Type: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 9. EBR-II, FCF-reprocessed, Mark-IA Fuel: Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 10. EBR-II, Mark-IA Fuel, "SL" Type (ANL-Chicago): Increase in Fuel Volume vs. Fuel-element Burnup.

Fig. 11. EBR-II, Mark-IA Fuel: Increase in Fuel Volume vs. Fuel-element Burnup.

From Figs. 6 and 9, it can be observed that irradiation swelling of the "MRC"-type U-5 w/o Fs alloy was significantly less than that indicated by the "MR" alloy type in the burnup range investigated. This indication held true for both Mark-I (48% nominal enrichment) and Mark-IA (52% nominal enrichment) fuel pins. It can also be observed that swelling-rate differences existed between the Mark-I and the Mark-IA fuel. Volume swelling of Mark-I, type "MRC" fuel alloy was approximately 5% at a maximum burnup level of 1.2 a/o, whereas swelling of this same "MRC" type in Mark-IA fuel was in the region of 7%. At this same burnup level, Mark-I "MR" fuel indicated swelling of approximately 10%, whereas Mark-IA "MR" alloy increased in volume about 12%.

For comparison, Fig. 10 indicates that the previously nonirradiated Mark-IA "SL" fuel (ANL-Lemont fabricated) swelled approximately $14\frac{1}{2}\%$ at this same burnup level.

Currently, considerable effort is being expended at both the ANL Idaho and Illinois sites in an effort to resolve these swelling differences. Experiments are underway to evaluate the effects of fuel-enrichment differences, variations in fuel-element design, trace-element-composition differences in the fuel, irradiation temperature, thermal cycling, casting variations, etc., upon the irradiation swelling characteristics of U-5 w/o Fs fuel.

b. FCF Equipment Improvements

(i) Processing Equipment. The assembly and disassembly operations of the Vertical Assembler-Dismantler (VAD) (see Sect. II.A.7) were demonstrated during the week of May 1. Motion pictures were taken of the operations, and the equipment was crated and shipped to Idaho.

A sleeved, 2-ft-dia hole has been installed in the floor of the mockup room of the Fuel Cycle Facility. The VAD machine will be installed over this hole for operational checkouts and training prior to installation in the Air Cell.

A pneumatic transfer tube is being designed to be installed between the Air Cell and the analytical caves to facilitate the transfer of samples between the Fuel Cycle Facility and the Analytical Complex. Drawings have been completed and are now being checked. Purchase orders for procurement of fabricated items have been initiated. The project is currently on a schedule which calls for completion by the second quarter of FY 1968.

A nondestructive method of evaluating irradiation damage to experiments and driver fuel is neutron radiography. Equipment to provide for this is being designed to be installed in the Argon Cell of the Fuel Cycle Facility; essential design has been completed and detailed drafting is proceeding. Design of specimen-positioning and foil-positioning devices has been completed. Items of design remaining to be completed are specimen holders, a source positioner, and the electrical diagram. Based on present evaluation, this project should be completed during FY 1968.

(ii) Auxiliary Equipment. One of the major problems with the maintenance and/or repair of the Argon Cell gas-recirculating fans is the level of contamination within the fan housing. One means of reducing this level of contamination would be to filter the gas in the Argon Cell prior to its entrance into the fan boxes.

A test filter system has been set up in the Argon Cell, using small test areas of various filter media proposed for the cell filters. The test system has been placed in the cell to obtain data for the pressure drop across the media versus time and flow. These data will be used to determine the fan blade setting and the filter-replacement schedule.

Present operations of the Fuel Cycle Facility require that all material that is transferred into or out of the Argon Cell must traverse the entire length of the Air Cell in passage. To alleviate this extensive and unnecessary material handling, a modification is being made in the transfer cell between the Air and the Argon Cells to allow unirradiated new process materials to be introduced directly into the transfer cell and from there into the Argon Cell. This transfer-cell modification can also be used to transfer samples or low-level-contaminated material from the Argon Cell. The modification will include a transfer lock, a viewing window, and a pair of master-slave arms inserted into the wall of the transfer cell in the basement of the Fuel Cycle Facility.

The frame for the window-transfer port combination has been installed in the transfer-cell wall. Through-tubes have been installed in the transfer-cell wall for the Model D and F master-slave units. A plug to match the through-wall frame was obtained from the vendor with all of the mechanism installed for moving the plug in operating the transfer port. After some minor modifications, the original mockup shielding window was installed in this plug. The only remaining job is to install the plug and remove the temporary shield from the transfer cell. This project is 90% complete.

As a means of decreasing the interdependency of the schedules of the Fuel Cycle Facility, outside reprocessing plant, and the Reactor, more storage capacity for subassemblies is desirable. A major interruption in the flow of subassemblies would result in a stoppage of these operations. An interim-storage facility providing for a one- or two-month holdup capacity would insure that creditable interruptions could be compensated for.

The feasibility of utilizing the existing storage pits in the passageway between the reactor building and the Fuel Cycle Facility was examined. Evaluation of these existing pits indicated that the number of subassemblies that could be stored would be insufficient and that the shielding problems to be overcome would be large. Therefore, a study of the feasibility of designing new storage facilities has been undertaken. Functionally, the new facility would provide storage of up to 40 complete subassemblies either from the Fuel Cycle Facility or from the Reactor with provision for shielding and cooling.

Transport of the subassemblies to and from the new storage facility would be in a shielded cask. Design concepts of this bottom-loading and bottom-discharging type of cask are being reviewed. This cask could also be used for transfer of subassemblies from the Fuel Cycle Facility to proposed reprocessor, ICPP.

(iii) Repair and Decontamination. A facility for the decontamination and repair of the electromechanical manipulators is needed. This facility could be most advantageously located in the wind-and-weather enclosure on the roof of the Air Cell above the existing shielded plug opening.

Preliminary concepts have been proposed and approved. The ventilation rates and flow patterns have been determined, and final design is being completed. Detailed crane drawings were inspected and approved for placement of an order. The entire job is estimated to be 5% complete at this time.

c. Hot Fuel Examination Facility (HFEF). Design studies in progress include an integrated complex of shielded examination cells and nonshielded support facilities. The examination cells will be used for disassembly and assembly operations as well as the nondestructive and destructive examinations of reactor fuels, specimens, and materials. The examination cells will be equipped to carry out a wide variety of examinations with remote-handling capabilities necessary to adapt readily to new techniques as they are developed. These cells will contain an inert gas atmosphere and incorporate features that insure essentially complete containment of particulate radioactive materials with a very high degree of reliability. The complex will also include shielded areas for the storage, decontamination, and maintenance of in-cell equipment.

The general technical requirements of the facility have been written and reviewed by the principal users within the Laboratory. Work is progressing on the preparation of descriptions and design, conceptual configurations of the cells, manipulators, buildings, and conceptual studies of the support systems.

B. Physics Development

1. ZPR-3

Work continued this month with experiments in Assembly 49, the second in a series of assemblies in an experimental program designed to provide a set of integral data for use in testing calculational methods and nuclear cross-section sets which can be used in the design of large plutonium-fueled fast breeder reactors. The composition of Assembly 49 is the same as the first assembly in the series, Assembly 48, except for the elimination of the sodium. Thus, the core of Assembly 49 has a composition made up of fuel with a U^{238}/Pu^{239} ratio of 4.5, graphite, stainless steel, and void. A comparison of data obtained from the two assemblies will give important information on the effect of sodium on various nuclear parameters.

In the experimental program this month, fission ratios at the core center were investigated in order to determine the effect of the loss of sodium on the central neutron flux and spectrum. Both depleted and enriched uranium foils have been irradiated between plates of the various core materials in the front end of a central core drawer to determine the fine flux distribution within the core-drawer loading. Reactivity measurements have been made at the core center for various fissile, fertile, structural, and poison-material samples. These samples have been in the form of small cylindrical and annular samples, and large (2-in. cube) samples. These perturbation measurements can then be compared with those made with Assembly 48 in a sodium environment. The data obtained from Assembly 49 are being analyzed and will be reported upon completion of the analysis.

2. ZPR-6

a. The Effect of Varying σ_p of U^{238} on Fission and Capture Rates. The series of experiments designed to vary the potential scattering cross sections of U^{238} have been completed. Most of the data associated with this series of experiments have been reported (see Progress Report for April 1967, ANL-7329). The comparison of the fission rates in U^{235} and U^{238} and the capture rates in U^{238} between the two cases in which the U^{238} plates were bunched (lower σ_p) and unbunched is reported at this time. The experimental details about the size of the foils, the exact positions of the foils in the three-drawer cell, and the reduction of the data were described in ANL-7329.

Figure 12 shows the fission rates in U^{235} across the three-drawer cell for the bunched and unbunched U^{238} plates. Also shown in Fig. 12 are the loading patterns for the two cases. The sets of data have been normalized to the average fission rates of foils 17 and 28. The assumption made in this normalization is that the neutron spectrum between the two 1/2-in. sodium cans is not affected by the bunching. Regardless of

the validity of this assumption, the relative distribution within each set of data is maintained. It is noted that in the bunched case the low-energy fluxes are augmented in the graphite and depleted in the U^{238} . The distributions of the fission rates in the unbunched case does not exhibit such peaks and valleys.

Fig. 12. U^{235} Fission Rates in Bunched and Unbunched Loading Patterns in ZPR-6 Assembly 5

Figure 13 shows the fission rates in U^{238} . The same normalization factor obtained from the data in Fig. 12 was used to normalize these data. The difference between the two sets of data is not significant. It is noted that the fission rates of foils 17 and 28 are within 5% of each other, indicating that the assumption of the constancy of the neutron spectrum at these points is perhaps reasonable.

Fig. 13. U^{238} Fission Rates in Bunched and Unbunched Loading Patterns in ZPR-6 Assembly 5

Figure 14 shows the capture rates in U^{238} for the bunched and unbunched cases. The same normalization factor used earlier was used for this data. The two prominent features of these data are: (1) the decrease in capture within the bunched U^{238} , and (2) the decrease in capture in foils 7 and 31 in the unbunched case. Foils 7 and 31 were placed between 1/16-in. and 1/8-in. U^{238} plates. This indicates that self-shielding within these plates is significant. The relative statistical errors in all the data is of the order of 0.2%.

Fig. 14. U^{238} Capture Rates in Bunched and Unbunched Loading Patterns in ZPR-6 Assembly 5

b. Streaming Effect on the Sodium-void Coefficient. The streaming effect on the sodium-void coefficient was investigated in the following manner:

(i) Three drawers in each half of the matrix, loaded in the unbunched pattern of the central drawer of the three-drawer cell (middle-drawer front view as shown in Figs. 12-14), were used in the measurement. These drawers were located at the center of the core, and their positions in the matrix were 24-23, 23-23, and 22-23 in both halves of the reactor.

(ii) The reactivity effect on the system due to the rotation through 180° of the materials in these drawers was measured. It is pointed out that the relative positions of the U^{235} plates are not affected by the rotation.

(iii) The sodium-void coefficients were measured for the front 8 in. and the entire 28-in. columns in the three drawers in each reactor half with and without rotating the material 180°.

(iv) The sodium-void coefficients were also measured with the 2nd, 4th, and 6th sections (each section being 4 in. long) of the materials in each drawer rotated 180°. Figure 15 shows this loading pattern. The objective of this loading pattern was to approximate half the width of the sodium channels which, because of their low density, may be seen as "windows" by the neutrons. The results are shown in Table XIV. It is noted that sodium-void coefficients when the 28-in. columns of sodium are removed from the three drawers in each half are the same with or without the 180° rotation of materials. With partial rotation (rotation of sections 3, 4, and 6--see Fig. 15), the sodium-void coefficient for the same amount of sodium decreased by 0.560 ± 0.116 lh or 13%. The coefficients of the front 8-in. sections were rather small and their associated errors were relatively large, so that no conclusions based on this data can be made about the streaming effect.

Fig. 15. Top View of Drawer Showing the Even-numbered Sections Which Can Be Rotated 180° and the Relative Position of the Sodium

TABLE XIV. Streaming Effect on Sodium-void Coefficients

Section Voided, (k Na)	Position of Material in Drawers	Reactivity Effect Caused by Sodium Voiding (lh)
Front 8 in. (0.326)	Original Position	+0.507 ± 0.136
(0.326)	Rotated 180° throughout drawer	+0.415 ± 0.312
(0.326)	180° Rotation of Sections 2, 4, and 6 (see Fig. 15) in drawer	+0.511 ± 0.116
Entire Column (2.270) (28 in.)	Original Position	-4.385 ± 0.136
(2.270)	Rotated 180° throughout drawer	-4.232 ± 0.100
(2.270)	180° Rotation of Sections 2, 4, and 6 (see Fig. 15) in drawer	-3.749 ± 0.137

The relative effect on the system due to the complete rotation of 180° of the three drawers per half was -0.384 ± 0.055 Ih. The same effect with partial rotation was 0.129 ± 0.100 Ih.

3. ZPR-9

The program of measurements in Assembly 17 (see Progress Report for April 1967, ANL-7329), the final in a series of polyethylene-moderated assemblies specifically designed to investigate the U^{235} Doppler effect, has been completed. The results indicate a substantial negative U^{235} Doppler coefficient in this spectra. Various spectral checks using, for example, cadmium ratios of dysprosium and manganese, fission ratios of Pu^{239} , U^{238} and Np^{237} with respect to U^{235} , and central danger coefficient measurements confirm that the spectrum in this assembly is considerably softer than normal fast reactor spectra.

Final analyses of the experimental results for Assemblies 13 through 17 are underway.

ZPR-9 is currently shutdown to allow the necessary modifications to be made to the Doppler equipment for plutonium and U^{233} Doppler measurements. Assembly 17 is being removed and replaced with Assembly 12 for a detailed series of Doppler measurements evaluating heterogeneity and environmental effects.

4. ZPPR

At month's end the construction contractor was placing earthfill around the reactor cell and workroom-vault area. All major concrete pours except the ring beam, AFSR floor, and reactor biological shielding have been placed. Exterior and interior blocks were being placed in the support wing. Air-handling equipment was being installed in the fan loft of the support wing. Electrical connections were being made to the installed equipment. Ductwork for ventilation was being placed.

The roof filter sand was in Idaho Falls awaiting final grading and washing. Because the original screens permitted excessive fines in the final sand, a new screen was ordered. ANL has performed a preliminary grading check on the sand. With the removal of fines, the filter sand should conform to the specification. A double water wash is required to remove the clay adhering to the sand.

During the month, as the result of a rain storm, some water entered the storage structure for the bed and tables. This water was removed by the contractor. In order to provide fire protection, a fire-detection system was installed in the storage structure and connected to the EBR-II fire station alarms system.

The month's end status of reactor components is as follows:

Poison Rod Drives: The poison-rod drives were completed and checked out at the vendor's plant. These were crated and ready to ship.

Personnel Shields: The first unit has been assembled and checked at the vendor's plant. Minor adjustments were being made.

Matrix Alignment: The alignment plates were checked at the factory and arrived on May 29. It appears that they were shipped in a flat position instead of in the edge position. Fortunately, the rib adjustments are capable of realigning the plates, if necessary. The remaining assembly equipment except the tube holding clips has been received.

Source Drive: The source drives were inspected and checked out at the vendor's plant by an ANL representative. They were then shipped to the ANL-ID West Area for storage.

Rod Drive Mounting Plates: The machining for the rod-drive mounting plates is scheduled for July 1.

Fuel Control Drawers: The drawers have been fabricated and were crated for shipment to be Scottsonized.

Boron Fabrication: The first attempt by the fabricator to hot press B_4C powder in the new furnace resulted in a density of 97-100% theoretical. The specifications call for 93% of theoretical. This is part of trial production runs before using the enriched boron being supplied by ANL.

Alpha Air Monitors: Specifications are being prepared for the ZPPR alpha air monitors.

AFSR Safety Plug: The ball screws, air cylinder, and gear motor have been received for the modification of the AFSR safety plug.

5. Plutonium Fuel Procurement

Fabrication of plutonium fuel elements for ZPPR, ZPR-6, and ZPR-9 is being carried out by the following process. The plutonium, from several sources and of different isotopic compositions, is being blended to a Pu^{240} content of 11.5% and cast as ingots, each weighing approximately one kilogram. The ingoted plutonium is alloyed with uranium and molybdenum to a fissile plutonium content of 25 w/o; the alloy composition is approximately U-28.3 w/o Pu-2.5 w/o Mo. This composition is superior in strength and oxidation resistance to the corresponding uranium-plutonium binary alloy.

The cast alloy bars are parted to core-plate length and are adjusted to specified weight by machining within the width tolerance of 1.900 to 1.960 in. The thickness of 0.200 to 0.205 in. is obtained as cast. All edges are chamfered to reduce chipping and make loading of the core plates into jackets easier. In order to achieve flexibility of experimental assembly construction,

elements of 4-, 5-, 6-, 7-, and 8-in. length are required; the cores for these elements are 0.200 in. shorter than the element length.

The core plates are jacketed in 0.015-in.-thick Type 304L stainless steel sleeves between end plugs that are welded to the sleeve. Tight fit of the end plugs and cores is assured by maintaining a 50-lb compressive load on the assembly while the weld is being made. The space between the sleeve and core is filled with argon helium gas at $1/2$ atmospheric pressure. The jacket sleeves are made from vacuum-melted Type 304L stainless steel strips from which welded tubes are formed. These are drawn to 1.350-in. ID by 0.015-in. wall tubing.

The tubing is annealed, flattened, and stretch formed by means of an expanding mandrel to inside dimensions of 1.960 x 0.210 in. Jacket sleeves are cut to length from the formed tubes. The end plugs are shaped by wire-drawing methods and machined to length.

Blending and ingoting of the plutonium is being done by Isochem Incorporated at Richland, Washington. As of May 31, 1075 kg of the plutonium have been ingoted. Analyses of about one-half of the blending batches showed the following results:

Summary of 93 Isotopic Analyses

<u>Isotope</u>	<u>w/o Range</u>	<u>Average w/o</u>	<u>2σ Variation</u>
Pu ²³⁸	0.011-0.067	0.032	0.018
Pu ²³⁹	86.03-87.57	86.79	0.414
Pu ²⁴⁰	11.27-11.88	11.54	0.330
Pu ²⁴¹	1.307-1.752	1.481	0.190
Pu ²⁴²	0.103-0.177	0.126	0.048

The Am²⁴¹ varied from 103 to 1182 ppm, with an average of 450 ppm. Approval has been given for the blending of an additional 375 kg of plutonium during the month of June.

The first 800 kg of ingots have been designated for the production of 3762 fuel elements, containing approximately 700 kg of fuel, by Dow Chemical Company at the AEC Rocky Flats Plant. By May 31, the installation of equipment and tooling to fabricate the fuel elements was essentially complete. The first castings were made on May 23; these showed satisfactory mold filling and surfaces. The castings were internally sound, but the analyses indicated that a slight adjustment of the melt charge will be required to meet the target composition.

