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Argonne National Laboratory

REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

July 1967

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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

July 1967

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Issued August 31, 1967

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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:

January 1967	ANL-7302
February 1967	ANL-7308
March 1967	ANL-7317
April 1967	ANL-7329
May 1967	ANL-7342
June 1967	ANL-7349

REACTOR DEVELOPMENT PROGRAM
Highlights of Project Activities for July 1967

EBWR Plutonium Recycle Experiment

Power operation of EBWR was terminated on June 29, 1967. Boric acid worth was verified and the nine-rod-bank worth, was measured. The nine-rod-shutdown criterion and the minimum boric acid concentration for satisfaction of the eight-rod shutdown was determined.

The natural uranium zone and the enriched uranium shim zones are now being unloaded. The plutonium will be removed after some critical loading parameters are measured for this type of fuel loading.

EBR-II

Run 25 has been completed except for a short running time allowance for power-coefficient measurements prior to reloading for Run 26.

There have been no fission gas releases since subassembly XO11 was removed from the reactor, confirming that XO11 was the source of the gas releases. Plans to examine the capsules in XO11 are being made.

Problems were encountered with the main core gripper during fuel handling when difficulty was encountered in releasing a subassembly. Removal of the gripper for investigation and replacement with a spare gripper have begun.

Gradually decreasing values in the magnitude of power coefficients in the region from 10 to 30 MW have been observed. A model which may explain the decrease has been developed and is being studied.

Fabrication of EBR-II driver fuel continued normally. Fifteen subassemblies were fabricated this month, and 25 alloy-preparation runs were made.

Work on the cold-fuel line continued. Operation is expected to begin by the end of August.

The Vertical Assembly-Dismantler Machine was installed, tested, and is in operation.

ZPR-3

Assembly 50, the third in the series of simple plutonium-fueled assemblies being studied in ZPR-3, was made critical this month. The worth of edge core drawers and the worth of control-rod drawers as compared with core drawers were measured. Analysis of data from Assembly 50 and previous assemblies continues. Some data for Assembly 48B, for which the analysis is complete, are given.

ZPPR

Good weather has aided the continued construction of the ZPPR facility this month. Concrete for the ring beam has been poured. The catenary array consisting of 80 cables has been pulled. The cell crane components have been placed in the cell. The roof of the support wing is complete, and finish work on the inside walls and doors is continuing. Installation of insulation, piping, and air-handling ducts has been started. Processing of the sand for the cell roof and emergency filter was completed.

The final safety analysis report has been transmitted to the AEC for approval.

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I. PLUTONIUM UTILIZATION--EBWR

A. Operations

The power operation of the EBWR was terminated June 29, 1967, thus completing the reactor's fuel irradiation for the Plutonium Recycle Program. During July, critical runs were made for foil irradiations and to obtain postirradiation measurements of boric acid worth, verification of nine-rod shutdown, and determination of minimum boric acid concentration for satisfaction of eight-rod shutdown.

Following these critical runs, the defueling operation began. The reactor defueling was interrupted for determining the postirradiation plutonium critical loading.

By the end of July, all postirradiation physics measurements were complete, and all fuel had been removed from the reactor to the fuel storage pit.

Lay-up of the EBWR facility began shortly after shutdown. To date, the reboiler facility lay-up work has been completed and lay-up of the turbine plant partially completed.

The second set of seventeen irradiated fuel pins was sent to Battelle-Northwest Laboratory via the "Yankee" cask on July 5. These pins arrived at Battelle and were removed from the cask without difficulty. The completion of the plant lay-up, transfer of fuel to another storage facility, and a third shipment of fuel to Battelle will complete the final phase of EBWR operation.

B. Reactor Physics

After waiting several days for EBWR to cool after shutdown, the worth of boric acid was remeasured by a procedure¹ that also yielded nine-rod-bank worth, verification of the nine-rod shutdown criterion, and the minimum boric acid concentration for satisfaction of the eight-rod-shutdown criterion.

Figure 1 contains curves of the nine-rod bank height for criticality and the eight-rod bank height for criticality (with the control rod of greatest influence withdrawn to its maximum allowable height) as functions of boric acid concentration. Extrapolation of the curve of critical

¹Kier, P. H., "Initial Critical Experiments of the Experimental Boiling Water Reactor (EBWR) Plutonium Recycle Program," Reactor Physics Division Annual Report, July 1, 1965 to June 30, 1966, ANL-7210 (Dec 1966) 29-34.