The inert-atmosphere gloveboxes, containing three milling machines, were installed and undergoing final leak testing before they were filled with argon gas. The gloveboxes, core-plate cleaning, inspection of the core plates, and assembly of the fuel elements were completed, and the equipment was being installed. The two welding machines were completed and were under test. Preliminary welds had been made at atmosphere pressure; they appeared to meet specification requirements.

Parameters are still being worked out for the subatmosphere welding. Preliminary manufacturing and quality-control procedures were written and submitted for Argonne National Laboratory review prior to the manufacture of process-qualification elements.

Stainless steel for jackets was vacuum melted and rolled into strip by the Allegheny Ludlum Steel Corporation. The strip was formed, welded, and drawn into tubing by the Pacific Tube Company for the entire requirement of jacket sleeves. About 5,300 ft of this tubing was received at Argonne National Laboratory during the month of May. This tubing was well within specified tolerances and of excellent quality.

Difficulty has been experienced in getting commercial fabricators to bid on the tube-forming operation because the close tolerance of wall thickness has precluded forming by conventional draw bench techniques. Argonne National Laboratory has set up tooling for a demonstration of process run of 8,000 jacket sleeves made by the stretch-forming process. The same technique will be used to form the sodium and sodium carbonate cans. Presently the jacket sleeves are being milled to finished length; this will be too costly for the full requirement, and a die-cutting method is under investigation. The problem is to support the rectangular tube wall so that it is not distorted and so that the burrs are minimized and easily removed.

The end plugs have been formed by Turks-head drawing. They are presently abrasively cut to length and the ends contour ground. A shaving operation is being investigated to finish the large requirement of end plugs. "Dead" annealed end plugs were found necessary to prevent warpage during the welding operation.

C. Other Fast Reactor Physics

1. Determination of Burnup of Fast Reactor Fuels

The program for the development of analytical methods for the determination of burnup of fast reactor fuels and for the measurement of fast fission yields is continuing. The development of a burnup method using

praseodymium-141 (stable) as a monitor was reported (see Progress Report for November 1966, ANL-7279, p. 18). In this method, praseodymium is separated from uranium and the other fission products, including the other rare earths, by a series of precipitation and ion-exchange separations. Praseodymium assay is accomplished by neutron activation, using manganese as an internal flux monitor, and subsequent gamma-spectrometric determination of the activation product, praseodymium-142.

This procedure has been used to determine the Pr^{141} content of a sample of EBR-I, Mark-III fuel ($\sim 0.3\%$ burnup). Results from five aliquots of a solution of the fuel gave an average Pr^{141} content of $10.817 \pm 0.193 \mu\text{g/ml}$ ($4.621 \pm 0.082 \times 10^{16}$ atoms/ml). To determine a value for the fast fission yield of Pr^{141} using the above data, it was necessary to relate the Pr^{141} content of the fuel to that of a fission product whose yield is known, namely, Cs^{137} . This was accomplished by analyzing other aliquots of the same EBR-I fuel solution for Cs^{137} by comparative gamma spectroscopy. Aliquots of a Nuclear Chicago ASTM cesium-137 standard and aliquots of the EBR-I fuel solution were presented alternately to a lithium-drifted germanium-diode detector system. Comparison of the counts under the 0.662-MeV Cs^{137} photopeak, after appropriate in-pile and out-of-pile decay corrections, gave a Cs^{137} content of $4.816 \pm 0.069 \times 10^{16}$ atoms/ml for the fuel solution. The ratio of cesium-137 to praseodymium-141 fission yields calculated from the above results is 1.04 ± 0.024 .

D. Component Development

1. Sodium Technology and Development

a. Fuel-pin Thermocouple. Thermocouples are being developed to indicate fuel-pin temperatures up to 2800°C in FFTF. For this development, previous experience with fuel-pin thermocouples have been evaluated, technology is being established to assure satisfactory design and performance, thermocouple assemblies and materials will be tested and evaluated extensively both out-of-pile and in-pile, and thermocouple fabrication techniques and specifications will be formulated.

The determination of the resistivity of the 99.9%-pure vitrified thorium insulators has begun. Vitrified beryllia and hafnia insulators have been ordered.

Capsule TU-2, a tantalum capsule containing UO_2 and dry stagnant helium, began to leak after 10.2 hr at 2410°C . Upon removal from the furnace, the capsule was found to be intact, and its exterior surface exhibited the large grain structure expected in tantalum heated to that temperature. A leak test showed that the capsule was leaking in the region of

the fuel. A length of tantalum tubing, identical to that used for the capsule construction but containing no UO_2 , was heated in the same manner; this piece of tubing, identified as sample T-1, had the same external appearance as capsule TU-2 after heating, but was found to be leaktight. When capsule TU-2 was disassembled, the UO_2 was found to be largely intact but somewhat spalled. Some regions of the interior wall of the tantalum were blackened with a fine powder, but no gross attack of the tantalum was evident.

Capsules TU-1 and TU-2, tantalum sample T-1, and an unheated piece of identical tantalum tubing, identified as sample T-2, are being examined metallographically.

The evaluation of reactor facilities suitable for in-pile testing the fuel-pin thermocouples has narrowed the choice to the Engineering Test Reactor (ETR) and the High Flux Beam Reactor (HFBR).

b. In-core Flowmeter. So that sodium flow through individual FFTF subassemblies or process tubes can be measured, technology is being established to assure that small magnetic, turbine, or other types of flowmeters can be obtained for use in severe temperature and radiation environments. Performance and reliability will be demonstrated in out-of-pile and in-pile tests.

(i) Thermal and Mechanical Tests of Magnet Materials. Thermal tests of the Alnico rod magnet samples A5-1 and A5-2 are continuing. Additional rod magnet samples of Alnico-5, -6, and -8 with length-to-diameter (L/D) ratios of 2, 4, and 8 are being prepared to determine the irreversible and material effects of these materials. An additional sample having L/D of 1 is also being considered for the Alnico-8 magnetic material. Alnico-8 is very brittle and breaks when subjected to modest shocks. The ability of this material to withstand mechanical shock as well as any loss of remanence is being determined. Discussions with magnet manufacturers relating to the use of Alnico-8 as a possible flowmeter magnetic material are continuing.

The furnace and turntable to measure magnetic properties with the magnetometer are being fabricated. A small, nonheated turntable is also being constructed for room-temperature magnetic measurements. Equipment for measuring the hysteresis loops of the magnets at room temperature after a heat cycle is being assembled.

(ii) Magnetic-flowmeter Prototype. The preliminary prototype permanent-magnet flowmeter has operated successfully in the sodium calibration loop at temperatures up to 700°F. Preliminary calibrations have been made at 400, 500, 600 and 700°F (see Table XV). All the data were

least-squares fit to determine the best straight-line characteristic; the equation, for instance, for 500°F is $E = 0.01711Q - 0.003$. The calculated sensitivity in Table XV is based on the design equations of Popper and Glass.¹ The measured offset is the zero-flow intercept as determined from the least-squares fit of the data. The maximum deviation from the straight line generally occurs at low flow rates, as is expected from an error analysis.

TABLE XV. Preliminary Data for Flowmeter Calibration

Temp (°F)	Sensitivity (mV/gpm)		Measured Zero Offset (mV)	Max Deviation of Any Data Point from Least-square-fit Line (% of Reading)
	Calculated	Measured		
400	0.01720	0.01708	-0.001	4.4
700	0.01643	0.01561	-0.004	4.7
600	0.01671	0.01681	-0.006	4.4
500	0.01699	0.01711	-0.003	2.3
400*	0.01720	0.01788	-0.004	3.9

*After operation at higher temperatures.

The ~4.5% increase in the measured sensitivity at 400°F after operation at higher temperatures is attributed to incomplete wetting initially. This effect could also account for the large discrepancy between the measured and calculated sensitivity at 700°F (the recirculation-loop portion of the calibration facility could not be heated above ~650°F on a continuous basis). The recirculation-loop heating capability has been increased, and operation of this portion of the facility at 1200°F is now expected to be achieved readily. After further calibrations indicate complete wetting has occurred, a long-term magnet stability run will be started. The flowmeter times at various temperatures are summarized in Table XVI; the flowmeter has operated satisfactorily for a total of ~350 hr above 400°F.

TABLE XVI. Temperature History of Flowmeter
at the End of May 1967

Temp (°F)	Time (hr)	Temp (°F)	Time (hr)
200-300	8.2	500-600	57.7
300-400	16.0	600-700	136.8
400-500	144.5	700-800	4.7

¹Popper, G. F., and Glass, M. C., The Design and Performance of a 1200°F Magnetic Flowmeter for In-core Application in Sodium-cooled Reactors, IEEE Trans. Nucl. Sci., NS-14, No. 1, pp. 342-347 (February 1967).

(iii) Magnet Irradiation Tests. Preparation of the experiment description and hazard analysis report has begun. Detail design of the irradiation capsule is continuing. Methods of magnet encapsulation in stainless steel covers are being studied; fabrication will begin as soon as the best overall technique is selected. Techniques to handle and identify the irradiated magnets in the hot cell are being developed; a magnet charger for the demagnetized irradiation samples is being prepared.

c. Fission-gas Pressure Transducer. To obtain an indication of the fission-gas pressure buildup in individual fuel pins while they are operating in a reactor, commercially available pressure transducers are being reviewed and evaluated for FFTF conditions, and will be tested out-of-pile and in-pile.

(i) NaK-filled Capillary System. The prototype system (see Progress Report for April 1967, ANL-7329) was tested further to include temperature effects up to 1200°F on heating as much as 9 ft of capillary tube leading to the bellows. Attempts were made to determine the zero-shift effect by (1) heating the bellows only, (2) heating the bellows and 4.5 ft

Fig. 16. Correction for Heated Capillary-tube Length of 9 ft

of capillary tube, and (3) heating the bellows and 9 ft of capillary tube; results were ambiguous. What appeared initially to be three defined curves were not adequately reproducible; the data ran together so much that it would seem that there was little effect of length of capillary tube on the zero-shift correction for the system. After additional tests, however, some reproducibility was found for the 9-ft length. Figure 16 shows the effect of temperature on 9 ft of capillary tubing and bellows.

It is interesting to note that beyond 800°F the curve flattens at about 0.300 mV, which is equivalent to ~4-psi zero shift. Therefore, careful temperature measurement beyond 800°F is not necessary for making zero-shift correction in this particular system.

A pressure transducer having a 1/8-in.-dia nickel-deposited bellows is being constructed; difficulty is being experienced in welding the bellows to an end fitting. It is planned to connect this bellows to a 0.018-in.-ID capillary tube, 12 ft long.

d. Signal Lead Connectors. To develop remote connectors for electrical and fluid signal leads for in-core instrumentation, connector lead-wire assembly designs and test procedures will be reviewed, changes

made to meet FFTF design criteria, new connector concepts considered, out-of-pile test programs established, and methods for terminating the sensor leads inside the reactor vessel, both above and within the sodium, will be studied and developed.

The test run described previously (see Progress Report for April 1967, ANL-7329) was terminated because the automatic temperature-control equipment failed, so that sodium in the test tank heated to 750°C, which caused one of the sodium-level probes to leak. No other damage to the system or to the connector was apparent, but it was decided to open the test tank and inspect the whole system. Inspection revealed that ~5 cm³ of sodium had entered the connector cavity. The connector showed signs of deterioration at both the bottom and the top. It was not possible to determine the actual way by which the sodium had entered (i.e., whether the sodium inside was the cause or the effect of the failure). Although the copper cladding of the gasket showed signs of oxidation and was spalled in many places, the gasket itself showed no measurable distortion; the sealing face of the connector itself was in nearly perfect condition, and no sodium was found inside the gasket. On the other hand, the bottom plug of the connector, which has a solid copper ring and was tightly torqued (and therefore thought to be leaktight), showed severe pitting of the sealing surface all the way to the thread. The sodium probably entered the connector through the bottom plug. Because of the questionable integrity of the test system, leak-rate measurements taken during the run have little significance. All that can be said is that the connector withstood a temperature cycle up to 750°C in a serviceable condition and that the leak rate was not large. The tested gasket may have sealed perfectly, but its copper cladding showed definite signs of deterioration.

After system cleanup and minor modifications, the bottom opening of the connector was welded closed, all components were leak-tested, and the connector was reassembled. An unplated Inconel-X seal identical to the previously installed gasket, except for the elimination of the copper cladding, was used. After several days of testing, this assembly has not shown any leak greater than what could be called "noise." Better measurement methods are being investigated.

e. Failed-fuel Locator. To find a method for determining which subassembly within a shutdown reactor core has one or more failed fuel pins, the feasibility of sensing the fission-gas release from the subassembly is being investigated. To prove the principle, a scheme will be demonstrated for efficiently disengaging noble gases from a continuously circulating liquid-sodium system at 1200°F. The equipment will include devices for disengaging the gas, trapping sodium vapor, viewing, monitoring radiation, and for injecting gas.

Modifications to the sodium-vapor-trap flow system were completed. A mass-spectrometer-valved "standard leak" was installed above the cover-gas system to admit a known small quantity of helium into the argon-gas supply of the high-temperature sodium-vapor trap. Tests to calibrate the sensitivity of the system to reject sodium but to pass helium to a mass-spectrometer leak detector will be conducted.

The performance of the vapor trap in returning liquid sodium to the supply tank was monitored by a series of photos taken through the sight glass containing the argon-gas blower. Buildup of sodium has been negligible. Operation with sodium vapor involved 192 hr at 1200°F and 1756 hr at 1100°F in a continuous run. The temperature was reduced from 1200 to 1100°F to conserve the integrity of the tubular heater while meeting the FFTF near-term design criterion of 1100°F.

2. Liquid Sodium Coolant Chemistry

a. Carbon Transport in Sodium Systems. The mechanisms of carburization and decarburization of structural materials in sodium systems are not well understood. Previous investigations of carbon transport have been hampered by uncertainties as to the nature and origin of the migrating carbon-bearing species. To aid in the detection, assay, and identification of such species, carbon tagged with beta-active carbon-14 ($t_{1/2} = 5770$ yr) is being used to trace the migration of carbon between structural materials that are exposed to liquid sodium.

In a preliminary experiment, the frequently reported finding, that carbon migrates from a high-carbon-activity ferritic steel via liquid sodium to a low-carbon-activity austenitic steel, was corroborated by the tracer technique. A cylindrical crucible (2.4 cm in dia by 3.5 cm) and matching lid were fabricated from an 18-mil-thick sheet of an Fe-C alloy enriched with carbon-14. Enough molten, reactor-grade sodium (9 g) was added to the crucible to completely immerse a stainless steel 304 coupon (2 cm by 1.5 cm by 10 mils). The coupon was suspended in the sodium by a tantalum wire welded to the crucible lid. The crucible was welded shut, heated at 650°C for 150 hr, cooled to room temperature, and opened for analysis. The filling, welding, opening, and sampling were done in a helium-atmosphere glovebox.

The surface of the coupon, after being washed free of sodium with ethyl alcohol, was found to be highly beta active. Moreover, significant amounts of beta activity persisted after polishing off several mils of the surface, which indicates that the tagged carbon had diffused into the coupon. Autoradiographs of the coupon showed a fairly uniform distribution of the carbon-14. The coupon and sections of the crucible that had been exposed

to molten sodium were analyzed for total carbon content, with the following results:

	<u>Crucible</u>	<u>Coupon</u>
Initial C, %	0.0263	0.075
Final C, %	0.0020	0.53

Since the initial carbon contents of the coupon and sodium could account for, at most, 20% of the coupon's final carbon content, it was concluded that the carbon in the ferritic crucible had been transported to the austenitic coupon.

The total carbon-14 content of the sodium, which represents carbon originating from the crucible, was also determined. The carbon from this source amounted to 0.17 ppm. A separate experiment showed that this tagged carbon could not pass through a 5 μ -porosity filter. [The value 0.17 ppm C is considerably greater than the measured solubility of carbon in sodium, <0.005 ppm (Ref. 2) up to 450°C]. Taken together, these observations suggest that the transport of carbon in sodium systems may involve carbon-bearing particulates.

E. Fuel Development

1. Metal Fuels

a. Irradiation of Uranium-Plutonium Alloys. With the renewal of operation of EBR-II, a slight increment of burnup has occurred in two fuel elements (U-15 w/o Pu-10 w/o Zr and U-15 w/o Pu-10 w/o Ti) being irradiated. The fuel elements are identified in Table XVII, and the new accumulated burnups are shown.

TABLE XVII. Status of Metal-fuel Irradiations in Progress

S/A No.	Spec. No.	<u>Design Parameters</u>					<u>Operating Conditions</u>			
		Fuel Composition (w/o)	Effective Density (%)	Clad Composition (w/o)	Clad OD (in.)	Clad Thickness (in.)	Max kW/ft	Max Clad Temp (°C)	<u>Burnup to Date</u>	
									a/o	fiss/cc x 10 ⁻²⁰ ^a
XG05	ND 24	U-15 Pu-10 Zr	63.6	V-20 Ti	0.208	0.015	10.0	535	5.1	11.7
XG05	NC 17	U-15 Pu-10 Ti	66.7	V-20 Ti	0.204	0.016	10.0	540	5.3	12.2

^aBased on effective density.

²Luner, C., Cosgarea, A., Jr., and Feder, H. M., Solubility of Carbon in Sodium, Paper SM-85/23, Symposium on Alkali-Metal Coolants--Corrosion Studies and System Operating Experience, IAEA, Vienna, November 28-December 2, 1966.

Fifteen U-15 w/o Pu-10 w/o Zr fuel elements, designated as Group M-3, have been shipped to Idaho and are waiting to be put in EBR-II. These fuel elements are identified in Table XVIII. Target burnup for these elements is 10 a/o at a maximum jacket temperature of 630°C.

TABLE XVIII. Dimensions of Group M-3 Specimens and Jackets

Fuel Material: U-15 w/o Pu-10 w/o Zr (93%) enriched
 Fuel Dimensions: 13.5 in. x 0.144 in. dia

Capsule/Specimen Number	Cladding			Specimen (in.)	Plenum Length (in.)	Specimen Plenum Volume (cc) (R.T.)
	Material	ID (in.)	Wall (in.)			
ANL-BA-01	304 SS	0.166	0.015	33.00	17.97	6.51
ANL-BA-02	304 SS	0.166	0.015	33.00	17.97	6.51
ANL-BA-03	304 SS	0.166	0.015	33.00	17.97	6.51
ANL-BA-04	304 SS	0.166	0.015	33.00	17.97	6.51
ANL-BA-05	304 SS	0.166	0.015	32.61	17.57	6.36
ANL-BB-01	304 SS	0.166	0.020	29.80	14.76	5.35
ANL-BC-01	316 SS	0.166	0.015	33.00	17.97	6.51
ANL-BC-02	316 SS	0.166	0.015	33.00	17.97	6.51
ANL-BC-03	316 SS	0.166	0.015	33.00	17.97	6.51
ANL-BD-03	V-20 Ti	0.166	0.015	24.68	9.65	3.49
ANL-BD-04	V-20 Ti	0.166	0.015	24.68	9.65	3.49
ANL-BD-05	V-20 Ti	0.166	0.015	24.68	9.65	3.49
ANL-BE-01	V-20 Ti	0.166	0.015	22.70	7.67	2.77
ANL-BE-02	V-20 Ti	0.166	0.015	22.70	7.67	2.77
ANL-BE-03	V-20 Ti	0.166	0.015	22.70	7.67	2.77

2. Oxide Fuels

a. Mixed-oxide Fuel-performance Studies. Postirradiation examinations of two mixed-oxide elements, which were among those being examined in March, continued through April and May in the Metallurgy Alpha-Gamma Hot Cell; additional elements remain in the EBR-II for further irradiation. Listed in Tables XIX, XX, and XXI are descriptions and operating conditions for each element.