Fig. 1. Control-rod Bank Heights as a Function of Boric Acid Concentration

eight-rod bank height to zero height yields a minimum boric acid concentration of 2.0 g/gal for satisfaction of the eight-rod-shutdown criterion. The eight-rod-shutdown criterion precludes direct verification of the nine-rod-shutdown criterion that the fully rodded, unborated system be subcritical. However, nine-rod shutdown can be verified if the extrapolation of the curve of critical nine-rod bank height to zero boric acid concentration yields a positive intercept. From the figure we see that the intercept is above 10 in.; hence, the nine-rod-shutdown criterion is satisfied for this loading.

Measured worths of boric acid and the nine-rod bank are given in Table I. Boric acid worth was measured for concentrations between 2.25 and 5.00 g/gal in five steps. The results yield a boric acid worth of -10.5 ± 1.6 mk/g/gal, with no

dependence of worth upon concentration. This measured worth in the irradiated loading is some 20% lower than the measured worth in the unirradiated loading. This result is in quantitative agreement with calculations² that predicted a reduction in boric acid worth of 10-25% for a burnup of 0.6% in the plutonium zone.

TABLE I. Worths of Boric Acid and Nine-rod Bank^a

Boric Acid (g/gal)		Nine-rod Bank (in.)		Reactivity Change (mk)	$(\Delta k / \Delta c)$ (mk/g/gal)	$(\Delta k / \Delta H)$ (mk/in.)
Range	Change	Range	Change			
5.00-4.44	0.56 ± 0.03	18.1-16.85	1.25 ± 0.1	5.67 ± 0.57	-10.1 ± 1.5	4.5 ± 0.9
4.44-3.90	0.54 ± 0.03	16.85-15.8	1.05 ± 0.1	6.09 ± 0.61	-11.3 ± 1.7	5.8 ± 1.1
3.90-3.28	0.62 ± 0.03	15.8-14.7	1.1 ± 0.1	6.31 ± 0.64	-10.2 ± 1.5	5.7 ± 1.1
3.28-2.77	0.51 ± 0.03	14.7-13.8	0.9 ± 0.1	5.62 ± 0.57	-11.0 ± 1.6	6.2 ± 1.2
2.77-2.28	0.49 ± 0.04	13.8-13.0	0.8 ± 0.1	5.37 ± 0.55	-11.0 ± 1.7	6.7 ± 1.3

^aCriticality measurements made on EBWR with plutonium loading showing relationship between boric concentration in water with the height of control rods (8- and 9-rod banks).

²Toppel, B. J., Vogelberger, P. J., and Wimunc, E. A., Safety Analysis Associated with the Plutonium Experiment in EBWR, ANL-6841 (Nov 1966).

II. LIQUID-METAL FAST BREEDER REACTORS

A. EBR-II

1. Operations

Reactor operation at 30 MW continued with experimental irradiation subassemblies XG05 and XA08 in the core (see Progress Report for June 1967, ANL-7349, p. 5) until completion of the 150-MWd run on July 4. No fission gas release was detected, confirming the evidence that subassembly XO11 was the source of the previous releases (see ANL-7349, pp. 4-5).

Following a scheduled 24-hr shutdown, the reactor was again started on the evening of July 5 and power was raised incrementally to 45 MW while observing for a further gas release. Run No. 25 continued until July 20, when it was shut down for the insertion of experimental subassemblies XO23 and XO24 into the core for a scheduled 24-hr irradiation at 30 MW.

During the return to power, a significantly reduced power coefficient was noted between 10 and 20 MW (see Sect. II.A.4.d). The power was then reduced to 50 kW while arrangements for rod-drop measurements and additional determinations of power coefficient were made. Subsequent increase of power to 30 MW and rod-drop experiments indicated that there had been no change in the prompt negative components of the power coefficient and that the operating limits for power coefficient would not be exceeded. The irradiation was then made as planned.

Following removal of XO23 and XO24 from the reactor after irradiation at 30 MW for 16 hr, power-coefficient measurements to 45 MW were made. After this run, a program of raising and lowering fuel and blanket subassemblies to randomize the core and blanket was started.

Problems were encountered with the main core gripper when it would not release sixth-row subassembly No. B-341. With difficulty, the subassembly was transferred to the transfer arm. During this operation it became evident that rotational control of the subassembly had been lost. An unsuccessful attempt was made to place the subassembly into the storage basket. Therefore, it was transferred to the IBC through the Fuel Unloading Machine (FUM). At month's end the subassembly had been transferred to the FCF Air Cell for inspection. Verification of damage to the core gripper was made by engaging an outer blanket subassembly and checking the rotational motion of the gripper. The unit could be rotated as much as 45° without interference, indicating that the rotational positioning blade may be damaged.

It is planned to remove the main core gripper for inspection and any necessary repairs.

Tests were conducted on the heat-rejection capabilities of the cooling tower and the heat rate of the reactor building with the air supply isolated (see Sect. II.A.1.b.ii).

a. Subassembly Transfers. Six experimental subassemblies: Nos. XO23, XO24, XO25, XO26, XO27, and XO30, one special control rod, and two driver subassemblies were received. Experimental subassemblies XO23 and XO24 were returned to the Fuel Cycle Facility.

b. Operational Support and Maintenance

(i) Spare Parts. Purchase orders have been placed for additional spare parts for the following equipment:

- (a) miscellaneous charged-wire monitor equipment;
- (b) components for the main turbine vapor extractor;
- (c) negator motors for the Fuel Unloading Machine;
- (d) components for the secondary EM-pump vacuum pumps;
- (e) ac motor for the FUM argon-system gas booster;
- (f) motor and gear box assembly for the fuel-handling main gripper and holddown.

The packet for purchasing the materials for fabrication of a spare subassembly holddown has been completed. The purchase orders will be issued after a final review is completed.

(ii) Maintenance. Work was performed on the following equipment during this report period:

The FUM gripper drive chains were removed, inspected, cleaned, and realigned. New bearings were installed where required and the FUM was put back in service.

The variacs on the auxiliary EM pump were replaced with new units. This was done as a preventive maintenance effort. The original variacs will be cleaned and restored for use as spares.

The reactor building suspect exhaust containment valve was leak rate tested. Results indicated a leakage rate of $0.8 \text{ ft}^3/\text{day}$, which is within the allowable limits.

An atmospheric heat-up test was performed on the reactor containment building. All building-supply and exhaust blowers were secured, and all building refrigeration units were kept operative. The starting and

equilibrium temperatures for the operating floor, basement, and subbasement were as follows, respectively: 73 and 79°F, 72 and 87°F, and 79 and 82°F. During the approximately 8-hr test, it was necessary periodically to vent the building to prevent excessive pressure buildup.

A new seal was installed in the reactor building air supply isolation valve. Test of the repaired penetration was completed and gave a leak rate of 1.8 ft³/day.

Inspection of the bearings on the motor-driven condensate pump revealed that they were in excellent condition.

The resistance thermometer for monitoring the field coil temperature in the No. 2 primary pump motor failed. Installation of a new thermometer required disassembly of the bell housing and upper bearing assembly. The pump was reassembled and is ready for operation.

The clutch of the subassembly holddown drive failed due to slippage of a shaft key. Repair of the clutch has been completed, and the holddown has been returned to service.

Routine equipment preventive maintenance, as well as the repair of minor steam leaks and equipment, continued.

c. Operator Training. Training activities consisted of rewriting training packages for the power plant, electrical, and coolant areas. Five-lesson-plan training packages were completed; two additional packages are 90% complete. Two operators have successfully met qualification standards. Two new hires completed the facility and plant orientation course.

d. Operating Manual. The preparation of new procedures and extensive revision to existing sections of the facility Operating Manuals continued. Revision Bulletin No. 53, containing revised master Startup Check Sheets for reactor startup and restricted and unrestricted fuel handling, was issued.

Revision of the System Data Sheets has been completed and the revised versions submitted for review and acceptance. Review of Division IV F-7, "Argon System"; IV-C, "Primary Sodium"; and V-D, "Induction and Resistance Heating," has begun.

The following Special Operating and Administrative Procedures were issued:

- (i) SOP No. 75--EBR-II Power Coefficient Measurements, Requirements for Reactor Operation.

- (ii) AP No. 33--Fuel Handling, Revision 1.
- (iii) AP No. 54--Mandatory Use of Protective Equipment during Handling of Acid and Corrosion Inhibitor.

2. Fuel Procurement

a. Mark-IA-IB Fuel Procurement. Good progress is being made by the Aerojet-General Corp., which was awarded a contract for the fabrication of 34,000 Mark-IA fuel elements. Aerojet has awarded subcontracts for the procurement of fuel-element tubing (Bishop and Co.) and tips (Pressed Metals), uranium conversion (NFS), scrap recovery (NFS), isotopic analyses (Isotopes, Inc.), and fissium alloying elements (Engelhard and Climax Molybdenum). Negotiations are underway on the procurement of restrainers, and of an alloy-preparation furnace, along with crucibles and molds. It is planned to accomplish fuel-element identification by roll-marking the tip with five 1/32-in.-high numbers, and a marking machine is being procured for this purpose. During a visit to the San Ramon plant in the middle of this month, progress in the modification of facilities was observed. Partitioning, rough plumbing, and lighting and power wiring were completed, along with air-conditioning ductwork. Hoods and some work surfaces have also been fabricated. Drawings for the injection-casting furnace, the centrifugal bonding machine, and the wire-wrapping machine were reviewed. The concepts appeared adequate to accomplish their intended purpose, and detailing is underway.

Aerojet has also applied for amendment of its SNM license and has submitted the necessary supporting documents.