TABLE XIX. Design Parameters and Operating Conditions for Mixed-oxide Elements Irradiated in EBR-II and Now Undergoing Postirradiation Examination

	Element Number	
	SOV-5	SOV-6
Fuel Composition	UO ₂ -20 w/o PuO ₂	UO ₂ -20 w/o PuO ₂
Fuel Synthesis	Pneumatically Impacted	Pneumatically Impacted
Fuel Form	Vibratorily Compacted	Vibratorily Compacted
Fuel Diameter, in.	0.254	0.254
Fuel Length, in.	11.5	11.5
Effective Density, %	83.4	83.3
Cladding Material	Type 304 SS	Type 304 SS
Cladding OD, in.	0.296	0.296
Cladding Wall, in.	0.021	0.021
Max Heat Rating, kW/ft	19.0	19.8
Max Clad Temp, °C	595	605
Burnup (est), a/o	2.7	2.9

TABLE XX. Status of Irradiations Vibratorily Compacted of (U-20 w/o Pu)O₂ in Progress in Subassembly XO11 of EBR-II--Fast Reactor Development

Spec. No.	Design Parameters				Operating Conditions			
	Effective Density (%)	Clad Composition	Clad OD (in.)	Clad Thickness (in.)	Max kW/ft	Max Clad Temp (°C)	Burnup to Date	
							a/o	fiss/cc x 10 ⁻²⁰ ^a
SOV-3	83.1	304 SS	0.296	0.021	21.4	610	3.6	7.5
SOV-7	85.1	304 SS	0.296	0.021	23.3	630	3.5	7.4
SOV-1	79.9	304 SS	0.296	0.021	20.03	590	3.5	6.9
HOV-10	84.8	Hast.-X	0.295	0.016	24.1	615	3.4	7.2
HOV-15	79.8	Hast.-X	0.295	0.016	21.4	590	3.4	6.8
HOV-4	80.0	Hast.-X	0.295	0.016	22.0	600	3.6	7.1
TVOV-1	76.8	V-20 w/o Ti	0.298	0.022	19.6	575	3.5	6.6

^aBased on effective density.

TABLE XXI. Status of Type 304 Stainless Steel-clad Cermet-fuel Irradiations in Progress in Subassembly XO11 of EBR-II--Fast Reactor Development

Spec. No.	Design Parameters					Operating Conditions			
	Fuel Composition (w/o)	Fuel Form	Effective Density (%)	Clad OD (in.)	Clad Thickness (in.)	Max kW/ft	Max Clad Temp (°C)	Burnup to Date	
								a/o	fiss/cc x 10 ⁻²⁰ ^a
5P-9	SS-40 PuO ₂	COEX ^b	98.0	0.301	0.015	10.4	495	4.2	3.2
5P-12	SS-27 PuO ₂	COEX	98.7	0.294	0.015	6.7	450	4.2	2.1
5U-14	SS-27 UO ₂	COEX	97.9	0.298	0.013	5.8	435	3.2	1.6

^aBased on effective density.^bCOEX = coextrusion of core and cladding.

Metallic inclusions observed in the fuel column of Element No. SOV-6 (see Progress Report for March 1967, ANL-7317, pp. 52-53) were qualitatively analyzed by an electron microprobe. They were found to consist mainly of ruthenium and molybdenum, with smaller amounts of technetium, rhodium, and palladium. Their optical appearance was similar to that of much larger particles found in the bottom of the central void. The typical inclusions in the unsintered peripheral zone, adjacent to the cladding, were less than about 1 μ dia. Inclusions in the sintering transition zone and in the columnar grains were about 5 μ dia and several of the particles in the bottom of the central void were larger than 225 μ in diameter.

Microprobe analysis disclosed the presence of significant amounts of barium, strontium, and cerium in a gray phase located in grain boundaries in the sintering transition zone.

Analytical and further microprobe tests are scheduled for samples being removed from Element No. SOV-6. Element No. SOV-5 is being held intact for transient testing in TREAT.

3. Carbide Fuels

a. Mixed-carbide-fuel Irradiations in EBR II. Irradiation of the 15 elements listed in Table XXII continued in EBR-II, and examination of the first three elements removed from EBR-II (see Table XXIII) continued in the Metallurgy Alpha-Gamma Hot Cell.

TABLE XXII. Status of Mixed-carbide-fuel Irradiations in Progress in EBR-II

S/A No.	Spec. No.	Design Parameters					Operating Conditions				
		Fuel Composition (w/o)	Fuel Form ^a	Effective Density (%)	Clad Composition	Clad OD (in.)	Clad Thickness (in.)	Max kW/ft	Max Clad Temp (°C)	Burnup to Date	
										a/o	fiss/cc x 10 ⁻²⁰ ^b
XG05	SMV-2	UC-20 PuC	VIPAC	83.8	304 SS	0.296	0.021	24.0	645	5.1	14.2
XG05	HMV-5	UC-20 PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	26.6	670	5.3	14.0
XG05	NMV-11	(U _{0.8} -Pu _{0.2})C	VIPAC	82.9	Nb-1 w/o Zr	0.284	0.012	25.5	645	5.4	14.7
XO08	NMP-2	(U _{0.8} -Pu _{0.2})C	PELLET	81.6	Nb-1 w/o Zr	0.284	0.013	17.4	545	3.3	9.0
XO08	NMV-4	UC-20PuC	VIPAC	80.0	Nb-1 w/o Zr	0.283	0.013	27.0	635	4.8	12.7
XO08	NMV-7	UC-20PuC	VIPAC	80.0	Nb-1 w/o Zr	0.283	0.013	26.0	605	4.4	11.6
XO08	NMV-12	(U _{0.8} -Pu _{0.2})C	VIPAC	85.5	Nb-1 w/o Zr	0.284	0.014	26.7	635	4.8	13.6
XO08	HMV-1	UC-20PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	25.2	640	4.3	11.4
XO08	HMV-4	UC-20PuC	VIPAC	80.0	Hastelloy-X	0.295	0.016	27.2	670	4.7	12.5
XO08	HWMP-1	(U _{0.8} -Pu _{0.2})C	PELLET	81.4	Hastelloy-X + W	0.296	0.020	16.9	555	3.3	8.8
XO08	HWMP-1	(U _{0.8} -Pu _{0.2})C	VIPAC	82.9	Hastelloy-X + W	0.295	0.018	26.4	685	4.8	13.1
XO15	NMP-1	(U _{0.8} -Pu _{0.2})C	PELLET	81.6	Nb-1 w/o Zr	0.284	0.013	16.5	535	0.75	2.0
XO15	NMV-3	UC-20PuC	VIPAC	80.0	Nb-1 w/o Zr	0.284	0.013	26.0	625	1.12	3.0
XO15	HMV-2	UC-20PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	27.0	670	1.22	3.2
XO15	TVMV-1	(U _{0.8} -Pu _{0.2})C	VIPAC	83.7	V-20 w/o Ti	0.298	0.021	23.4	620	0.98	2.7

^aVIPAC = vibratorily compacted into cladding.

PELLET = pressed and sintered pellets.

^bBased on effective density.

TABLE XXIII. Design Parameters and Operating Conditions for Carbide Elements Irradiated in EBR-II and Now Undergoing Postirradiation Examination

	Element Number		
	SMP-1	VMV-1	SMV-1
Fuel Composition, w/o	(U _{0.8} -Pu _{0.2})C	(U _{0.8} -Pu _{0.2})C	UC-20 w/o PuC
Fuel Synthesis	Arc Melted	Arc Melted	Arc Melted
Fuel Form	Pellets	Vibratorily Compacted	Vibratorily Compacted
Fuel Diameter, in.	0.253	0.253	0.257
Fuel Length, in.	13.4	14.0	14.0
Effective Density, %	81.4	85.9	80.0
Cladding Material	Type 316 SS	Vanadium	Type 316 SS
Cladding OD, in.	0.306	0.301	0.306
Cladding Wall, in.	0.024	0.024	0.025
Max Heat Rating, kW/ft	18.9	29.5	25.3
Max Clad Temp, °C	600	680	675
Max Burnup, a/o	2.1	3.02	2.59

Contrasting sintering behavior during irradiation had previously been observed in solid-solution and physically mixed materials loaded by the vibratory-compaction method. Reported in the April 1967 Progress Report, ANL-7329, were the results of examination of transverse and longitudinal sections of the elements at low magnification. Extensive sintering was disclosed in the solid-solution material but not in the physically mixed material at locations having comparable heat ratings. Investigations of sintering and structural disparities continued with microscopic examinations and autoradiography of the transverse and longitudinal sections.

Figure 17 shows a cross section removed from the hottest location in Element No. SMV-1 (physically mixed particles), which operated at 25.3 kW/ft. Particles have accommodated their shapes along mating surfaces, and voids have rounded surfaces. However, the original particle size is still evident, and many voids are still very irregular in shape.

Fig. 17. Cross Section at the Hottest Location in Vibratorily Compacted Element No. SMV-1 after Irradiation to 2.59 a/o Burnup at 25.3 kW/ft

11X

products moved radially outward, apparently concentrating in the peripheral, unsintered, fine-fraction PuC in Element No. SMV-1 or $(U_{0.8}-Pu_{0.2})C$ in Element No. VMV-1. Migration appeared to be more pronounced in the PuC fine fraction of the physically mixed material (Element No. SMV-1) than in the $(U_{0.8}-Pu_{0.2})C$ fine fraction of the solid-solution material (Element No. VMV-1).

b. Reactivity of Fluidized-bed Carbides with Impurities in Helium Atmosphere of Glovebox. The preparation of high-purity $(U-15 \text{ w/o Pu})C$ by a fluidizing-bed technique has been successfully demonstrated (see Progress Reports for December 1966, ANL-7286, p. 26, and February 1967, ANL-7308, p. 31). Efforts are now being directed at preparing fuel for irradiation testing from the fluidized-bed products by (1) pressing and sintering of pellets in the Chemical Engineering Division, and (2) preparation of dense granules for vibratory compaction in the Metallurgy Division. During the fabrication operations, the $(U,Pu)C$ powder will be exposed to a helium atmosphere in a glovebox; therefore, knowledge of the reactivity of the carbide with impurities in the helium is important.

Tests were reported earlier (ANL-7308, p. 31) in which the presence of adsorbed hydrogen was thought to have a significant effect on

Figures 18 and 19 illustrate alpha and beta-gamma autoradiographs, with corresponding macrographs, of physically mixed material and solid-solution material at the location of maximum temperature. The alpha autoradiographs, which indicate relative uranium and plutonium concentration, show that significant plutonium relocation did not occur in either material. Significant migration of fission products, however, was indicated by the beta-gamma autoradiographs. In both elements fission

Microphotograph--7.5X

Alpha Autoradiograph--7.5X

Beta-Gamma Autoradiograph--7.5X

Fig. 18. Autoradiographs of Center Section of Vibratorily Compacted UC-20 w/o PuC
Element No. SMV-1 after Irradiation to 2.59 a/o Burnup at 25.3 kW/ft

Macrophotograph--7.5X

Alpha Autoradiograph--7.5X

Beta-Gamma Autoradiograph--7.5X

Fig. 19. Autoradiographs of Center Section of Vibratorily Compacted $(U_{0.8}-Pu_{0.2})C$
Element No. VMV-1 after Irradiation to 3.02 a/o Burnup at 29.5 kW/ft

the reaction rate of (U,Pu)C with a helium atmosphere in a glovebox. However, the differences in reaction rates could also have been caused by differences in the carbon content of the powders. Recent results have shown that the effect of hydrogen is small or negligible for (U,Pu)C powders of identical carbon contents. A portion of one batch of fluidized-bed carbide (4.74 w/o C) that had been cooled under hydrogen was subsequently treated with helium at 800°C in a fluidized-bed reactor to remove the absorbed hydrogen. The hydrogen content of this helium-treated material was determined to be 8 ppm. The remainder of the batch was not treated with helium and retained a high hydrogen content of 190 ppm. The low-hydrogen (U,Pu)C gained 0.13 w/o in 15 days of exposure to helium containing 2 to 4 ppm H₂O and ~100 ppm O₂, whereas the high-hydrogen (U,Pu)C gained 0.12 w/o in a similar test. The reaction rate for the helium-treated specimen is shown in the lower curve of Fig. 20.

Fig. 20. Weight Gain in UC and (U,Pu)C Powder in Ambient High-purity Helium Atmosphere. H₂O = 2 to 4 ppm and O₂ $\cong$ 100 ppm.

In another test, the reactivity of UC powder prepared by the fluidized-bed technique was determined. The UC had essentially the same initial carbon (4.72 w/o) and oxygen (800 ppm) concentrations as the (U,Pu)C. The reaction rate for the UC powder is shown in the upper curve in Fig. 20. The weight gain for the UC was about six times as great as that for the (U,Pu)C. This difference is partly accounted for by the threefold difference in median diameter of the powders [$\sim 15\mu$ for UC versus $\sim 45\mu$ for (U,Pu)C] and consequent difference in surface area.

Short-term tests were also made in which the (U,Pu)C and UC powders were exposed to the purest helium atmosphere (2 ppm H₂O and 10 ppm O₂) attainable with the glovebox purification system. In $6\frac{1}{2}$ hr, the UC sample gained 0.073 w/o, whereas the (U,Pu)C sample gained 0.049 w/o.

It is apparent from these experiments that short exposure of the fluidized-bed carbide powders to a helium atmosphere in a glovebox during fabrication operations will not contaminate the powders significantly.

4. Fuel Cladding and Structure

a. Effect of Fast-neutron Irradiation on Jacket Materials. A total of 420 tensile-type and 103 tube-burst specimens of V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr, AISI Type 304 stainless steel, Hastelloy-X, and Inconel 625 are being irradiated in EBR-II subassemblies XA08, XO10, XO16, XO18, and XO20 at temperatures between 500 and 650°C.

The specimen temperatures of the vanadium-base jacket-material capsules from EBR-II subassembly XO13 were evaluated. The temperatures were ascertained by determining which pellets had melted in a group of four Tempil pellets placed at various positions in the specimen capsules. The melting temperature of the pellets ranged from 427 to 704°C. Not all of the pellet groupings could be evaluated, since some fell apart during disassembly with a subsequent loss in identity. The results of the evaluation are given in Table XXIV. The indicated specimen temperatures are grouped according to distance from the reactor center, which represents different rates of gamma heating. At a distance of 13.96 to 14.02 cm from the reactor center, the specimen temperatures in capsules AS-35, AS-36, and AS-45 appeared to range between 593 and 649°C. At a distance of 12.89 to 13.00 cm from the reactor center, the specimen temperatures in capsules AS-41, AS-44, and AS-54 appeared to range from 593 to 704°C in the bottom half of the specimen column and from 538 to 593°C in the top half of the specimen column. The specimen temperatures at a distance of 11.86 to 12.05 cm from the reactor center are inconsistent, and the specimen temperatures of capsules AS-38, AS-39, AS-41, and AS-43 range from 510 to 704°C. At a distance of 10.87 cm from the reactor center, the specimen temperatures in capsule AS-40 ranged from 538 to 593°C near the bottom half of the specimen column and from 649 to 704°C at the top of the specimen column.

TABLE XXIV. Evaluations of Tempil-pellet Temperatures from EBR-II Subassembly XO13

Position from Top of Specimen Column	Distance from Reactor Center (cm)	Capsule Number	Melting Temperature of Pellets (°C)	Indicated Specimen Temperature Range (°C)
1	13.96	AS-35	538, 593, 649, 704	593-649
3	14.02	AS-36	510, 538, 593, 649	593-649
3	14.02	AS-45	510, 538, 593, 649	593-649
7	13.96	AS-35	454, 510, 593, 704	510-593
11	14.02	AS-45	427, 454, 510, 593	593+
1	12.89	AS-54	538, 593, 649, 704	538-593
3	12.89	AS-44	510, 538, 593, 649	538-593
3	13.00	AS-42	510, 538, 593, 649	538-593
5	12.89	AS-54	510, 538, 593, 704	593-704
11	13.00	AS-42	427, 454, 510, 593	593+
1	11.86	AS-43	538, 593, 649, 704	649-704
1	12.05	AS-38	510, 593, 649, 704	510-593
3	11.86	AS-39	510, 538, 593, 649	649+
5	11.86	AS-43	510, 538, 593, 704	593-704
7	12.05	AS-38	454, 510, 593, 704	510-593
7	12.05	AS-41	454, 510, 593, 704	510-593
11	11.86	AS-39	427, 454, 510, 593	593+
1	10.87	AS-40	538, 593, 649, 704	649-704
5	10.87	AS-40	510, 538, 593, 704	538-593

Short-time tensile tests were conducted at a strain rate of 2% per minute at a temperature of 550°C with V-20 w/o Ti and V-15 w/o Ti-7.5 w/o Cr specimens from EBR-II subassembly XO13. The V-20 w/o Ti

was given a preirradiation heat treatment of one hour at 900°C, and the V-15 w/o Ti-7.5 w/o Cr was given a preirradiation heat treatment of one hour at 850°C. The results of the tests on the irradiated specimens were compared to the results of tests on unirradiated specimens and on unirradiated specimens given a 30-day anneal at 550°C to approximate anticipated in-reactor conditions. The results are given in Table XXV.

TABLE XXV. Postirradiation Tensile Properties of Vanadium-base Alloys at 550°C

Material (w/o)	Condition	Fluence (n/cm ² x 10 ²¹)	Estimated Irradiation Temp ^a (°C)	0.2% Yield Strength (kg/mm ²)	Ultimate Tensile Strength (kg/mm ²)	Uniform Elongation (%)	Total Elongation (%)
V-20 Ti	Unirradiated	-	-	36.1	59.2	18.2	24.6
V-20 Ti	Unirradiated	-	-	36.1	59.8	18.6	24.3
V-20 Ti	Unirradiated 30 days at 550°C	-	-	36.4	56.5	17.8	24.5
V-20 Ti	Irradiated	2.3	593-649	39.6	60.9	16.0	21.7
V-20 Ti	Irradiated	4.2	593-649	47.6	66.6	12.0	17.8
V-15 Ti-7.5 Cr	Unirradiated	-	-	41.7	65.7	16.8	21.9
V-15 Ti-7.5 Cr	Unirradiated 30 days at 550°C	-	-	40.6	65.7	15.8	21.2
V-15 Ti-7.5 Cr	Irradiated	2.2	593+	43.0	69.1	12.0	16.5
V-15 Ti-7.5 Cr	Irradiated	3.4	593-704	50.7	71.9	12.4	17.6

^aTemperatures estimated from Tempil pellets.