The specifications for fuel-element tubing are being reviewed to determine the applicability of hydrostatic, flanging, and flattening tests. Invitations have been issued for bids on the fabrication of upper and lower subassemblies of reactor core hardware which will be used to contain the fuel elements, and comments on the specifications have been received from the AEC.

3. Reactor Improvements

a. New Control-rod Subassemblies. The original control-rod design has been modified to increase the hydraulic holddown effect; control rods of 50% greater reactivity worth are being developed to permit the reactor to operate with eight, instead of the original 12, control rods (see Progress Report for June 1967, ANL-7349, pp. 13-14).

An all-absorber control-rod design is being studied to replace the present fuel-type control rod so that four of the present 12 control-rod locations might be made available for the irradiation of experimental devices that require access from outside the primary tank.

Studies of the boron carbide absorber-type control rod are continuing. Various cross-sectional configurations were investigated to determine relative volume percent of the absorber material, sodium coolant, and stainless steel. Associated gas pressures and material stresses are being calculated.

Plans are being formulated to perform in-reactor tests of the boron carbide absorber to obtain data regarding the volume of gas generation at various burnups. Physics aspects and compatibility with adjacent experimental capsules for in-reactor tests will be evaluated to establish optimum placement and design of boron-carbide capsules.

b. Irradiated-fuel Monitor. To provide for positive and immediate identification of spent, refabricated, or new fuel subassemblies during fuel-handling operations within the primary tank, a scanning device employing an ionization chamber to monitor individual subassemblies is being developed and tested.

Assembly of the irradiated-fuel-monitor drive mechanism was completed. The control and instrumentation cabinet was assembled, wired, and connected to the drive motor and limit switches on the motor-drive mechanism. The mechanism was operated from the control cabinet. During tests of the drive mechanism, a slight interference was exhibited between the left support column and the vertically operated instrument-support plate; this binding was attributed to a bend in the left support column, which apparently can be corrected by filing or grinding the column. Further tests will be performed after electrical installation is complete. The instrumentation cables are being connected to the traveling preamplifier; the instrumentation will be tested without a radiation source and at room temperature. Subsequent hot testing will be performed at the EBR-II Idaho facility:

c. Oscillator Rod Mark-II. To permit transfer-function measurements even at low rotary speeds, where excessive bowing of the rod caused the rod to rub against its thimble, an improved rod is being developed; other design modifications will improve handling and storage.

The assembly drawing for the reduced-diameter rod using standard-size boron carbide and aluminum oxide capsules was approved, and detailed fabrication drawings were started. Fabrication drawings for the oscillator-rod storage-basket adapter sleeve were completed and are being checked.

It has been determined that a stainless steel dummy rod would be required to be inserted in the reactor when the oscillator rod is removed; a second storage-basket adapter sleeve would be necessary for the dummy oscillator rod.

d. Reactor and Primary-coolant-system Equipment

(i) Seal for Intermediate Heat Exchanger. A study is being conducted to determine whether a seal or baffle in the annular gap between the intermediate-heat-exchanger tube bundle and primary-tank nozzle would reduce the thermal stresses in the welded joint between the nozzle and primary-tank-cover base plate. If a sodium-flow restrictor is found to be warranted, it will be designed.

Preloaded spring segments are being considered as a means of providing a restriction to the flow of hot sodium in the annulus formed between the tube bundle and primary-tank nozzle. Engineering sketches have been completed for the spring segments; plans have been made to fabricate a typical baffle from Inconel-X and to conduct a test to determine spring-constant and loading characteristics of this design.

Engineering drawings are being prepared for a test rig and for a typical baffle. Test baffles will be fabricated from Inconel-X in 0.031-, 0.025-, and 0.015-in. thicknesses. The test rig will simulate heat-exchanger curvature and will be used to determine desirable baffle spring tension.

e. New Subassemblies Design and Experimental Support

(i) Mark-II Core. To improve the attainable burnup and reliability of the driver fuel, and also to reduce costs by simplifying its fabrication and assembly, the driver fuel is being modified to accommodate greater fuel growth by changing fuel-element dimensions; to maintain the same overall dimensions of the subassembly, this requires changes to the upper and lower sections of the axial reflector. Out-of-pile and in-pile tests are being employed to verify the design.

(a) Fuel Subassembly. The Mark-II fuel assembly is being flowtested to ensure that its pressure drop matches that of the existing fuel subassemblies (see Progress Report for June 1967, ANL-7349, pp. 11-12).

Hydraulic flowtests of the Mark-II fuel subassembly (24-in. element length) are being continued with various types of upper and lower shield sections. On the bases of these tests and shielding-effectiveness evaluations of the various shield sections, the Model-SCPS (Stainless Central Plug Shield) shield section has the most desirable configuration. Flow characteristics match those of the original standard core subassemblies.

Thermal-stress analyses of the Model-SCPS shield pieces indicate that thermal-shock stresses will be of an acceptable magnitude.

(b) Fuel Elements. To raise achievable burnup from ~1.2% with the Mark-I and Mark-IA driver fuel to ~2% with the Mark-II fuel element, which will have the same U-5% Fs alloy fuel, Type 304ELC stainless steel jacket material, jacket outer diameter, and spirally wrapped spacer wires, the Mark-II fuel element will have a larger gas plenum at its top, a thicker jacket, a thinner fuel pin, and thicker sodium bond to allow greater fuel-pin swelling and fission-product gases to escape from the fuel alloy.

Mark-II fuel-element mockups were flowtested in a partial subassembly that had Model-SCPS shield sections.

(ii) Irradiation Subassemblies. Prototypes of various types of special irradiation subassemblies are being designed, fabricated, and tested to provide vehicles for irradiation experiments. Each of five types is being designed in several models to accommodate experimental fuel elements or capsules of different sizes and to achieve specific experiment environments.

(a) Mark-E. The Mark-E (or "non-melt-through") irradiation subassembly, which would be used for experiments involving risk of melting the samples, would contain a partial fuel meltdown and confine all damage within the subassembly.

Detail drawings of a prototype Mark-E subassembly are being made. After slight modifications, the set of assembly drawings for the Mark-E19 model is complete. When fabricated, this test model will be flowtested to determine pressure drop. A series of special tests will determine the effectiveness of the upper and lower particle-collector devices.

f. Instrumented Subassembly System. To provide the capability for measuring operating conditions of experimental fuel and related components while they are being irradiated in the reactor core, a system is being developed for installing and removing an instrumented experiment subassembly in a control-rod location in the core by use of the normal fuel-handling equipment. Special tools will handle the instrument leads, which will extend out through the reactor-vessel cover and rotating shield plug to appropriate receiving instruments.

(i) Design. The system and its components are being designed and evaluated, specific problems are being identified, necessary research and development is being determined, and effects of the instrumented-subassembly system on other EBR-II systems and operations are being determined.

(ii) Subassembly Attachment and Lead Severance. Various concepts are being designed and evaluated for coupling and decoupling the top of the instrumented subassembly to the extension tube and for severing the instrument leads so the assembly can be removed from the core.

Another coupling (see Fig. 2) has been designed, together with a simplified version of an axial cutter (see Progress Report for January 1967, ANL-7302, pp. 15-18). This combination does not rely on a gripper for the operations consisting of (1) inserting and lifting the subassembly and (2) severing the connection between extension tube and subassembly after completion of an irradiation experiment. The gripper designed to go with this coupling and cutter system serves as a backup for a case of malfunction of the coupling or of a subassembly's being stuck above the position where the reactor's internal fuel-handling mechanism can reach it.

With this arrangement, insertion into the reactor proceeds by lowering the subassembly and extension tube, which are preassembled with the coupling into one rigid unit; a hoisting device or crane attachment is connected to the upper end of the extension tube. Similarly, the subassembly can be raised, when this is required for operation of the reactor's rotary plug, by pulling it up by means of the extension tube.

Severance of the lead wires and rupture of the coupling at the end of an experiment proceeds as follows: (a) the subassembly is raised $3/8$ in. with the extension tube; (b) the tube is locked in this position by a collar above the reactor plug; (c) the cutter is inserted into the extension tube; (d) the leads, which are near the inner wall of the extension tube, are detached and threaded through the bore of the cutter when the cutter is inserted; (e) as the cutter passes beyond the position shown in Fig. 2, the leads are sheared; (f) the cutter continues its downward motion until it "bottoms" on the clamping ring and forces the subassembly downward and out of the extension tube; (g) after deflecting the diaphragm and its backup rings a few thousandths of an inch, the backup ring bottoms against the (slightly relieved) shoulders of the corresponding clamping rings and the diaphragm is sheared; (h) the extension tube is withdrawn while the cutter holds the subassembly down. After disengagement is completed, the cutter can be withdrawn while a sensing rod (not shown) is held on top of the subassembly lifting fixture to ensure that the cutter does not lift the subassembly.

Several separations were carried out with a model in which the cutting of the diaphragm was accomplished without difficulty. Measured forces showed that the lead wires were cut completely before the diaphragm was severed; the force required to cut the diaphragm was consistent within 2%.

The subassembly may also be raised by means of the gripper attached to the subassembly lifting fixture (see Fig. 2). Projecting ridges inside the gripper sleeve engage corresponding slots in the base of the subassembly lifting fixture (inducing a slight rotation of the sleeve). To prevent the gripper from disengaging, a locking blade is inserted into the slot at the top of the lifting fixture before lifting is started. In addition to

Fig. 2. Coupling and Axial-type Lead Cutter for EBR-II Instrumented Subassembly. Portion to left of centerline depicts cutter; portion to right of centerline relates to a gripper, which serves as backup.

providing a backup, use of the gripper is also considered a desirable alternative for raising the assembly by a means other than by the extension tube.