The 30-day anneal at 550°C did not appear to affect the strength or ductility of the unirradiated V-20 w/o Ti alloy. Irradiation to a fluence of 2.3×10^{21} n/cm² increased the yield strength over that of the unirradiated alloy by approximately 10% and decreased the percent total elongation by 11%. Irradiation to a fluence of 4.2×10^{21} n/cm² increased the yield strength over that of the unirradiated alloy by 32% and decreased the percent total elongation by 27%.

The 30-day anneal at 550°C did not significantly affect the strength or ductility of the unirradiated V-15 w/o Ti-7.5 w/o Cr alloy. Irradiation to a fluence of 2.2×10^{21} n/cm² increased the strength over that of the unirradiated material by 3% and decreased the percent total elongation by 25%. Irradiation to a fluence of 3.4×10^{21} n/cm² increased the yield strength over that of the unirradiated alloy by 22% and decreased the percent total elongation by 20%.

b. Fabricability of V-15 w/o Cr-5 w/o Ti. Scale-up and fabrication development of the V-15 w/o Cr-5 w/o Ti alloy has continued since the last progress report (see Progress Report for March 1967, ANL-7317, pp. 57-58).

Two production-size (3.8-in. dia) ingots have now been consolidated by practices established for the earlier V-15 w/o Ti-7.5 Cr alloys. Resulting chemical analyses are shown in Table XXVI.

TABLE XXVI. Chemical Analyses of Production (3.8-in. dia)
Ingots of V-15 w/o Cr-5 w/o Ti

Ingot Number	Alloy Content (w/o) ^a		Interstitial Content (ppm) ^b			
	Cr	Ti	Carbon	Oxygen	Nitrogen	Hydrogen
AM-101	15	5.1	235	815	150	50
AM-103	15.2	5.0	255	615	-	-

^aAverage of three values from top, center and bottom of ingot.

^bAverage of two values from top and bottom of ingot AM-103; from center and bottom on AM-101.

The first ingot (AM-101) was double-extruded and wrought into limited quantities of high-quality finished products for evaluations studies. The second ingot (AM-103) was broken down by primary extrusion. The resulting bar stock is being prepared for secondary extrusion into rod, sheet-bar, and tube-blanks.

A homogenization treatment (2 hr at 1200°C) of both the ingot and primary extruded product seems to improve subsequent fabricability. The secondary extruded products are presently being subjected to a 2-hr heat-treatment at 900°C before warm breakdown (300°C) rolling or tube-reducing. Subsequent working is carried out at room temperature. A typical microstructure of V-15 w/o Cr-5 w/o Ti 0.030-in.-thick sheet stock is illustrated in Fig. 21. All evidence of banding has been eliminated and the dispersed phase is well distributed.

150X

750X

Longitudinal Sections

Fig. 21. Microstructure of Rolled and Annealed, 0.030-in.-thick V-15 w/o Cr-5 w/o Ti Sheet Stock (50% Cold Work--900°C--One-hour Anneal).

While progress is being made in the secondary fabrication of V-15 w/o Cr-5 w/o Ti wrought products, optimum processing procedures have not been established. The failure of extruded material to submit to room-temperature working has been troublesome, and efforts to eliminate the presently used warm working operations will continue.

c. Corrosion Inhibition by Dissolved Getters in Liquid-sodium Environment. Construction of an isothermal, dissolved-getter loop has been completed. Apparatus and control systems for gas purification and high-temperature operation have been assembled to the loop. Satisfactory pumping operation has been demonstrated for sodium at 650°C. Departures from an isothermal condition of only about 3°C have been observed throughout the liquid-sodium section of the loop.

A reduction in flow rate from the design value was made to improve check-valve operation and prolong the life of the bellows pump after initial operation under the design conditions. The rate remains to be directly measured, but is believed to be satisfactory for the purposes of the program.

Spectroanalysis has revealed 500 wt ppm (certain within a factor of two) of calcium impurity in the test sodium employed. Samples from various alternative sodium sources have been submitted for analysis.

A new loop loading with reactor-grade sodium is planned, to initiate a study of the corrosion behavior of vanadium alloys in magnesium-gettered sodium at 650°C.

d. Development of Refractory-metal Alloys for Service in Oxygen-contaminated Sodium. Small weight gains were reported (see Progress Report for April 1967, ANL-7329) for V, V-5 w/o Cr, and V-15 w/o Ti-7.5 w/o Cr exposed for 117 days to flowing (6.1 m/sec) sodium containing about 8 wt ppm of oxygen (cold trap at 120°C) to differ from one another less in tests at 550°C than at 650°C. Specimens of V-15 w/o Cr-5 w/o Ti were introduced into the continuing test, and tests with specimens in low-oxygen sodium at 450°C were started.

The test at 550°C has been terminated after a 155-day period of exposure, and the weight gains are listed in Table XXVII. The alloys have a faint brown tint but otherwise appear metallic, and the weight gains are believed to represent the relative pickup of impurities (mainly oxygen) from the flowing stream. The weight gain of V-20 w/o Ti at 550°C was one-third the value obtained at 650°C. The other alloys were less affected by the reduction in test temperature.

TABLE XXVII. Corrosion of Vanadium Alloys in Sodium at 550°C,
Flowing with a 6.1-m/sec Velocity

Alloy or Oxygen Content ^a	Weight Gain (mg/cm ²) at Indicated Time (days)							
	6.9	20.9	27.8	53.8	65.8	88.7	116.5	154.5
V	0	0.06		0.08		0.16	0.16	0.20
V-5 Cr	0	0.032		0.064		0.075	0.075	0.097
	0	0.043		0.065		0.086	0.086	0.097
V-20 Ti	0.089	0.20		0.25		0.29	0.31	0.33
	0.079	0.21		0.26		0.30	0.33	0.33
V-15 Ti-7.5 Cr	0.077	0.15		0.22		0.30	0.32	0.32
	0.088	0.17		0.22		0.30	0.32	0.31
V-15 Cr-5 Ti			0.06		0.11			
Annealed			0.06		0.09			
V-15 Cr-5 Ti			0.07		0.13			
(Cold Work 50%)			0.07		0.13			
Oxygen Level, w/o (Hg amalg.)		7.8	8.5	9.7		6.5	8.5	

^aNumbers in composition formulas indicate percent by weight.

As usual, a small increase in corrosion was noted for cold-worked specimens (V-15 w/o Cr-5 w/o Ti) as compared with annealed samples. The corrosion tabs will be sectioned for metallographic examination and analyzed chemically to determine the extent and type of penetration.

Two oxygen-refreshed autoclave systems are continuing to operate at 450°C (cold traps at 110 and 175°C, respectively). Weight losses have been recorded for both systems after the initial one-week exposure. V-20 w/o Ti shows less weight change than either V-15 w/o Cr-5 w/o Ti or V-15 w/o Ti-7.5 w/o Cr in both experiments. Weight changes are relatively small at both oxygen levels [i.e., max loss of V-15 w/o Cr-5 w/o Ti (CT. 175°C) exposed 28 days, $\Delta W = 2.3 \text{ mg/cm}^2$; min loss of V-20 w/o Ti (CT. 110°C) exposed 35.6 days, $\Delta W = 0.3 \text{ mg/cm}^2$].

5. Fuel Reprocessing

a. Processes for Fast Reactor Fuels. Studies are being made of a pyrochemical process for ceramic (e.g., oxide or carbide) fast breeder reactor fuels clad with stainless steel. The process applies salt-transport separations and liquid metal-molten salt extractions to the separation of fissile and fertile fuel constituents from the fission products. A conceptual flowsheet for the process has been described previously (see Progress Report for September 1966, ANL-7255, pp. 30-31). In the currently proposed pyrochemical flowsheet the fuel is decled by dissolution of the stainless steel cladding in molten zinc.

A laboratory experiment was performed to determine whether the zinc used for decladding could be purified for reuse by precipitating the dissolved stainless steel constituents (Fe, Cr, and Ni) as aluminum intermetallic compounds. Aluminum metal was added to a liquid zinc solution containing ~4 w/o Fe, 0.7 w/o Cr, and 0.6 w/o Ni at 800°C. The quantity of aluminum added was sufficient to precipitate 80% of these metals as FeAl_3 , CrAl_7 , and NiAl_3 . The temperature was reduced from 800 to 500°C, and the liquid zinc phase was sampled. The zinc contained 0.005 w/o Fe, 0.025 w/o Cr, 0.018 w/o Ni, and 0.9 w/o Al. These results indicate excellent removal of the stainless steel constituents, but the quantity of aluminum which remained in the zinc could be great enough to cause reduction of the oxide fuel when the zinc solution is reused for decladding. It may be possible to reduce the amount of aluminum used since the Fe, Cr, and Ni removals were greater than would be required for this purpose.

The first step after the decladding of mixed-oxide ($\text{PuO}_2\text{-UO}_2$) fuels is the reduction of the oxide to metal (plutonium and uranium) by a Cu-33 w/o Mg alloy in contact with a molten salt (see ANL-7255, pp. 30-31). The alkali, alkaline earth, and rare earth fission products, and the solid MgO formed as a byproduct of the reduction reaction remain in the waste salt. The plutonium, uranium, and noble and refractory-metal fission products remain in the Cu-Mg alloy; the plutonium is in solution and the uranium is present as a precipitated metal phase.

Previous reduction experiments have indicated that the MgO reduction byproduct, which is relatively insoluble in the salt, interferes with the reduction reaction. Removal of the byproduct oxide from the reaction site should increase the rate of reduction. This may be accomplished by using a reductant such as calcium, whose oxide is soluble in the salt. Several exploratory reduction experiments have been performed in which calcium metal was added to the Cu-Mg alloy in slight excess of the amount required to reduce the fuel oxide. A $\text{CaCl}_2\text{-20 m/o CaF}_2$ salt mixture was used because of the high solubility of the CaO reduction byproduct in this mixture. The reductions achieved with this system (at 800°C and 800-rpm agitation) were 98% complete in 2 hr using powdered UO_2 , and >99% complete in ~4 hr using sintered UO_2 pellets.

Prepilot-plant engineering studies for the plutonium salt-transport process are being carried out in new experimental equipment that has been assembled and installed in a walk-in hood. The equipment will handle liquid-metal and molten-salt solutions containing up to 1 kg of uranium and 200 g of plutonium. Experiments in this facility will provide data on engineering concepts that will be incorporated in the design of a pilot plant for the demonstration of the plutonium salt transport process. A schematic diagram of the equipment is presented in Fig. 22. The equipment is similar to but smaller in scale than that used for the pilot-plant investigation of the salt-transport process for uranium (see Progress Report for May 1966, ANL-7219, pp. 21-24).

Fig. 22. Plutonium Salt-transport Equipment

A series of shakedown experiments has been performed with uranium, but without plutonium, to investigate individual steps of the plutonium salt-transport process. All of the process steps were successfully demonstrated in a final shakedown run using uranium and 28 inactive fission product elements. The first run with plutonium will be made following receipt of complete analytical results for the final shakedown run.

F. Design Concept Analyses and Advanced Systems Evaluation

1. 1000-MWe Plant

a. Combustion Engineering. At the first monthly progress meeting with Combustion Engineering, Inc., the contractor's plan and procedures were discussed. The contractor has supplied his procedure for LMFBR parametric core survey and physics analysis.

b. Babcock & Wilcox. At the monthly progress meeting, B&W representatives reported completion of the four conceptual designs in their Task-I effort. The description of Concept 1 has been issued in draft form; the other three descriptions are undergoing final editing. All four descriptions will be incorporated into the B&W Task-I report.

Representatives of Babcock & Wilcox, AEC, and ANL met with Byron Jackson Pumps, Inc., to exchange information developed by Byron Jackson on its pump-development contract and by Babcock & Wilcox on its 1000-MWe LMFBR Follow-on Study contract.

c. General Electric. The contract with General Electric Co. was executed; it provides that the work be done on a cost-plus-fixed-fee basis between May 2, 1967 and November 2, 1968:

General Electric methods for accomplishing and reporting the work were reviewed. The company has supplied first-draft PERT diagrams, which indicate completion of Task I of the contract on March 29, 1968, as well as expanded definitions of the system-evaluation parameters and brief descriptions of the planned study approach in each area.

G. General Research and Development

1. Fast-reactor Core-parameter Study

A breeding ratio has been calculated from the output of a two-dimensional diffusion problem as a check on the values obtained previously from one-dimensional techniques. The reactor considered had a core height-to-diameter (H/D) ratio of 0.6 and an average core power density of 400 kW/liter. The core conversion ratio was 0.8514 and the reactor breeding ratio was 1.279 (as compared with 0.8495 and 1.240 for the one-dimensional analysis). The core conversion ratios are almost identical, but the breeding ratio is about 3% higher in the 2D case. This appears to be caused by the blanket material in the corner area, which was not accounted for in the one-dimensional analysis.

A sodium-void coefficient has also been computed for the two-dimensional problem. At full sodium density, k was found to be 1.00090. The excess over unity is attributed to the corner area, which was not accounted for in the one-dimensional self-consistent diffusion-theory procedure from which the core composition was obtained. When 40% of the core sodium was removed, a k of 1.00940 was obtained, giving $\Delta k = 0.0085$. The result obtained earlier with a one-dimensional calculation (including a transverse buckling correction) for k was 0.0085. A similar check is being made at a core H/D of 0.2.

Two-zone cores have been studied in one-dimensional cylindrical geometry, assuming the same extrapolated height as the corresponding one-zone core treated previously by self-consistent diffusion theory. The cases considered have been for $H/D = 0.4$ and 0.8, at an average power density of 400 kW/liter. Calculations have been made for cores having outer zones made up of one, two, and three rings of subassemblies. For the case $H/D = 0.4$, the minimum radial peak-to-average source density was 1.234 at a critical mass of 1893 kg for a two-ring outer zone. For the

"unflattened" core, these values were 1.773 and 1.772 kg. Breeding ratios here were found to be affected very little by zoning. The sodium-void coefficient, however, was found to increase with the ratio of plutonium density in the outer core zone to that in the inner zone, as was expected. In going from the "unflattened" to "flattened" case above, the void coefficient for uniform removal of 40% of the core sodium increased by 27%.

One three-zone core was partially optimized to yield a radial peak-to-average source density of 1.159 and a critical mass of 1938 kg.

Work has begun on a study of the effects of thermal expansion of the core on reactivity.

The self-consistent diffusion-theory model used to study one-zone cores has been programmed for one-shot solution on the computer.

The computer code for performing reactor thermal-hydraulic analyses has also been revised to calculate the effective diameter of each of the distinct heat-transfer regions in a core. Use is made of the core-region dimensions to establish the fuel-enrichment zones in cores that are multizoned for power flattening.

Calculations are being made to determine the core compositions of carbide-fueled reactors. The basis for these calculations, with respect to attainable burnup, fuel lifetime, and thermal-hydraulic considerations, is similar to that used for the oxide-fueled cores currently being studied.

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Comparison of Experimental Neutron Spectra for a Zoned and a Homogeneous Fast Critical Assembly

Zoned reactors allow a substantial reduction in material inventory relative to reactors of the same central composition but homogeneous throughout. The usefulness of zoned reactors as a substitute for homogeneous reactors can be judged relative to the ease with which measurements can be referred to the homogeneous case.

An approximate equivalence of central neutron spectra in both configurations will simplify considerably any intercomparison of reactivity effects, and zoned reactors are generally designed to reproduce the reference central spectrum rather closely.

Experimental measurements of neutron spectra by proton-recoil techniques³ have been made with both Assembly 5 (homogeneous) and Assembly 12 (zoned) as constructed at the ZPR-6 and ZPR-9 facilities at Argonne. The composition of the homogeneous reactor simulates that of a large, dilute, uranium carbide power reactor. The zoned reactor had a central region (radius = 26 cm) with the composition of the homogeneous reactor and a concentric annulus with a high concentration of fissionable material.

Figure 23 is the spectrum as measured in the reference core (Assembly 5). Resolution is approximately 10% (fuel width at half maximum)

Fig. 23. Central Spectrum of Assembly 5 Compared with an Energy-group Calculation

³Bennett, E. F., Nucl. Sci. Eng. 27, 16, 28 (1967).

except at the lower energies, where it worsens because of statistics in proton ionization. The spectrum contains recognizable features. The large sodium resonance at 3 keV depletes the flux severely, as do resonances in the lighter elements in the core at various energies across the spectrum. The result in Fig. 23 is similar to an older measurement⁴ except that an improved energy calibration was available, a correction for a non-Gaussian component in the detector-response function (due to finite detector size) was made, and the resolution was improved.

A group histogram from the MC² code⁵ is also shown in Fig. 23; normalization is arbitrary. There is general agreement as to shape. Data below 5 keV are probably subject to systematic errors in excess of statistics, as discussed by Bennett.³

The spectra above 1 MeV are subject to a sizeable correction for wall-and-end effects. These have not been made, but should not greatly influence a comparison of spectra from the two assemblies.

Figure 24 is a plot of the percentage deviation of the zoned assembly from the homogeneous assembly versus energy. The measured spectrum for both cases was averaged over lethargy intervals of the same width (0.25) as the histogram in Fig. 23. The spectrum for the zoned assembly was normalized (by area) to the reference spectrum. Except at low energies (below 5 keV) the deviations observed were, for the most part, within estimated statistical errors.

Fig. 24. The Percent Deviation of the Zoned Spectrum from the Reference Spectrum vs. Energy

⁴Karam, R. A., et al., ANL-7281 (Dec 1966).

⁵Lewis, R. A., ANL, private communication.

Below 4 keV, differences in the spectra were generally in excess of statistics. These deviations are probably not real, however. Reproducibility of spectra at low energies has generally not been within statistics; the explanation appears to be instrumental in nature and improved counting apparatus can be expected to reduce these sources of systematic error substantially.

2. Reactor Computation (ARC)

A special version of RABBLE⁶ is being written for use in the analysis of measurements performed in the fast critical assemblies ZPR-6 and ZPR-9. This version, which is limited to slab cells, differs mainly from the standard RABBLE in its use of periodic boundary conditions at the surfaces of the cell, instead of the usual zero-current boundary conditions. These periodic boundary conditions specify that the rate and angular distribution at which neutrons leave one surface of the cell are identical to the rate and the angular distribution which neutrons impinge upon the other surface.

3. Evaluation of Cross Sections

The CRECT code, which was included on the ENDF/B program tape recently received from Brookhaven, has been made operational on the CDC-3600. This code makes corrections in ENDF/B BCD tapes. The input tape is copied onto another tape with the corrections being made as the copying proceeds. Corrections consist of deleting cards, inserting cards, or changing certain fields of existing cards.

Corrections and improvements in the preliminary version of the DAMMET code have been made in accordance with the version distributed on the ENDF/B program tape. The preliminary version had been made operational on the 3600 by APDA, since DAMMET is used to produce an ENDF/B binary tape suitable for input to ETØE from an ENDF/B BCD tape.