(iii) Bending Tests. Bending tests were performed on the top-end fixture, the hexagonal support, and the internal-bayonet-latch couplings (see Progress Report for June 1967, ANL-7349, pp. 12-13 and Fig. 2 on p.13). Preliminary to applying lateral loads, free play in each coupling was measured in three axial planes normal to the faces of the hexagonal lower tube. The top-end fixture was spring-loaded for the test; thus there was virtually no free play in this coupling. However, a gap on one side between the subassembly and the extension tube began to open early in the loading sequence. The gap increased with load and at a loading moment of 1,000 lb-in. the test was stopped. Upon release of load to zero, this coupling failed to return to its original configuration, apparently due to friction between the parts induced by the retaining springs. Jarring the test specimen upon removal from the machine caused the coupling to return to its original alignment. The coupling has been redesigned to eliminate the spring loading.

Free play in the hexagonal support coupling for the three positions averaged $1^{\circ}05'$. Additional distortion was apparent as load was applied up to a moment of 2,500 lb-in., where the test was stopped according to plan. (This moment, arbitrarily set as a measure of desired stiffness of the assembly, is much above expected sodium drag moments in the reactor.) Upon release of load to zero, this coupling returned to its original configuration, indicating elastic behavior.

Free play in the internal-bayonet-latch coupling for the three positions averaged $0^{\circ}31'$. A small gap opened at one side between parts as load was applied. However, the coupling withstood the planned loading moment of 2,500 lb-in. and returned to its original configuration upon release of the load.

(iv) Torsional Tests. Torsional tests were conducted on a mockup of the extension tube. The Type 304 stainless steel tube was of 2.5-in. OD of 0.120-in. wall thickness and 24 ft 4 in. long. Torques at intervals of 20 lb-ft up to 300 lb-ft were applied in both directions, producing total twists of 4° for both cases. The test results agreed closely with calculated values.

Torsional tests on the test specimens for the top-end fixture, the hexagonal support, and the internal-bayonet-latch couplings are in progress. It was found necessary to modify two of these specimens to adapt them for the torsional tests.

(v) Extension-tube Seal. Because the extension tube and attached instrumented subassembly must be raised ~8 ft to permit reactor-fuel handling, a sliding seal is being designed to prevent leakage of

radioactive cover gas around the extension tube where it passes out of the reactor vessel through the control-rod-drive penetration in the small rotating plug.

Two methods of sealing the extension tube are being investigated. One method employs a motor-actuated packing gland similar to the seals used for sealing the vertical shafts of the cover-lifting mechanism and gripper and holddown mechanism. The second method involves a set of nesting bellows attached to the extension tube and the rotating plug so that the bellows can be expanded over the full 8-ft operation of the extension tube.

The packing-seal approach requires force-limiting devices to protect the subassembly system from excessive frictional forces imposed by the packing gland; it also may require maintenance, such as replacement of seal rings, each time a subassembly is replaced. Because the bellows-seal method does not have these disadvantages, it is preferred. The feasibility of using bellows for this application is being assessed. A bellows assembly is being procured for testing.

(vi) Drywell Concept. To simplify subassembly fabrication by shortening the sheathed part of the instrument leads above the subassembly from ~26 ft to ~7 ft, introduction of a sodium-free drywell space between the extension tube and an inner hexagonal tube is planned. The sheathed instrument leads would be sealed at the bulkhead of the drywell, where they would be connected to conventional flexible wiring.

It appears that a brazed bond between the leads and the bulkhead will be most practicable. Because the brazing process will be one of the final steps in the assembly of the instrumented subassembly, high integrity of the seal is sought without impairing any of the leads. Work has begun for evaluating brazing alloys and techniques and preparing samples.

Methods for welding the sheathed mineral-insulated cables to the bulkhead were considered, but the close tolerance and difficulties in applying consistently reliable welds make this method an alternate only. For further evaluation, mockup parts for a bulkhead have been fabricated and mineral-insulated cables will be welded into these parts.

(vii) Subassembly Pressure Drop and Power Output. The total pressure drop has been estimated for sodium flow through an EBR-II instrumented subassembly by computation and by using experimental flow data determined by previous flowtests for various control-rod parts.