A code called MERMC2 is being written to merge two MC² library tapes, with deletions if desired. Thus, if it is desired to add a few ENDF/B materials to an existing MC² tape produced by ETØE, it will be necessary to run an ETØE problem with only the new materials rather than all the materials desired. Also MERMC2 could be used to merge an MC² tape produced by ETØE with an MC² tape not containing ENDF/B data which had been produced by the MC² library program LIB9. The MERMC2 code is almost complete.

⁶Kier, P. H., A Method of Computing Resonance Integrals in Multi-Region Reactor Cells, Intern. Conf. Research Reactor Utilization and Reactor Mathematics, Mexico City, May 1967.

B. Reactor Fuels and Materials Development

1. Chemistry of Fuel Materials

a. Electron Microprobe Analysis of Irradiated Oxide Fast Reactor Fuel Pins. Transverse cross sections obtained from the bottom of EBR-II UO_2 fuel pins (13% enriched) irradiated to 0.75 and 4.37 a/o burnup in MTR have been examined in an unshielded electron probe microanalyzer. A description of the pins and the analysis of a longitudinal section of the pin irradiated to 0.75 a/o burnup have been given previously (see Progress Report for March 1967, ANL-7317, p. 73).

Numerous small inclusions, up to several microns in diameter, having a metallic luster were dispersed throughout the matrix of both fuel pins. In addition, large porous inclusions up to $100\ \mu$ in diameter, which were round and dark, were found dispersed throughout the outer 0.6 mm of the 4.37 a/o burnup pin. The transverse cross section of the higher burnup pin is shown in Fig. 25 (a portion of the cladding broke off during polishing) and an enlarged view of the porous area is shown in Fig. 26. The porous areas contained uranium, cesium, barium, lanthanum, cerium, praseodymium, neodymium, samarium, and zirconium. Small shiny metallic inclusions within the porous area were stainless steel particles with no detectable fission products and stainless steel particles containing the transition metal fission products molybdenum, ruthenium, technetium, rhodium, and palladium.

Fig. 25

Transverse Cross Section of Irradiated
 UO_2 Fuel Pin (4.37 a/o burnup;
4 mm in diameter)

Fig. 26. Enlarged View of Porous Inclusions in Peripheral Area of Irradiated UO_2 Fuel Pin (4.37 a/o burnup)

Over a hundred of the shiny metallic inclusions and porous inclusions were examined in both pins. The results may be summarized as follows:

- (i) Stainless steel impurity inclusions migrated during irradiation to the outer portion of the fuel pin. No stainless steel inclusions remained in the columnar grain-growth region 0 to 0.10 cm from the fuel core void edge (see Fig. 25).
- (ii) All inclusions analyzed in the columnar grain-growth regions of both pins (ten in the low-burnup pin and 45 in the high-burnup pin) contained metallic-appearing alloys of molybdenum, ruthenium, technetium, and rhodium in a ratio approximating their fission yield. The small electron beam of the probe (about $1/3 \mu$ in diameter) allowed quantitative analysis of the largest of these inclusions ($2-3 \mu$). It was possible to demonstrate that these inclusions were free of uranium, indicating that no intermetallics or solid solutions of uranium-noble metal fission products were formed by reduction of the UO_2 matrix. Quantitative analysis of the smallest inclusions was not possible.

(iii) Inclusions in the peripheral areas of the pins were either stainless steel or stainless steel containing the transition metal fission products molybdenum, ruthenium, technetium, rhodium, and palladium. There was no consistent ratio of fission product distribution in the steel-containing inclusions, although molybdenum and ruthenium were the major fission products.

(iv) The porous inclusions were formed in the region beyond the columnar grains. These appear to be high-burnup regions associated with the fully enriched UO_2 particles ($\sim 20\text{-}100\ \mu$ in diameter) of the original fuel material.⁷ During the irradiation the temperature in this region remained below that required to effect the isotopic exchange of the fully enriched UO_2 and natural UO_2 particles. All the rare earth oxides, zirconium oxide, etc., also remained in the areas of the fully enriched and high-burnup UO_2 particles, and were essentially undissolved by the surrounding natural uranium oxide matrix. However, in the columnar grain-growth region the formation of solid-solution oxides and uranium isotopic exchange were both virtually complete. An examination of the uranium oxide in the center region showed dissolved cerium and neodymium throughout the oxide matrix, with a slight peak in concentration near the edge of the columnar grains. This is demonstrated in Fig. 27 in which the concentration of neodymium is shown as a function of position in the fuel pin. Figure 27 also shows most of the main features of the various areas of the fuel.

Fig. 27. Relative Neodymium Concentrations and Locations of Structural Features in MTR Irradiated, 13% Enriched UO_2 (Burnup 4.37 a/o)

The stainless steel was not a planned inclusion in the fuel and apparently originated from the fabrication equipment. Originally, stainless steel was dispersed throughout the fuel. The liquid stainless steel which

⁷ The oxide fuel, which was 13% enriched, was prepared by blending fully enriched and natural UO_2 particles and sintering the blended material.

diffused out of the hot central zone contained some of the noble metal fission products as metallic alloys. Stainless steel which originated far enough out in the cooler regions of the fuel remained virtually free of the fission products. A few stainless steel particles were found that contained unusually high concentrations of fission product palladium. This may be associated with a particular point of origin in the temperature gradient of the fuel.

2. Fabrication and Evaluation

a. Irradiation of Th-U-Pu Alloys. Six Th-U and Th-U-Pu fuel alloys, jacketed in V-20 w/o Ti tubing, are being irradiated in CP-5. The objectives of the experiment are to determine the relative swelling behavior of the fuels, the restraint characteristics of the jacket, and the maximum attainable burnup before jacket failure. Table XXVIII summarizes design details and irradiation variables for these metal fuel experiments.

TABLE XXVIII. Status of Metal-fuel Irradiations in CP-5

Specimen No.	Design Parameters ^a		Operating Conditions			
	Fuel Composition (w/o)	Effective Density (%)	Max kW/ft	Max Clad Temp (°C)	Burnup to Date	
					a/o (U + Pu)	fiss/cc x 10 ⁻²⁰ ^b
1N16	Th-20U	74.7	8.7	590	24.0	11.5
4N19	Th-20U	74.0	8.7	560	24.0	11.5
2N17	Th-10Pu-10U	74.1	9.2	590	27.3	12.8
5N20	Th-10Pu-10U	73.0	9.2	590	27.3	12.8
3N18	Th-10Pu-20U	73.5	8.7	560	16.2	11.8
6N21	Th-10Pu-20U	73.5	8.7	560	16.2	11.8

^aAll specimens are clad in 0.015-in.-thick V-20 w/o Ti alloy, with an OD of 0.198 in.

^bBased on effective density.

b. High-temperature Materials

(i) Studies of Corrosion in Lithium at Elevated Temperatures.

During previous studies of corrosion inhibition in liquid lithium, a silicide coating (Ta₂Si) formed in a Li-1.5 a/o Si solution at 1200°C resulted in corrosion protection of the substrate metal (tantalum) from lithium attack. The production of this coating requires diffusion of silicon or tantalum through the in situ silicide layer. The present study is centered on identification of the principal diffusion element and on the rate of formation of the protective coating. The results of an initial test designed to identify the diffusing element are of interest, although the specific objective of the test was not obtained. The following describes the experimental conditions and results.

In the test, fine tungsten wires were used as markers (since tungsten is chemically inert in lithium at 1200°C). The wires were

spot-welded on a well-polished and degreased tantalum insert coupon. The coupon, lithium, and 1.5 a/o silicon were electron-beam (EB) sealed in a tantalum capsule. The system was then tested in a prepurified helium-filled furnace for four days at 1200°C under isothermal conditions.

During the course of this test, a small leak developed at the EB seal of the capsule. Consequently, the lithium became highly contaminated, as was evident from its dark brown discoloration. Some of the tungsten wires were embrittled and loosened. Since the importance of interstitial contamination in the corrosion of refractory metals in lithium is well-known, it was expected that the corrosion of tantalum would be highly accelerated. However, results of metallographic examination of the tantalum insert coupon revealed that the coupon was well protected by a newly formed layer. The protective layer apparently formed in a contaminated lithium-silicon solution during the course of this test. The nature of this layer has not yet been determined. Study of the Ta-Li-Si system is continuing.

(ii) Irradiations. A series of irradiations is in progress on ceramic fuel materials being developed under the high-temperature materials programs. The materials under irradiation include UC-20 w/o PuC, and PuC and US clad in Nb-1 w/o Zr. The irradiations are being made in instrumented capsules in the MTR. A summary of the irradiations is shown in Table XXIX.

TABLE XXIX. Status of Ceramic-fuel Irradiations in MTR

Capsule No.	Specimen Number	Design Parameters				Operating Conditions		
		Fuel Composition (w/o)	Effective Density (%)	Cladding OD (in.) ^a	Cladding Thickness (in.)	Max Cladding Temp (°C)	Burnup to Date	
							a/o (U + Pu)	fiss/cc x 10 ⁻²⁰ ^b
56-11	MV-2	UC-20 PuC	79	0.281	0.012	470	7.6	19.6
56-8	MV-3	UC-20 PuC	81	0.281	0.012	715	8.1	21.5
56-8	MV-5	UC-20 PuC	80	0.281	0.012	705	7.8	20.4
56-11	MV-6	UC-20 PuC	80	0.281	0.012	480	8.1	21.2
56-13	Z-4	UC-20 PuC	79	0.174	0.015	665	3.0	7.7
56-13	Z-5	UC-20 PuC	79	0.174	0.015	585	3.0	7.7
56-13	Z-7	UC-20 PuC	79	0.174	0.015	570	3.0	7.7
56-13	C-45	PuC	84	0.174	0.009	700	3.0	8.2
56-8	S-7	US	80	0.281	0.012	535	6.4	12.4
56-8	S-8	US	89	0.281	0.012	725	8.4	18.1
56-8	S-9	US	76	0.281	0.012	750	8.4	15.5
56-8	S-10	US	91	0.281	0.012	690	8.4	18.6
56-11	S-15	US	82	0.281	0.012	380	5.5	10.9
56-11	S-16	US	90	0.281	0.012	510	7.4	16.1
56-11	S-17	US	88	0.281	0.012	500	6.1	13.0
56-11	S-18	US	77	0.281	0.012	610	7.9	14.8

^aAll cladding is Nb-1 w/o Zr alloy.

^bBased on effective density.

3. Techniques of Fabrication and Testing

a. Fabrication Development of Tungsten-base Alloys. High-quality tubing of both W-30 Re-30 Mo (a/o) and W-25 w/o Re has been produced from single-extruded tube-blanks by a sequence of primary ductile-core

drawing and secondary-mandrel drawing. The final tube size is 0.250-in. OD by 0.018-0.023-in. wall. Both outer and inner tube surfaces are extremely smooth and free of all extrusion striations, as well as other scratches or irregularities. Processing of either alloy to smaller diameters by mandrel drawing appears, from preliminary results, to be feasible. Efforts in this area will continue.

The fabrication sequence is detailed in Table XXX. For example, the molybdenum jacket is removed from the extruded tube blank (by machining and acid dissolution), after which the blanks, with molybdenum cores intact, are sized and rounded-up by ductile-core drawing. Following an annealing operation the molybdenum cores are removed in preparation for mandrel drawing. A molybdenum nose filler is used to facilitate swage pointing, and the hardened mandrels are made of M-1 or T-1 tool steel, useful up to about 650°C. The tooling arrangement for the mandrel drawing operations is shown in Fig. 28.

TABLE XXX. Fabrication History for W-Re-Mo and W-25 Re Tubing

Operation	Method	Temperature	Reduction	Lubricant	Annealing Conditions	Final Product Size	Product Quality
Extrusion (Pressed and sintered sleeve--1.25-in. OD x 1.0-in. ID)	"Filled-Billet" (Mo core and jacket). Single extrusion.			Zirconia-coated dies--MoS ₂ and graphite in synthetic oil			OD and ID striations resulting from extrusion operation
W-Re-Mo (Ext. No. 637)	As above	1400°C	6.45:1		No postextrusion anneal	~0.485-in. OD x 0.050-in. wall	As above
W-25 Re (Ext. No. 647)	As above	1500°C	As above		As above	As above	As above
Primary--Ductile Core Drawing	Jacket removed, Mo cores intact.			WS ₂ in alcohol (1-2% solids)			OD improved; ID striated.
W-Re-Mo	As above	600°C	2 passes ~13-18% area reduction	As above	1500°C for 1 hr (vacuum) after 2 passes	~0.450-in. OD x 0.043-in. wall	As above
W-25 Re	As above	As above	As above	As above	As above	~0.450-in. OD x 0.046-in. wall	As above
Secondary--Mandrel Drawing	Cores removed, Mo nose filler, hardened mandrel.			WS ₂ in alcohol (1-2% solids)			Very good surfaces on OD and ID. NDT tested; no defects over 10% of wall thickness.
W-Re-Mo	As above	400°C	10 passes ~77% area reduction	As above	1500°C for 1 hr prior to drawing and after each pass	~0.250-in. OD x 0.017-in. wall	As above
W-25 Re	As above	600°C	11 passes ~66% area reduction	As above	As above	~0.250-in. OD x 0.023-in. wall	As above

Fig. 28
Mandrel Drawing with Ductile Nose Filler

Two-foot finished tube lengths are being processed with the present resistance tube-heating furnace. Induction tube heating is under development. This means of heating will permit the processing of longer lengths of finished tubes. Mandrel extraction appears to be no problem, and no special chilling techniques have been found necessary. An annealing operation has followed each mandrel draw and has lowered the as-drawn tube hardness from $\sim R_C-51$ to R_C-43 . Further processing economies may result from the successful elimination of some of the mandrel drawing and annealing cycles; this remains to be established. Alternate means of processing the starting tube blank are being investigated. Elimination of the "filled billet" extrusion operation, and hence of its associated problems of core removal and of surface striations, should effect further economies in the processing of these materials.

b. Ultrasonic Instrument and Transducer Development. Two ultrasonic transducer probes of Clevite's lead zirconate-titanate were fabricated of sintered tungsten--62 and 52% dense, respectively. The porous tungsten samples of 52% theoretical density and of two transducer probes with tungsten backings have been evaluated. The two probes have PZT-4 transducer elements and backings with densities of 52 and 62%, respectively.

The ultrasonic longitudinal velocity in the 52%-dense samples was measured with a through-transmission technique (separate transmit and receive probes), and the resultant acoustic impedance (Z) was calculated. This completes the data on -325 mesh, porous tungsten in the density range from 52 to 80%. A graph of longitudinal velocity and acoustic impedance as a function of theoretical density shows that -325 mesh tungsten in the 52 to 80%-density range has acoustic impedance values from about 24 to $58 \times 10^5 \text{ g cm}^{-2}\text{sec}^{-1}$. Samples with these impedances should be particularly useful in matching ceramic transducer elements, which have " Z " values in the range of 20 to $34 \times 10^5 \text{ g cm}^{-2}\text{sec}^{-1}$.

The performance of the probes with commercial ultrasonic equipment was observed on a Sperry UM-721 Reflectoscope. A contact technique and aluminum reference blocks were utilized in the test. With the receiver sensitivity set to permit the visualization of five back reflections (medium sensitivity), an initial pulse length of $15 \mu\text{s}$ was obtained in both probes. Apparently, ultrasonic energy from the back face of the transducer element is penetrating the backing material and reinforcing the energy from the front face. Means of reducing this energy from the back face, such as the introduction of a conical rear face on the backing member, will be studied further.

c. Development of a Neutron-image Intensification System. A new video amplifier is being tested for use with the vidicon camera of the neutron television system. The amplifier was designed and built by W. K. Brookshier, Electronics Division. Bench tests with the prototype model

showed that one could vary the gain of a portion of, or all of, the video signal and that the level at which the variable gain was introduced could be varied. Gain factors of the order of 10 appeared readily usable. Gains of the order of 100 times can be obtained, but it will require actual television tests to determine whether or not the noise becomes so great as to render the television picture unusable at this higher gain setting.

Television tests with this amplifier are just getting under way. It is hoped that the variable gain and level features of the system will permit observers to detect better contrast sensitivity in neutron-television examinations of material such as uranium.

d. Determination of Elastic Moduli of High-temperature Materials by Ultrasonics. The required length of metal samples for ultrasonic measurement at high temperatures can be reduced by a factor of six if the sample is mechanically bonded to a buffer rod by means of a screw coupling. The technique has been applied to ASTM 6061 aluminum and stainless steel. These tests were completed in May, as well as tests on cold-rolled carbon steel. Three-inch samples of both Type 304 stainless steel and of cold-rolled steel were fabricated from the same stock as the respective buffer rods. A low-temperature solder was used to bond the transducer to the Type 304 stainless steel buffer rod. This type of bond greatly increased the amplitude obtained with the commonly used Nonaq couplant. This bond also prevents any impedance change due to transducer movement.

Transit times of shear and longitudinal velocities have been measured for the cold-rolled carbon steel to 800°C and the stainless steel to 1000°C. Calculations of the ultrasonic velocities and elastic moduli for these two materials are now being made.

C. Engineering Development

1. Development of Master-Slave Manipulator Systems

a. Electric Master-Slave Manipulator, Mark E4A. The new synchro gearing (see Progress Report for December 1966, ANL-7286, p. 48) has been installed in the "X," "Y," and "Z" motions. With this gear ratio, the positional nulls are further apart--25 in., measured at the wrist joint, instead of 10 in.--and inadvertent "jumping" should be eliminated. However, before the ratio can be adequately tested, additional gain must be provided in the amplifiers. This is scheduled to be done next month.

All parts for the slave-motion brakes have been received. The brakes have been installed on the "Y" and "Z" motions. The others will be installed when personnel are not needed on more pressing work.

b. Low-inertia Servo-Motor for Manipulators. Fabrication has started on the special 2-pole, 2-phase low-inertia motor. Two such motors are being built to test the motor characteristics and to test the motor in a force-reflecting servo. Different rotor designs may also be tested. A gear drive for testing the motor in a servo system has been designed.

2. Heat Transfer and Fluid Flow

a. Niobium-1% Zirconium Loop. A facility has been built to investigate the heat transfer and two-phase flow characteristics of boiling sodium to a temperature of 2100°F and a pressure of approximately 8 atm. Among the variables to be investigated are boiling heat flux and temperature difference up to the occurrence of the critical flux, boiling and adiabatic two-phase pressure losses, vapor volume fraction, boiling stability parameters, and ultimately the transient behavior of these same quantities.

Sodium temperatures ranging up to 1200°F and heat fluxes up to 100,000 Btu/hr/ft² have been obtained. Flowmeter calibrations for sodium temperatures ranging up to 1200°F have been completed. Additional instrumentation has been added and the thermal-radiation heater was rebuilt. The lower insulator on the heater hung up on the Nb-1% Zr flow tube and caused some bending of the tungsten heating elements. The clearance between the insulator and the flow tube was increased to eliminate this difficulty.

b. Heater Experiments

(i) Electron-bombardment Heater (EBH) Experiment. The original tests have been terminated following satisfactory operation; further EBH tests will be conducted in the new EBH test facility that is under construction. This facility will enable the testing of heaters to their maximum heat flux capabilities prior to use in the Nb-1% Zr loop.

c. Heat Transfer in Double-pipe Heat Exchangers

(i) Countercurrent Turbulent Liquid-metal Flow. The third heat-exchanger test section has been installed in the loop. Cleanup operation, in progress now, will be followed by performance of experiments.

3. Engineering Mechanics

a. Core Structural Dynamics. Tests of a single simulated fuel element continued (see Progress Report for April 1967, ANL-7329). The pump-induced vibration of the test section has been reduced approximately 50% by the use of flexible piping and damping material at support points. Displacements of the rod at water flow velocities of 0, 2, 4, 6, 8, and 10 ft/sec have been measured. The zero flow is obtained by allowing water

flow through the bypass with the test section full of static water at the usual static test pressure (15 psig); this allows measuring the rod response to pump-induced vibration only. Two mutually perpendicular transducers provided a complete representation of the rod displacement at axial positions of $1/4$ and $1/2$ the free length of the rod.

The data suggest several significant phenomena. Rod motion having a pattern usually described as first mode occurs at all flows; at zero flow the pump vibration still provides enough energy to cause detectable motion. Significant amplitude increase occurs with increasing flow velocity. At flow velocities in excess of 6 ft/sec, the rate of amplitude increase, contributed from the frequency band centered at the first-mode natural frequency, decreases until at 10 ft/sec no increase over the test at 8 ft/sec occurs. If the turbulence energy increases with flow velocity, this could indicate an increase in damping with flow velocity, with the result that the resonant peak in the harmonic response function for the rod decreases. Were this the case, higher velocity flow could cause a further decrease in the contribution of the resonant frequency band to the mean-square displacement. The present loop geometry and pump are limited to flow velocities of less than 11 ft/sec. Loop modification or tests in a different loop will permit study of the effect of flow at higher velocities. The present loop can provide higher flow velocities with a test section of smaller inside diameter.

The reduction of rate of increase of first-mode amplitude is accompanied by significant motion at other frequencies. These can correspond to various mode shapes and harmonics, and are being investigated in more detail.

The large-amplitude motion in both directions requires that future experimental work be performed with specimens with a greatly preferred axis of motion or that a vector-type representation of rod motion be used. Analog equipment to provide the vector representation is being investigated.

b. Anisotropy Studies. The analytical solution and associated numerical computations for a sample thermoelasticity problem concerning the stress state in heat-generating anisotropic cylinders have been essentially completed. This work is an extension to the case of transverse isotropy of an isotropic solution investigated previously⁸ and is the last in a series of preliminary studies whose primary aim was to indicate the degree of numerical difficulty associated with the introduction of anisotropic material effects in elasticity problems of the type likely to occur in reactor applications.

⁸Valentin, R. A., On the Steady-state Thermal Stresses in Cylindrical Fuel Pins due to Axial Variations in Their Internal Heat Generation Rate, ANL-7263 (Dec 1966).

All work completed to date in the study of anisotropic effects has been based on steady-state analysis. Since several of the anticipated applications of these studies will be related to time-dependent phenomena, an effort has been initiated to analyse certain relevant properties of the transient thermal-stress field produced during rapid initiation of internal heat generation. The first such transient (quasi-static) problem being considered has been formulated for isotropic material and is intended to reveal the time required to achieve maximum stress levels, and thereby determine the relation between the temperature at the instant of peak stress and the final steady-state temperature. This information is needed to determine how reasonable it is to assume temperature-independent material properties when treating transient stress phenomena. Much of the analytical work relating to the solution of the chosen problem has been completed; numerical evaluations are being initiated. Preliminary computations have indicated that there may be considerable difficulty associated with these evaluations (particularly for short times); it is not yet clear how much effort will be required to produce acceptable accuracy in evaluating the transient stress field.

D. Chemistry and Chemical Separations

1. Fluoride Volatility Processes

a. Recovery of Uranium and Plutonium from Low-enrichment Fuels: Laboratory Support Work

(i) Chemistry of Ruthenium Fluorides. A significant effort has been started to determine the identity, rate of formation, and reactions of the ruthenium fluoride species formed in the fluid-bed fluoride volatility process. From the data to be obtained, the distribution of ruthenium in the various steps (oxidation, fluorination with BrF_5 , and fluorination with fluorine) are to be determined. RuF_5 , RuF_6 , and RuOF_4 , one or more of which are probably formed in this process, will be studied.

Two runs have been performed in new apparatus scaled for the preparation of 2- to 5-g batches of ruthenium hexafluoride by the reaction of ruthenium metal with fluorine. The first run at 400°C produced mainly RuF_5 ; in the second run, at 550 to 600°C , RuF_6 containing CF_4 and ClF_3 impurities was produced. Future work with RuF_6 will include (1) purification of the RuF_6 , (2) determination of the sorption properties of RuF_6 on NaF , (3) determination of the thermal stability of RuF_6 , and (4) determination of its spectrum in the ultraviolet, visible, and near infrared regions.

An investigation has begun on the kinetics of the reactions of BrF_3 and BrF_5 with ruthenium metal and RuO_2 under reaction conditions simulating, insofar as possible, those currently used in laboratory-scale

process-development studies. In addition to measurements of the rates of reaction, information will be sought on the species of ruthenium fluorides formed in these reactions. Three preliminary experiments were performed in which ~0.5-g samples of ruthenium metal tagged with Ru^{106} were reacted with BrF_3 in a horizontal tubular reactor; the fluorination rates were monitored during the experiments by measuring the change in activity of the metal sample (in a nickel boat) and of a 400°C NaF trap downstream from the metal sample. It was found (1) that the overall reaction rate can be determined by this technique, (2) that a gas-phase temperature of at least 250°C is necessary to ensure effective volatilization of the product (probably RuF_5), (3) that RuF_5 , which is apparently formed first, reacts with metal to form RuF_3 which, in turn, reacts with BrF_3 to form RuF_5 , (4) that the rate of reaction between ruthenium metal and BrF_3 is considerably lower than the rate of reaction between ruthenium and fluorine, and (5) that all of the ruthenium fluoride species that reach the NaF trap are absorbed on the NaF at 400°C.

(ii) Fluorination of UO_2 - PuO_2 -Fission Product Pellets. A 2-in.-dia fluid-bed reactor is being used for development studies to establish optimum conditions for fluorinating UO_2 - PuO_2 pellets containing fission products. In the process, the pellets are first oxidized to U_3O_8 - PuO_2 fines by reaction with oxygen. Next, the uranium oxides in uranium-plutonium mixtures are converted to volatile UF_6 by BrF_5 , which converts plutonium dioxide to nonvolatile PuF_4 . In a subsequent step, plutonium is recovered as volatile PuF_6 by the reaction of PuF_4 with fluorine.

Two experiments (Purse-18 and Purse-19) were performed recently to determine whether plutonium holdup in the fines would be decreased by utilizing an alumina bed consisting of 10% -100 mesh and 90% 48-100 mesh alumina instead of 100% 48-100 mesh alumina normally used. It was anticipated that alumina fines would provide a precoat on the upper reactor surfaces before plutonium-bearing fines would collect there and would also provide a vehicle for the return of plutonium fines upon blow-back of the filter. The alumina mixture used was chosen on the basis of results of plutonium retention and fines elutriation tests in a mockup of the 2-in.-dia reactor.

In each of the runs, about 650 g of UO_2 - PuO_2 -fission product oxide pellets together with 0.6 g CsF , 0.15 g RbF , and 0.5 g NpO_2 were added to 1100 g of alumina containing 10% fines. The processing sequence consisted of: oxidation of the pellets with 20 v/o oxygen at 450°C for 4 hr, fluorination with 11 v/o BrF_5 at 300°C for 2 hr, and fluorination with 90 v/o fluorine for 3 hr at 300°C, for 5 hr while the reactor temperature was increased from 300 to 550°C at 10°C per 12 min, and for 3 hr at 550°C. The major difference between the two runs was an extra 6 hr of fluidization between the oxidation step and BrF_5 step of run Purse-18, while a temperature controller was being repaired.

Uranium, plutonium, and neptunium concentrations in the final alumina beds were comparable to the levels usually obtained with the 48-100 mesh alumina beds. The reactor dismantled after each run and the fines were removed from interior surfaces and analyzed. Concentrations of plutonium found in the fines recovered by rapping and brushing the disengaging and filter chambers were 0.005 w/o for run Purse-18 and 0.065 w/o for run Purse-19. However, the plutonium concentrations in the fines recovered by cleaning the filters and the uppermost parts of the filter chamber differed widely for the two runs--0.0003 w/o for run Purse-18 and 1.8 w/o for run Purse-19, the latter representing about 6 percent of the plutonium charged. After each of these two runs, an agglomerate of highly fluorinated alumina, weighing about 63 g, was found in the enlargement section between the 2-in.-dia reactor and the 3-in.-dia disengaging chamber. Conclusions from these experiments will await the results of an additional run. Prior to the next run, the alumina bed will be fluorinated in an attempt to prevent the formation of an agglomerate so that the alumina fines added to the bed may fulfill the intended purpose.

b. Recovery of Uranium and Plutonium from Low-enrichment Fuels: Engineering Work

(i) Engineering-scale Alpha Facility. The major objectives of experiments conducted in the engineering-scale alpha facility are to demonstrate the handling of PuF_6 on an engineering scale and to determine the feasibility of fluoride volatility flowsheets. The process equipment includes a fluid-bed fluorinator for fluorinating mixed oxide fuel to UF_6 and PuF_6 , a converter reactor for converting PuF_6 to PuF_4 by thermal decomposition, and a system of condensers and chemical traps for collecting hexafluoride products.

Two experiments (runs Pu-14 and Pu-15) were performed to determine the recovery of plutonium as the hexafluoride from bed material generated in earlier laboratory-scale experiments (see Progress Reports for December 1966, ANL-7286, p. 55, and January 1967, ANL-7302, pp. 76-77). The bed material represented the fluid bed after the uranium had been recovered by fluorination of oxidized UO_2 - PuO_2 -fission product pellets with BrF_5 . In runs Pu-14 and Pu-15, 6 kg of this material was contacted with 80 v/o fluorine at 300-550°C. The plutonium concentrations in the bed grab samples leveled off at an average plutonium concentration of 0.013 w/o after only 10 hr of fluorination. This level represents a plutonium removal of about 98.5%. Evaluation of equipment performance in these two runs has been continued.

The efficiency of the PuF_6 cold-trap system (two traps in series, chilled trichloroethylene loop) in recovering about 25 g of plutonium hexafluoride in each run was determined. Collections of PuF_6 in the cold

traps for the two experiments were 90% for run Pu-14 and 96% for run Pu-15. The 6% difference in PuF_6 collection may reflect the higher gas velocity and slightly higher coolant temperatures present in run Pu-14. Apparently, all of the plutonium loss through the cold traps can be accounted for from vapor pressure considerations.

Additional information was obtained in these experiments on the sorption efficiency of NaF for trapping PuF_6 . Downstream from the cold traps, a $2\frac{7}{8}$ -in.-ID by 24-in.-tall trap operated at about 100°C removed the remaining PuF_6 in the first 8-in.-long section of NaF. Traps of 14-in. height will be tried in future applications.

Information has now been obtained on ruthenium distribution in runs Pu-1 through Pu-5 (see Progress Reports for November 1965, ANL-7122, pp. 63-65, through May 1966, ANL-7219, pp. 51-55). Runs Pu-1 through Pu-3 were two-zone oxidation-fluorination runs with UO_2 - PuO_2 -fission product oxide pellets which involved the formation of a mixture of UF_6 and PuF_6 , which was condensed in cold traps. Subsequently, plutonium was separated from the mixture by thermal decomposition of PuF_6 to solid PuF_4 at about 300 - 350°C . Runs Pu-4 and Pu-5 were cleanup fluorination runs in which fluorine was cycled through the equipment to remove plutonium deposits indicated by a portable neutron survey meter.

Overall results based on a 50% accountability of ruthenium are as follows:

- (a) Relatively large amounts of ruthenium were found with the UF_6 which passed through the thermal decomposer.
- (b) A somewhat smaller amount than in (a) was recovered from the cold traps during the cleanup-fluorination runs.
- (c) A measurable amount of ruthenium (perhaps 5% of that in the charge) was found in the final alumina bed of the fluorinator.
- (d) Detectable amounts were found in the activated alumina and NaF traps used in the processing of the off-gas downstream of the cold traps.
- (e) Within the current limits of analytical techniques (perhaps down to levels equivalent to 4% of that charged), no ruthenium was detected in the final thermal decomposer PuF_4 bed or in the UF_6 - PuF_6 which remained unfed in the thermal decomposer feed vessels.

IV. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Direct Conversion

1. Liquid-metal Linear Generator

a. Condensing Injector. Construction of an analytical model for condensing-injector flow processes is proceeding satisfactorily. A stratified-flow model, which includes the possibility of weak shocks in the vapor flow as suggested by the experimental data, adequately predicts the axial-pressure profiles as far as the assumption of well-defined stratified flow is experimentally verified. The effects of variations in liquid-to-vapor flow ratio and inlet liquid subcooling compare well with experimental results. An attempt is being made to simulate the final pressure rise which is diminished by the incomplete collapse of the vapor phase. This part of the analytical effort should indicate what similarities exist between the observed phenomena and one- or two-phase normal shocks, and the influence of vapor condensation on the pressure rise.

b. Pressure Recovery of a Two-phase One-component Flow through a Diffuser. Experiments have been completed on two-phase flow diffusers having an included angle of 12.5 and 15°. For void fractions larger than 20%, the efficiency of these diffusers was slightly better than the efficiency of diffusers having an included angle of 5°. For optimum pressure recovery by a constant area ratio of the diffuser, the angle of divergence probably has to be larger than 15° for void fractions greater than 20%.

A test section for investigating the performance of a supersonic flow diffuser has been designed; preparations are being made for measurements on this diffuser.

B. Argonne Advanced Research Reactor (AARR)

1. General

Specifications for Construction Package No. 1 for rough grading and excavation have been sent to prospective bidders. Proposals are due in June.

Preliminary specifications for the reactor-building foundation mat and for the containment liner are being reviewed. Burns & Roe has prepared an equipment list which will be used as a basis for preparing a list of Schedule-X procurements.

The specifications for the semipermanent and removable beryllium pieces have been sent to prospective bidders; proposals are expected in June. The proposal date for the permanent beryllium has been extended to

June 9. The date for receipt of proposals for primary heat exchanger has been extended to June 14 at the request of the vendors.

About 30% of the Title-II design has been completed. Design activity has increased on the reactor and associated buildings in Construction Package No. 5.

2. Reactor Physics

a. Thermal-neutron Flux in the Instrument-tube Zone. The shielding program MAC⁹ was used for previous calculations of the thermal-neutron flux in an instrument-tube zone outside the reactor pressure vessel,¹⁰ in the radial direction of the two through-tubes. The calculations were performed with the use of a spherical-geometry model of the core and reflector zones, because of the limitation of MAC to one-dimensional geometries. The original version of MAC was limited to slab geometry; the sphere and cylinder options for the neutron flux were added here.

It has been discovered that the previous results are in error because of a numerical instability in the calculation when a thin water gap is at the outer surface of the pressure vessel. In the sphere calculations, the computed thermal-neutron flux was amplified (erroneously) to approximately ten times its level in the instrument-tube zone. With the identical region dimensions and compositions and region partitionings, slab calculations were made, indicating erroneously that the thermal-neutron flux was amplified to twice its level when the thin water gap was introduced. Although fine meshes were used in the spatial subdivisions of the water gap and of the two regions flanking it, it is conceivable that a still finer mesh is required to minimize the error. This will be investigated.

Calculations with the transport-theory program SNARG (SN-Argonne) for the vicinity of the water gap have confirmed that the thin water gap should change the thermal-neutron flux by less than 10%.

Two other methods of analysis are being applied to the resolution of this problem: (1) computations with the program NRN¹¹ and (2) computations of the multiregion reactor with a transport-theory program using higher-order anisotropic-scattering approximations to the scattering cross sections of the various nuclides. The latter calculations are being performed by Oak Ridge National Laboratory.

⁹Peterson, E. G., "MAC--A Bulk Shielding Code," HW-73381 (April 1962).

¹⁰Reactor Development Program Progress Report for August 1966, ANL-7249, p. 72, Fig. 27. [Note: The ordinate label for Fig. 27 should read: "MAC Group-31 (Thermal) Neutron Flux."]

¹¹Mittelman, P. S., "Advances in Power-Reactor Shield Analysis," Power Reactor Technol., Vol. 9, No. 1, Winter 1965-1966, pp. 19-24.

The best estimate now available for the radial-average thermal-neutron flux in that instrument-tube zone is 2×10^9 n/cm²/sec in the case of a 3.5-in.-thick Inconel pressure vessel. This flux is uncorrected for the perturbation effects of the neutron detectors. At present, the minimum flux sought is 2×10^9 (perturbed) at 100 MW.

b. Reactivity-control Worths of Soluble Neutron Poisons. Calculations by multigroup diffusion theory of the reactivity-control worths of soluble neutron poisons have been completed. Two poisons were studied: boron and cadmium. In each case, the calculation was made for the fresh reactor, with control rods withdrawn and with the target tubes in the Internal Thermal Column (ITC) fully loaded.

Figure 29 displays the reactivity-control worth as a function of the concentration of the natural element (boron or cadmium) in the reactor water, which is assumed to be uniformly poisoned. Figure 30 provides details of the contribution of the poisoning by cadmium from each of the various radial reactor regions, as obtained directly from multigroup calculations, not from perturbation-theory analysis.

Fig. 29. Reactivity-control Worths of Soluble Poisons in the AARR (Fully Loaded ITC)

Fig. 30. Reactivity-control Worths of Cadmium as Soluble Poison in: (1) ITC; (2) Fuel Zone; or (3) Radial Reflector.

Cross-section sets appropriate to the particular levels of neutron poisoning were calculated and used in these analyses as a higher-order correction. However, all the calculations were performed in one-dimensional (cylinder) geometry, and the additional reactivity losses resulting from the poisoning of the axial reflectors are not included. Therefore, Figs. 29 and 30 should somewhat underestimate the reactivity-control worths of the soluble neutron poisons.

The reference system of soluble neutron poison for the AARR utilizes cadmium in the form of a highly concentrated solution of cadmium

nitrate. This system provides a "backup" control capability. The poison may be injected into the primary coolant system while the reactor is at pressure, but only after manual operation of valves. This is not a "scram"-type system of reactivity control.

c. Heating Rates in the Beryllium Reflector. The shielding program MAC has been used to determine gamma-ray fluxes in the radial beryllium reflector during reactor operation at full power (100 MW). From these fluxes, heating rates in the beryllium were inferred. The results are summarized in Fig. 31. Since MAC calculates gamma-ray fluxes only in slab geometry, the output was corrected to account for the finite height of the reactor core.