The computations were performed for a hexagonal tube of 1.824-in. ID, containing 19 elements, each of 0.360-in. diameter and 40 in. long. Each element is wrapped with a 0.040-in.-dia spiral spacer wire on

a 6-in. pitch. Fluid conditions were: an average temperature of 800°F (700°F inlet and 900°F outlet) and specific heat of 0.305 Btu/lb/°F. The lower adapter, having inlet orifices and containing a reflector section, is identical to that of a control-rod subassembly. Therefore, pressure-drop data from previous water-flow tests of this part were used in estimating the total pressure drop. Then the total pressure drop through the subassembly was broken down as follows: Δp_1 entrance losses through six orifices in the lower adapter and loss through the lower reflector section, Δp_2 entrance loss to the element bundle, Δp_3 exit loss after the element bundle, Δp_4 friction loss through the element bundle, Δp_5 entrance loss to the upper adapter, Δp_6 friction loss through the upper adapter, and Δp_7 exit loss after the upper adapter. Consideration has been given to the case in which the lower reflector section is deleted and to the case in which a flowmeter (of 1-in. ID and 4 in. long) is included in the subassembly. With the available primary-system coolant pressure drop between the high-pressure plenum to upper plenum (at Row 7) of 40.8 psi,³ it appears that, with the corresponding sodium flow rate of about 50 gpm, a heat rate of 460 kW may be produced at a temperature rise of 230°F. The 40.8-psi pressure drop is related to a total reactor power of 62.5 MW.

g. Constant Power Supply. The constant power supply provides power for reactor control, nuclear instrumentation, control, and instrumentation of the secondary and steam systems. The existing equipment consists of a motor-generator which operates either from the normal power supply system through a rectifier or, in case of an emergency, from a storage battery system. The improvements consist of adding a synchronous motor in tandem with the dc motor. The new system will then provide power for the normal operation with the synchronous motor driving the generator. Under an emergency condition when normal power is not available, the dc motor will drive the generator from a storage battery system.

The synchronizing switches have been received from the manufacturer, uncrated, and placed into position. Overhead cable trays and cabling necessary to interconnect to the existing system are being installed. Final interconnection, checkout, and testing will be performed early in September 1967.

h. Cooling Tower for EBR-II Plant Auxiliary System. The present cooling water system is constructed so that cooling water for both the plant auxiliary systems and the main condenser is circulated through one large cooling tower. Installation of a separate cooling tower to handle the plant auxiliary cooling load offers the following operating advantages:

³Koch, L. J., et al., Addendum to Hazard Summary Report Experimental Breeder Reactor-II (EBR-II), ANL-5719 (Jan 1964), pp. 63-64.

(i) Transferring the auxiliary cooling load to the small tower will permit shutdown of the main tower for maintenance and cleaning during periods of reactor shutdown.

(ii) Load separation will minimize the problems of tower icing during period of cold-weather operation and minimum cooling demand, and also will eliminate operation above ambient temperature of the auxiliary systems when operations require full reactor steam bypass.

The new cooling tower will greatly increase operating flexibility and reduce maintenance requirements, which will improve both safety and plant availability.

Approval to proceed with this Work Project was received from DRDT during the month and engineering work for this system was initiated. Arrangements, criteria, and specifications are being prepared to provide the architect-engineering information to be used for design services on this project.

i. Fuel Unloading Machine (FUM) Revisions

(i) Mechanical System. Operation of the FUM with the prototype Mark-III gripper continued with no difficulties encountered. Approximately 35 subassembly transfers were made. Due to the fuel-transfer schedule, the FUM became available for modification work for one week this month. The system was still operating satisfactorily, but there seemed to be excessive friction in the elevation-drive and jaw-actuator systems. The entire elevation-drive system including the gripper and main chains was removed and disassembled for inspection. The cause of the excessive friction was found to be in the upper chain-guide assembly which houses four chain-guide rollers. Two of the guide rollers were seized and had started galling between the jaw-actuator chain and the rollers. The main chains were cleaned, and all damaged areas were dressed and polished.

The upper chain-guide box was redesigned to accept ball bearings (rollers were originally on bronze bushings) on all four rollers. The side clearance for the chains on the rollers was increased, and the diameter of the two upper rollers was decreased to lessen chances of binding as the chains enter the guide assembly. Several ball bearings on the main drive system were also replaced.

The Mark-III prototype gripper was also cleaned and disassembled for inspection. All parts were in excellent condition. There was no indication of binding or dragging of the jaws due to sodium oxide deposits. The mechanical interlock was also in perfect condition, with no evidence of improper operation. The gripper was reassembled and installed with no maintenance required on any of the parts.

Two new main gripper chains were also ordered. These will be fabricated by the Argonne Central Shops and will be identical to the existing chain installation.

The entire elevation-drive system was reassembled and the FUM was leak tested. Normal fuel transfers were started again. At present the operation of the gripper and actuation system is very good, with less friction encountered than ever before.