Fig. 31. Axially Averaged Heating Rates in the Beryllium Reflector at 100 MW

In these calculations, the power distribution in the fuel was taken to be constant at the average level. The heating rate in the beryllium near the fuel zone varies axially because of two effects: (1) the typical gross axial variation of power in an unflattened system (a cosine-type gross distribution) and (2) an accentuation of the power distribution in the vicinity of the core midheight because the control rods typically are vertically antisymmetrical, with the black portions of all control rods ending at the same distance from core midheight. The combination of these two-effects is estimated to increase the computed averaged heating rate in the beryllium near the fuel by roughly 40% at core midheight. This effect becomes smaller with increasing distance from the core, but it would be prudent to apply a factor of 1.4 to the data of Fig. 31 to estimate the maximum axial heating rate at each radial position. The axially integrated heating rate in the beryllium at each radial

position would be given approximately as the product of the data of Fig. 31 and the total mass of beryllium in the radial reflector per unit plan area.

These calculations apply to heating rates in the sector containing the two through-tubes. The void, aluminum, and water associated with the through tubes were volume-averaged with the other materials of the radial reflector over the 20-in.-high zone corresponding to the axial boundaries of the active fuel zone.

3. Instrumentation and Control

a. Flow-coastdown Dynamics. To provide preliminary design requirements for safety instrumentation needed to prevent the hot-spot heat flux from rising high enough to cause more than local boiling of the coolant, coastdown characteristics for the AARR coolant have been evaluated for the case in which utility power for the main pumps fails (see Progress Report for March 1967, ANL-7317, pp. 94-100). The response of the reactor core to this flow coastdown has been calculated, and a rough analysis has been made that indicates that the flow-sensing delay in the flux-to-flow-ratio safety channel has a serious effect (see Progress Report for April 1967, ANL-7329). This analysis was based on a rough approximation to the channel-response characteristics. Since on this basis the flow-sensing delay appears serious, a more precise analysis of channel response time has been made.

(i) Flow-sensing Delay Time. Primary emphasis in this analysis was on showing the influence on flow-sensing delay time of the length and diameter of the lines used to connect the differential-pressure cell to the primary-flow venturi. Accordingly, the cell was characterized very approximately by assuming that its influence on the system response is that of a pure "spring constant" (i.e., it provides no damping of the fluid motion in the long lines). If this is the case, then damping is primarily that due to flow friction losses in the lines themselves; entrance and elbow losses are found to be insignificant for the line lengths under consideration. Under conditions of interest, the system was found to be describable as a second-order linear system. Expressions are given for the undamped natural frequency and damping ratio of this linear system in terms of the differential-pressure-cell spring constant and the length and diameter of the lines. The responses of a second-order linear system to the flow coastdown determined previously are given for values of natural frequency and damping in the range of interest.

For the system in Fig. 32, it was found that if $L/d > 250$ and if $\Delta \dot{p}_m < [\Delta \dot{p}_m]_c$ then the system response is given by

$$\frac{1}{\omega_0^2} \Delta \ddot{p}_m + \frac{2\zeta}{\omega_0} \Delta \dot{p}_m + \Delta p_m = \Delta p, \quad (1)$$

in which the undamped natural frequency ω_0 is given by

$$\omega_0 = \frac{d}{2} \left(\frac{\pi \frac{dp_m}{dV}}{2\rho L} \right)^{1/2}$$

and the damping ratio ζ is given by

$$\zeta = \left(\frac{32\mu}{d^3} \frac{2L}{\pi\rho \frac{dp_m}{dV}} \right)^{1/2},$$

where L is the length of lines (assumed equal) from the differential-pressure cell to the venturi in feet, d is the inside diameter of the lines

Fig. 32. Physical Flow-sensing System Considered

from the venturi to the cell in feet, Δp_m is the pressure drop measured between chambers of the cell in lb_f/ft^2 , $[\Delta p_m]_c$ is the rate of change of measured pressure at which critical velocity (above which flow may become turbulent) occurs in the lines in $\text{lb}_f/\text{ft}^2/\text{sec}$, ω_0 is the undamped natural frequency in radians/sec, ζ is the dimensionless damping ratio (ratio of damping coefficient to that required for critical damping in a second-order linear differential equation), Δp is the actual venturi pressure drop in lb_f/ft^2 , dp_m/dV is the ratio of change of diaphragm Δp to change in chamber volume on one side of the diaphragm (thus is the "spring constant" of the cell) in $\text{lb}_f/\text{ft}^2/\text{ft}^3$, ρ is the density in slugs/ft^3 (1.94 at 80°F and 600-psi water), and μ is the dynamic viscosity in $\text{lb}_f/\text{ft}^2/\text{sec}$ (1.80×10^{-5} in 80°F 600-psi water).

Representative values of the above quantities for 1/8-in.-ID lines, 20 ft long with a differential-pressure-cell volumetric displacement

of 0.1 in.³, are: $L/D = 1920$, $[\Delta p_m]_c = 3415 \text{ lb}_f/\text{ft}^2/\text{sec}$, $\omega_0 = 4.98 \text{ radians/sec}$, and $\zeta = 0.276$.

To predict the response of the flow-sensing channel during coastdown, it was necessary to describe flow as a function of time for the flow-time curve determined previously (see Progress Report for March 1967, ANL-7317, pp. 94-100). The exponential function gives a good fit, and for this reason the approximate input function used in this analysis was $Q_A/Q_0 \approx e^{-\alpha t}$ with $\alpha = 0.415 \text{ sec}^{-1}$, where Q_A is the actual primary flow rate in gal/min, Q_0 is the initial value of the primary flow rate in gal/min, α is the exponential decay rate of Q_A with time during flow coastdown.

Fig. 33. Flow-measurement Channel Response to Forcing Function.

values of ω_0 and ζ are given in Fig. 33. These are as calculated from the analytical solution of Eq. (1) for this input, which is

$$\frac{\Delta p_m}{\Delta p_0}(t) = Ae^{-2\alpha t} + 2e^{-\zeta\omega_0 t} \left(B_1 \cos \sqrt{1 - \zeta^2}\omega_0 t + B_2 \sin \sqrt{1 - \zeta^2}\omega_0 t \right), \quad (2)$$

where the dimensionless constants A , B_1 , and B_2 are functions of ω_0 , α , and ζ .

As can be seen from Eq. (2), the response is a damped sine wave superimposed on an exponential function. This exponential function is given by the dimensionless constant A times the input function. Thus, after the sinusoidal term has decayed, the pressure response approaches

$$\frac{\Delta p_m}{\Delta p_0}(t) \approx Ae^{-2\alpha t} = A \frac{\Delta p}{\Delta p_0}. \quad (3)$$

A comparison of the approximation to the actual Q_A/Q_0 is shown in Fig. 33. The solid line is the function $e^{-\alpha t}$ and the plotted points are the actual values of Q_A/Q_0 . Since Δp varies as Q_A^2 , the actual forcing function in Eq. (1) is then

$$\Delta p = \Delta p_0 e^{-2\alpha t} = \Delta p_0 e^{-0.83t},$$

where Δp_0 is the initial venturi pressure drop in lb_f/ft^2 .

Responses of the system to this forcing function for indicated

If we define a constant τ such that $A = e^{2\alpha\tau}$, then Eq. (3) can be expressed as

$$\frac{\Delta p_m}{\Delta p_0}(t) \approx (e^{+2\alpha\tau})(e^{-2\alpha t}) = e^{-2\alpha(t-\tau)} = \frac{\Delta p}{\Delta p_0}(t-\tau).$$

Thus the pressure response of the system approaches a function which is simply the true venturi pressure drop delayed by τ sec, where τ is given by $\tau = (1/2\alpha) \ln A$.

Since Q_m/Q_0 is $(\Delta p_m/\Delta p_0)^{1/2}$, then, from Eq. (3), Q_m/Q_0 may be expressed as $Q_m/Q_0 = \sqrt{Ae^{-\alpha t}} = (e^{\alpha\tau})(e^{-\alpha t}) = e^{-\alpha(t-\tau)}$. Hence, Q_m lags Q_A by τ sec also. (Q_m is the measured primary flow rate in gal/min.) Power drops off due to temperature feedback during the coastdown transient; it is very nearly linear with time and can be expressed as $P/P_0 = 1 - ct$, where P is reactor power in MW, P_0 is the initial reactor power, and c is the slope of the P/P_0 versus time curve during coastdown and is 0.12 sec^{-1} . This P/P_0 ratio is plotted in Fig. 33.

In summary, the relations developed here form the basis for preliminarily investigating possible scram-channel designs and indicate, in particular, how the length and diameter of the instrument lines affect the response. The signal delay time derived here includes only that involved in transmission of the venturi pressure to the differential-pressure cell. Additional delays arise in the generation of the electrical-output signal, extraction of the square root, etc.

4. Component Development

a. Sample-irradiation Device. To evaluate the temperature conditions of the vertical experimental facilities (located in the ITC and in the reflector) during reactor conditions of steady state, postulated accident, and shutdown, heat-transfer analyses are being made. A computer program is being developed for the CDC-3600 to calculate the temperature behavior of the structure within the core region that contains the material to be irradiated and the flowing coolant.

In the present model, the structure referred to as the "rabbit tube" is a vertical tube with a thin wall. The material to be irradiated is considered to be a solid cylinder, called a "rabbit," located concentrically within the rabbit tube so that there is an annular space for coolant flow.

The assurance of safe operation of rabbit facilities under the usual class of postulated accident conditions requires a computer program

that relates the temperature of the structure and rabbit with time. For example, while the reactor is operating, the rabbit is inserted into the core position by the coolant flowing around the rabbit; the rabbit is removed from the core position by reversing the flow of rabbit coolant. However, the rabbit might be in the ITC when main electrical power fails, in which case the reactor scrams and its power drops immediately by a factor of 16--but power generation in the rabbit decreases only ~50% and the rabbit coolant flow rate decreases to ~10% of normal flow. To learn whether the rabbit would be heated enough to form a steam void in the coolant, which could lead to reactivity effects and possibly to burnout conditions, detailed analyses are being made.

Fig. 34. Comparison of Calculated Transient Temperatures (four lower curves) of a Stationary Rabbit within ITC after Pump Power Failure Shows Good Agreement between Two Codes. Two upper curves show assumed drops in heat generation and coolant flow after loss of pumping power.

Preliminary calculations were made with the CHIC-KIN code to estimate the heat-transfer conditions of the rabbit after the loss of main electrical power. These calculations assumed: an inlet coolant water temperature of 120°F, water flow of 6.0 gal/min in the rabbit tube and 4.3 gal/min past the outside of the rabbit tube, a 0.5-in.-dia aluminum rabbit, an aluminum rabbit tube that is of 0.625-in. ID and of 0.75-in. OD, and that the normalized flow rate and power of the rabbit drop after electricity fails as shown in the upper curves of Fig. 34.

To use the CHIC-KIN code, some approximations were required: (a) the rabbit tube itself could not be accounted for, (b) the ITC primary water and the tube water could not be separately accounted for, and (c) the rabbit had to extend the full 27.56-in. length of the heated zone, which would be longer than the 20 in. an actual rabbit might be.

The new computer program (TRANTEMP) being written for the heat-transfer analysis of the hydraulic rabbit eliminates the approximation deficiencies of CHIC-KIN and, in addition, will be used for heat-transfer analysis of the flow-reversal and rabbit-ejection process [the CHIC-KIN code is written for analyzing a stationary plate (or rod) and cannot be used to analyze a moving rabbit].

Figure 34, which compares the preliminary TRANTEMP calculations of the temperature history of the stationary rabbit with the CHIC-KIN results, shows good agreement between the two codes. TRANTEMP calculates lower temperatures because it does not employ the three approximations necessary with the CHIC-KIN code.

V. NUCLEAR SAFETY

A. Accident Analysis

1. Linking of the Heat-transfer, Two-phase Coolant Flow and Point Kinetics Modules

The linking of the improved heat-transfer and the two-phase flow-coolant ejection modules has been checked out (see Progress Report for December 1966, ANL-7286, p. 75, and for January 1967, ANL-7302, p. 85).

A modified version of the point kinetics formulation of Kaganove¹² has been programmed. The modifications were necessary to take into account:

- a. the difference between the time increments used for point kinetics and those used for heat transfer;
- b. the nature of the transient feedback reactivities associated with heat transfer to voiding sodium. In Kaganove's treatment, the feedback reactivities are assumed to be specified as explicit functions of the neutron flux.

The linking of the point kinetics module with the combined heat transfer-coolant flow modules is essentially complete. The feedbacks used in the test problems solved to date are:

- a. the Doppler reactivity specified as a function (for example, logarithmic) of the regional averaged fuel temperatures;
- b. the coolant-void reactivity specified as functions of (for example, directly proportional to) regionally averaged void fractions.

It is noted that the present link-up considers only one representative fuel cell, namely, the average cell. Work is in progress to incorporate several fuel cells to obtain a better representation of the effects of the radial profile of the neutron flux in the core.

B. Coolant Dynamics

1. Superheat

To determine the amount of liquid superheat required to initiate nucleate boiling in sodium under various conditions simulating a reactor environment, an experiment has been set up to measure the independent and combined effects of pressure, dissolved gas, heat flux, surface characteristics, and the pressure-temperature history of the system.

¹²Kaganove, J. J., ANL-6132 (Feb 1960).

Installation of the thermal-radiation heater was completed; the facility is ready for operation at higher heat fluxes. Reduction of the data from the earlier experiments is continuing.

2. Critical Flow

Calculations indicate that critical flow may occur in the core of a sodium-cooled reactor during an accidental power surge. Critical flow could lead progressively to detrimental effects such as voidage of the coolant channel, shock phenomena, and pressure buildup.

Critical-flow data for sodium have been collected, including exit critical pressure ranging from 0.5 to 8 psia, flow rates up to 1200 lb/sec/ft², and stagnation temperatures up to 1600°F; typical data were given earlier (see Progress Report for April 1967, ANL-7329). Data collection is continuing. Attempts are also being made to measure the initial superheat sustained before forced-circulation boiling takes place for different system pressures.

3. Electron-bombardment Heater Tests

The operation of various electron-bombardment heaters (EBH), which will simulate in-pile experiments pertaining to reactor safety, is being examined in a large evacuated vessel. Both single-pin and multipin heaters will be built and tested.

Fabrication of the first EBH test section is nearly complete.

4. Convective Instability

An interlock system has been designed and is partially installed on the control apparatus and the coolant units. All components for the test apparatus have been received and are being installed. Recorders and thermocouples have been calibrated as units. Some knowledge of the technique of measuring temperature has been gained recently. A unique welding unit is expected to be capable of silver-brazing the 10-mil stainless steel-sheathed thermocouples to the 1-mil foil used for the test section; if this is not possible, a previously derived comparison method will be used for measuring temperature.

Analytically, the first approximation has been obtained for the case of a constant wall heat flux via the method of integral relations. The third approximation for the constant-wall-temperature boundary condition still has not worked out successfully.

Fig. 35. Results with 4-pole Annular Linear Induction Pump (Solid Iron Core)

5. Thermophysical Properties of Sodium

A rough draft that recommends "best values" for several properties of the liquid and vapor phases of sodium is being reviewed. The draft is being modified to include the vapor-phase specific heat, Prandtl number, and thermal diffusivity; consideration is also being given to incorporating estimates of the compressibility and sonic velocity of the liquid phase, as well as estimates of the critical parameters: temperature, pressure, and density.

C. Fuel Meltdown Studies with TREAT

1. Pump Development for the Mark-II Integral Sodium TREAT Loop

The Mark-II loop design includes an annular linear-induction pump (ALIP) as a replacement for the thin-walled Faraday-type electromagnetic sodium pumps used in the Mark-I loops. This change was dictated by the establishment of a design pressure rating for Mark II of an order of magnitude greater than the pressure rating used in the older loops.

A four-pole and a five-pole version of the ALIP pump have been assembled. The four-pole assembly has been tested in the components test loop and found to fulfill the specifications established for the Mark-II loop.

The pump was tested using a solid Armco iron core because the diameter of the core could readily be changed. Three different core diameters were used during these initial tests. The effect on pump performance is shown in Fig. 35. Pumping efficiency with the nonlaminated core for

some operating conditions are also shown. The four-pole pump is being modified to use graded end poles and a laminated core to improve the efficiency. A large improvement in efficiency is not expected because of the rather thick tube wall of the pump and the large required thickness of thermal insulation.

Some insulation difficulties have arisen in tests with the 5-pole pump, and more adequate insulation is being applied to correct this problem. Other design modifications are also being made to the pumps.

D. Materials Behavior, Equation of State, and Energy Transfer

1. Physical Properties of UC at Very High Temperatures

The apparatus designed to measure some of the physical properties of carbide fuels has been described in Progress Report for November 1966, ANL-7279, pp. 92-93. Some design changes have been made.

Fig. 36. Schematic Figure of a Modified Graphite-sensing Rod Showing Tantalum Carbide Tip (Dimensions in inches)

It was observed that UC in a molten state rapidly dissolved the disc on the end of the graphite rod. A new design using a flared graphite end piece covered with TaC (see Fig. 36) was found to be acceptable for short time contact with liquid UC.

It was found that, at temperatures exceeding 1500°C , the helium gas between the heater filament, and the graphite rod and crucible became electrically conductive, making it impossible to use the high-frequency beat method to measure the distance between the UC and the crucibles and cover.

The cause for this behavior has not yet been determined. The distance between rod and heater filament is nearly 1.5 in. and the gas pressure is slightly above one atm. Thermoionization is not possible, since the ionization voltages of He and A are too high and the effect was observed

at temperatures as low as 1500°C . Some current measurements were made using helium atmosphere; the data obtained are shown in Table XXXI.

TABLE XXXI. Currents Measured Mode at Various Temperatures with Helium at One atm of Pressure and the Graphite Rod Connected to the Negative Side of the Power Supply

T (°C)	I _s (mA)	T (°C)	I _s (mA)
1750	3.6	2500	1350
2000	78.0	2650	2200
2250	480	2725	3240

Another method that uses direct electrical contact with the sample was substituted. The expansion coefficients of the fuel can only be determined as the difference of the observed micrometer measurements and the expansion of the TaC crucible and graphite rod.

The expansion coefficient of TaC was first determined using empty crucibles, while the coefficient for the ATJ graphite rod was calculated from handbook data.¹³ Preliminary investigation of the transition from the liquid to solid state of UC gave a linear value expansion between 3.3 to 5.5%. In this investigation pieces of sintered UC were melted, heated to about 2600°C, and slowly cooled. The UC and TaC reacted and formed several transition layers.

E. TREAT Operations

Two Zircaloy-clad, sintered-pellet UO₂ samples (CEN-230-T and -231-T) and two Zircaloy-clad, vibratory-compacted UO₂ samples (CEN-233-T and -234-T) were irradiated in room-temperature water in transparent capsules. Sample CEN-230-T was water-logged prior to the irradiation to simulate conditions that might exist if a water-reactor fuel element had defective cladding. Samples of the blanket gas from these capsules have been submitted for mass-spectrometer analysis to determine the extent of metal-water reaction. Development and analysis of the photographic film from these experiments is in progress.

Prototype Fast Ceramic Reactor (FCR) fuel samples C-4-A and C-4-D were irradiated and returned to the General Electric Company for examination. These were the FCR samples of sintered-pellet type, mixed Pu-U oxide fuel scheduled for irradiation in TREAT until preirradiated samples become available.

A cluster of five stainless steel-clad UO₂ pins was irradiated under steady-state conditions in a capsule designed for the measurement of pressure and sodium displacement during violent meltdown of fuel pins in sodium.

¹³Industrial Graphite Engineering Handbook, Union Carbide Corporation.

Radiochemical data obtained from analysis of these pins will be used to specify the reactor energy release required to obtain the desired test conditions during future transient tests using this capsule.

A test capsule containing three prototype fuel rods for the Power Burst Facility Reactor was loaded into the reactor as part of a series of one-hundred transient tests designed to produce maximum fuel temperatures similar to those expected during normal transient operation of the Power Burst Facility. Ten transient irradiations were completed during this period; the one-hundred series is scheduled for completion by July 15, 1967.

Investigation of the stud failures in the safety head flange installed in the large TREAT sodium loop (see Progress Report for March 1967, ANL-7317, p. 110) has revealed that the flange surfaces under the nuts were not flat, and differential thermal expansion between the B-7 alloy studs and the stainless steel flanges is calculated to cause excessive stress in the B-7 alloy studs at the design temperature for the loop. Either of these conditions may have influenced stud failure, but neither one fully explains why the studs did not yield significantly prior to failure. The fractured surfaces of the studs were coated with deposits of sodium oxide during the sodium fire; metallographic examination of these surfaces has not produced any useful information. Studs made from A-286 alloy have been ordered to replace the B-7 alloy studs in all the bolted-type flanges on the loop, but a delivery delay of twelve to sixteen weeks has been quoted by the vendor. Since there are only two flanges of this type which could cause a sodium leak, during test of the out-of-pile portion of the loop, A-286 alloy studs were fabricated by the ANL-Idaho shop for installation so that system testing can proceed.

The companion A-194 Type 3 nuts which have been ordered are scheduled for delivery by the first week in June. The safety head flanges are being spot faced to obtain flat surfaces under the nuts.

F. Chemical and Associated Energy Problems (Thermal)

1. Calculation of the Extent of Metal-Water Reaction, Core Heating, and Core Motion after a Loss-of-coolant Accident (CHEMLOC-II Program)

Analysis of the effect of the reaction of steam with cladding metal in a loss-of-coolant accident in a water-cooled reactor has continued. The loss-of-coolant accident, assumed to be caused by a break in a primary-system water pipe, has been taken to be the "maximum credible accident" for most of the power reactors built to date. The loss of water (moderator) following the pipe break causes the reactor to shut down (nuclear) and leaves the core in contact with steam. However, in the absence of auxiliary cooling, decay of fission products in the core releases sufficient energy to cause the cladding on the fuel to be heated, within a few minutes, to temperatures at which metal-steam reactions can occur.

Analyses have previously been made of the core heating and chemical reactions which can occur following a loss-of-coolant accident. In the first method of analysis (see Progress Reports for April 1965, ANL-7045, p. 67, and March 1966, ANL-7193, p. 94), simplification was achieved by dividing the core into 156 equal-volume segments, each of which was assigned to one of 10 groups according to its rate of heating due to fission product decay. The cladding reaction was calculated for each section assuming an adiabatic system. The effects of heat transfer in the core by conduction, radiation, and convection, and the effects of unequal steam distribution to the various segments due to hydrogen formation were neglected.

In the second method of analysis, CHEMLOC-I (see Progress Report for October 1966, ANL-7267, p. 78), a computer program was developed to describe core heating and chemical reaction up to the time of fuel melting. In the CHEMLOC-I analysis, the core is divided into a number of radial zones, each having the same number of fuel rods. The decay heating is considered to vary axially within each zone, as well as radially from zone to zone. This program takes into consideration the effects of evaluating separately the temperatures of the gas, cladding, and fuel. The effects of transfer of heat by axial conduction in the cladding and fuel, by radial conduction between cladding and fuel, and by radial radiation from cladding to cladding are considered. The effects of heat transfer by convection between the cladding and the flowing gas, and the effects of changing hydrogen concentration in the stream, are also examined.

The CHEMLOC Fortran computer program has recently been extended so that it describes the core heating and chemical reaction (cladding-steam) through core melting, core movement, and heating of the bottom grid plate. The extended program has been designated CHEMLOC-II. Besides the effects considered in the CHEMLOC-I program, the CHEMLOC-II program includes the effects of transfer of heat by axial radiation, and the effects of transfer of heat from the core to the wall surrounding the core, to the top grid support plate, and to the bottom grid support plate. Also included are the effects of core movement, due to slumping (melting), on the steam flow and transfer of heat, both axially and radially.

At temperatures up to the failure temperature of the core, an input parameter in the program, the heat and mass balance for an elemental section of a fuel rod together with the associated gas-flow area are essentially the same as for CHEMLOC-I. In addition, axial heat transfer by radiation has been included in CHEMLOC-II. The net rates of axial heat radiation from the cladding of an increment of rod, j , to the adjacent higher increment, $j + 1$, and to the adjacent lower increment, $j - 1$, are described as follows:

$$-q_{rj+1} = -\frac{A_v}{\Delta \ell} \pi r_0 \sigma \epsilon (T_{cj}^4 - T_{cj+1}^4), \text{ cal/}(\text{sec})(\text{cm}), \quad (1)$$

and

$$-q_{rj-1} = -\frac{A_v}{\Delta \ell} \pi r_0 \epsilon (T_{cj}^4 - T_{cj-1}^4), \text{ cal/}(\text{sec})(\text{cm}), \quad (2)$$

where

A_v = a geometrical view factor for vertical radiation, $\approx$ average radial distance between a rod surface and adjacent rod surfaces, cm;

$\Delta \ell$ = length of rod increment, cm;

r_0 = outside radius of rod, cm;

ϵ = emissivity factor for radiation;

σ = radiation constant, 1.37×10^{-12} cal/(sec)(cm²)(°K⁴);

T_{cj} = temperature of rod-cladding surface, increment j , °K;

T_{cj+1} = temperature of rod-cladding surface, increment $j + 1$, °K;

T_{cj-1} = temperature of rod-cladding surface, increment $j - 1$, °K.

a. Core Collapse and Meltdown Model. It is assumed that after the melting temperature of the cladding is reached, the major portion of the cladding and cladding oxide continues to adhere to the fuel, which remains erect until a collapse temperature is reached. This temperature is an input parameter to the program. At this temperature, the fuel-cladding mixture slumps to a lower position in the core. Some of the slumped material forms bridges between rods, some drips to lower positions, some falls down to the bottom of the core, and some falls through the bottom support plate and out of the core. As more of the fuel rods fail, bridging between rods forms a

support for the failed-fuel-rod material. It is assumed that the upper section of the rod is unsupported, and that it will rest upon the surface of the bridged material. As failure proceeds, the rods collapse into the slumped material, now partially molten, in a manner similar to a rod of metal being melted in a crucible.

Fig. 37. CHEMLOC Calculational Model Showing Idealized Slumped Sections of Two Rods

Figure 37 shows idealized slumped sections of two rods which are used for the purpose of calculations. Where two adjacent rods are slumped and come in contact, heat transfer, both axial and radial, is by conduction only. When a predetermined number

of rod increments in a zone have slumped, the steam flow is assumed to be blocked in that zone. The slump factor, S_F , is the ratio of the idealized length of the slumped material to its original length. After the rod increments have slumped, the radial heat transfer by radiation ceases and radial heat transfer by conduction commences. Radial heat transfer by conduction for slumped increments is accounted for as follows: the heat-transfer rates from a rod in zone i to zone $i + 1$ and to zone $i - 1$ (per cm of rod) are

$$-q_{ri+1} = -S_F \frac{4\pi}{N_r} (i) k_{ai} k_{am} (T_{cij} - T_{ci+1,j+n_s}), \text{ cal/((sec)(cm))}; \quad (3)$$

$$-q_{ri-1} = -S_F \frac{4\pi}{N_r} (i-1) k_{ai} k_{am} (T_{cij} - T_{ci-1,j+n_{ss}}), \text{ cal/((sec)(cm))}, \quad (4)$$

where

S_F = slump factor;

N_r = number of rods per zone;

k_{ai} = thermal conductivity of solid fuel, cal/(sec)(cm)(°K);

k_{am} = ratio of conductivity of slumped (molten) fuel to that of solid fuel;

n_s = number of increments of length displacement in zone $i + 1$ from zone i due to unequal slumping;

n_{ss} = number of increments of length displacement in zone $i - 1$ from zone i due to unequal slumping;

T_{cij} = temperature of cladding of increment j in zone i , °K;

$T_{ci+1,j+n_s}$ = temperature of cladding of increment $j + n_s$ in zone $i + 1$, °K;

$T_{ci-1,j+n_{ss}}$ = temperature of cladding of increment $j + n_{ss}$ in zone $i - 1$, °K.

Because of the material that falls below the bridged fuel-cladding mixture, three input parameters are required in the program: (a) a drip factor, D_F , by which the effect of heat transfer due to dripping of fuel from a slumped increment to the unslumped increment below is accounted for; (b) a fall factor, F_F , by which the effect of fuel falling from a slumped increment to the bottom of the core is accounted for; and (c) a fuel-retained factor, F_{RF} , by which the effect of fuel falling entirely out of the core is accounted for.

The major effect of dripping of fuel from a slumped increment to a lower, colder increment is to increase the axial heat transfer rate. Thus, the normal heat-transfer rate is multiplied by the drip factor, $D_F \geq 1$.

When an increment of fuel slumps, a fraction F_F remains at the increment and a fraction $1 - F_F$ falls to the bottom of the core. Of the material $(1 - F_F)$ falling to the bottom of the core, a fraction F_{RF} remains at the bottom, depositing around the base of the fuel rods, and a fraction $1 - F_{RF}$ leaves the core entirely, falling through the bottom grid support plate to the bottom of the reactor vessel. Thus, when an increment of fuel slumps, a fraction F_F of the fuel remains at the increment, a fraction $(1 - F_F)(1 - F_{RF})$ falls completely out of the core, and a fraction $(1 - F_F)(F_{RF})$ is retained in the bottom of the core.

Fig. 38. CHEMLOC Rod-slumping Models Showing the Limiting Cases of Extent of Dripping and Falling of the Slumped Fuel

Figure 38 shows the two limiting cases of the extent of dripping and falling of the collapsed fuel. With no dripping or falling of the collapsed fuel, the drip factor D_F and the fall factor F_F are both equal to 1. In the most severe case, all of the collapsed fuel immediately falls to the bottom of the core, where it fills the void portion of the core. For this case, $F_F = 0$ and D_F can be roughly calculated for a given core. The actual case would tend to be somewhere between these limiting cases and requires an estimation of the three factors D_F , F_F , and F_{RF} .

When increments of the rods in a zone, i , slump, the following changes are made in the program calculations:

1. The steam flow in the zone is assumed to be blocked when a predetermined number of increments of the rods melt and slump (this number of increments is also an input parameter).
2. The radial transfer of heat by radiation is replaced by radial transfer of heat by conduction for the slumped increments. The effects of the unequal slumping in the various zones are taken into account.
3. The axial transfer of heat by radiation is discontinued within the slumped increments.
4. The area for the axial conduction of heat is increased because of the slumping. Corrections are made for the decrease in increment length.
5. The axial conduction of heat is increased through the use of the drip factor D_F .
6. Material is lost from the core through the use of the fuel-retained factor F_{RF} .
7. Material falling to the bottom of the core is accumulated there by use of the fall factor F_F .

The readout of the program includes the following for pre-selected time intervals:

1. steam flow rate;
2. decay heat as a fraction of reactor power;
3. fuel and cladding temperatures throughout the core;
4. thickness of cladding reacted and fraction of steam (or hydrogen) in the gas throughout the core;
5. average thickness and fraction of cladding and cans (extraneous oxidizable materials) reacted;
6. fraction of hydrogen in the exit gas;
7. total heat of combustion of the hydrogen being released;
8. total sensible heat of the gases being released;
9. accumulated quantity of hydrogen released;
10. average temperatures of the bottom grid plate, wall surrounding the core, and top grid plate including their associated parts.

b. Computer Calculations. Computer calculations were made for three reactors as follows: a 2255-MWt boiling-water reactor (BWR), a 2758-MWt pressurized-water reactor (PWR), and the 50-MWt Loss-of-fluid Test (LOFT) reactor. Some of the important dimensional parameters for these reactors are shown in Table XXXII.

TABLE XXXII. Dimensional Parameters for Boiling-water and Pressurized-water Reactor Cores

Reactor Model	BWR	PWR	LOFT
Power, MWt	2255	2758	50
Fuel Rods UO ₂			
No. of Fuel Rods	35476	39372	3328
No. of Fuel Assemblies	724	193	52
Length of Fuel, in.	144	144	36
Thickness of Zircaloy Cans, in.	0.085	0.061	0.025
OD of Zircaloy Cladding, in.	0.570	0.422	0.390
Cladding Thickness, in.	0.036	0.0243	0.025
Fuel Pellet Dia, in.	0.488	0.366	0.340
Pitch (Square of Rods)	0.738	0.556	0.580
Control Rods			
No. of Assemblies	177	53	12
Rods per Assembly	1 ^a	20 ^b	1 ^a
Control Rod Material	B in SS	Cd in Ag-In	B in SS
Pitch, in.	12		
Equivalent Core Dia, in.	182.7	133.7	42
Vessel Inside Dia, in.	251	173	58
Reactor Pressure, psig	1000	2250	2330

^aCruciform.

^bRound.

The rate of steam generation is an important input variable. Figure 39 shows a generalized chart for steam generation as a function of bottom head thickness (bottom thickness of the pressure vessel) and the

Fig. 39. Generalized Steam-generation Rate Used as Input Parameter for CHEMLOC [lb/(hr) (sq ft of head surface) in contact with water]

square feet of the head in contact with water. The chart is based on a reactor operating temperature of 285°C (saturated steam pressure of 1000 psi) and a water boil-off temperature of 127°C.

The values of the additional important input information (such as the physical properties of reactor materials, reactor power levels, and fraction of heat initially retained in the core) are listed in Table XXXIII. The values of the input parameters shown were selected as being reasonable for a hypothetical accident. However, several of these values depend on the conditions postulated for

the assumed accident. Thus, the results of the calculations performed in this study are general in nature, and not representative of calculations for a specific reactor.

TABLE XXXIII. Important Variable Parameters for Boiling-water and Pressurized-water Reactor Cores

Reactor Model	BWR	PWR	LOFT
Specific Reactor Operating Power, cal/cm ³	34.4	67.4	61.1
Steam Evaporation Temperature, °C	127	127	127
Ambient Temperature, °C	285	285	285
Slumping Temperature of Fuel-cladding Mixtures, °C	2800	2800	2800
Fraction of Core Heat Retained at End of Blowdown, %	75	75	75
Weights: Bottom Grid Plate, g	6×10^6	4×10^6 (10 ⁷) ^a	5×10^5
Top Grid Plate, g	7.5×10^6	5×10^6	2.5×10^5
Core Wall, g	1.5×10^7	10^7	10^6
Radiation Emissivity of Cladding	0.62	0.62	0.62
Ratio of Heat Conductivity of Molten to Solid Fuel Mix	2.0	2.0	2.0
Fraction of Core Material Falling out of Core on Slumping	0	0	0
Steam Flow Rate (Variable ^b), lb/hr, at Indicated Time after Blowdown			
0 sec	86,666	41,333	4,600
50 sec	33,849	16,143	1,796
100 sec	24,903	11,877	1,321
1000 sec	8,185	3,903	129
2000 sec	5,525	2,766	10.6
3000 sec	3,786	2,151	0.9
Core Decay Heating (Variable), % of Reactor Operating Power, at Indicated Time after Blowdown			
0 sec	6.33	6.33	6.33
10 sec	4.96	4.96	4.96
100 sec	3.08	3.08	3.08
1000 sec	1.66	1.66	1.66
2000 sec	1.34	1.34	1.34
4000 sec	1.06	1.06	1.06

^aCase 2.

^bSee Fig. 39 for relative variation with time.

Figure 40 illustrates the results for the BWR, Fig. 41 the results for the PWR, and Fig. 42 the results for LOFT, assuming no auxiliary cooling. These figures show the steam flow rate, the percent cladding and cans reacted, the percent hydrogen in the exit gas from the core, the number of zones with steam flow, the average bottom grid support plate temperature, the average inner wall temperature, and the zone-1 temperatures as a function of time.

It will be noted that the PWR and LOFT core materials heat up much faster than the BWR core material. This is mainly due to greater specific reactor power (about twice as great as for the BWR) and smaller amounts of core material per gram of fuel to be heated up for the PWR and LOFT cores. Also, the fraction of cladding reacted in LOFT is greater than for PWR or BWR. This is caused by the fact that the LOFT rods are only $1/4$ as long as the rods in PWR and BWR, and hence have roughly $1/4$ as much cladding per rod. However, the steam flow rate per rod for oxidizing cladding in LOFT is initially as great as for PWR (although it falls off more rapidly). Although BWR has a greater steam flow rate than PWR (due to larger pressure vessel diameter), this is offset by a greater quantity of cladding material and a slower heating rate to reaction temperature for BWR.

Percent Cladding and Can Reacted

Bottom Grid Plate Temperature

Zone 1 Temperatures

Figure 40

CHEMLOC Calculational Results for a 2255-MWt Boiling-water Reactor (BWR)

Percent Cladding and Can Reacted

Bottom Grid Plate Temperature

Zone 1 Temperatures

Fig. 41. Calculational Results for a 2758-MWt Pressurized-water Reactor (PWR)

Percent Cladding and Can Reacted

Bottom Grid Plate Temperature

Zone 1 Temperatures

Fig. 42. Calculational Results for the 50-MWt Loss-of-fluid Test (LOFT) Reactor

From the heating rate curves, it is estimated that the bottom grid plates, for the conditions assumed, would fail in periods of time as follows:

LOFT: 2000 sec
PWR: 3000 sec
BWR: 4000 sec

G. Plutonium Volatility Safety

1. Chemistry of Tellurium Fluorides

Studies are being carried out with an air- TeF_6 gas-flow system to determine the efficiency with which several promising materials remove low concentrations of TeF_6 from air streams. A tellurium decontamination factor (DF) of >700 is required. Earlier work with activated alumina (see Progress Report for April 1967, ANL-7329) showed that the required TeF_6 decontamination factor can be readily achieved with this sorbent. In recent work, magnesium fluoride, Linde Molecular Sieve 13X, and BPL-type activated charcoal have been evaluated. Sorption of TeF_6 on MgF_2 at 25°C was slight, and the DF obtained with Molecular Sieve 13X was less than 700 except at the low TeF_6 loading of 1.60% (percentage of nitrogen B.E.T. surface area covered by a monomolecular layer of TeF_6).

Sorption studies on activated charcoal were carried out using a 2-in.-high bed, a gas velocity of 20 ft/min, a temperature of 25°C , and a TeF_6 concentration of 500 ppm. It was observed that at a bed loading of 2.5%, the bed showed good sorption stability but that activated charcoal begins to desorb TeF_6 rapidly at bed loadings above 3.8%.

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