(ii) FUM Cooling System. The design of a new filter unit for the FUM argon gas system has been completed. The design incorporates a micrometallic filter medium and a baffle-type separator. The purpose of the new design is to prevent particulate matter from entering the heat exchanger and the turbines. Installation drawings for the new filter unit are being prepared.

(iii) FUM Cooling Instrumentation. Sequential control of the FUM argon gas system is through programmed relay operation. The present control relays are spring return on de-energize and have been the source of many spurious malfunctions. These relays will be replaced with a relay using both springs and gravity to open when de-energized.

To reduce installation time, the replacement relays have been assembled into a new cabinet; interconnected, timed operations set; and programmed operation checked. The interchange of relay cabinets will be made at the next scheduled maintenance down time.

j. Rotating Plug Seal Cleaning System. Both the large and small rotating plugs on EBR-II make use of annular dip ring seals to isolate the primary-tank argon blanket gas from the atmosphere. A low-melting-point bismuth-tin eutectic alloy is used as the sealant. Drilling of access holes into the outboard, air-side of both seals along with use of a "vacuum cleaning" technique to remove oxide and dross material has greatly improved performance of the rotating plugs. A considerable amount of new eutectic alloy seal material has been added to both seals.

As a long-term solution to the seal problem, a seal-alloy removal and cleaning and replacement system is under development and design. This system will not only maintain the purity and cleanliness of the seal alloy, but will also assure that the time required for heating and melting the seals prior to plug rotation is at a minimum.

The areas of the large and small rotating plugs where the alloy cleaning system will be installed are under preparation. Work continued on surveillance of seal melting times and temperature distributions throughout the rotating plug seal troughs.

The prototype equipment developed at Argonne, Illinois, for this operation was received in Idaho, and preparations are proceeding to erect the equipment for further testing and development.

The alloy in the troughs of the large and small plugs was cleaned by brushing with a stainless steel wire brush, which was found previously to be effective in removing the dross. This technique simply involves inserting a round bristled brush, about 3/4 in. in diameter, into the trough then rotating and oscillating it in the molten alloy. The dross is worked into the bristles and largely remains there when the brush is withdrawn from the access hole in the plugs. This material is then tapped out of the brush into a container.

Approximately 143 and 68 lb of dross and alloy were removed, respectively, from the large and small plug troughs.

k. Reactor Systems Instrumentation Improvements

(i) Temperature-monitoring Devices. The MV/I converters have been delivered and shop calibration completed. The units have been installed in the respective drawers for system operation. Installation of this equipment is now scheduled following reactor Run No. 26.

(ii) Multipoint Recorders. The installation of the replacement units has been completed. The replaced units have been returned to the factory.

This portion of the task is now completed.

(iii) Pressure and Temperature Sensors. The review of the complete thimble assembly has resulted in several changes in the design. Several welds have been modified, the assembly has been shortened $1\frac{1}{4}$ in., clearance and tolerances have been relaxed, the bottom step in the assembly has been redesigned to strengthen it against possible fatigue failure, and a sodium-leak detector has been added to the instrumentation.

Fabrication of the parts making up the upper flange portion is complete. These parts are prerequisite to fabricating the pressure sensors by an outside vendor and represent "long lead" items. The schedule for this plant modification has slipped approximately two months since its inception due to review, redesign, and a shortage of drafting time. It is now hoped that this instrumentation can be installed during the long reactor shutdown tentatively planned for March 1968.

l. Radioactive Sodium Chemistry Loop and Cell. A single piping loop in a shielded cell with provisions for installation of sampling and in-stream analysis devices, and with the necessary instrumentation (temperature, flow, pressure, radiation, etc.) for safe, efficient operation of the loop

will be extended from the existing primary sodium-purification system. The loop will provide capabilities for: 1) improved sampling of radioactive EBR-II primary sodium coolant for chemical analyses, and 2) proof-testing, under practical plant operating conditions, of sodium-coolant sampling and analytical devices undergoing development by ANL and others.

These capabilities will help to assure safe and reliable EBR-II reactor operation insofar as coolant purity is concerned. They will also be used in defining the coolant environment of experimental fuel elements and materials specimens undergoing irradiation in EBR-II, which is at present the only reactor available for fast-neutron irradiations in the U.S.A. supporting the LMFBR program. The testing of developmental sampling equipment and analytical devices likewise supports the LMFBR program, since these are being developed for application in future liquid metal-cooled reactors.

A general design description for the loop and cell was prepared, as well as a listing of the types of sampling and analysis equipment to be utilized and tested. Various cell and piping arrangements were investigated, and flow, pressure-drop, and piping-stress calculations were made. Detailed design of the loop and cell is scheduled to begin in August. The target date for initial usage of the loop for EBR-II sodium-quality monitoring and equipment testing is November 1968. The loop and cell description and projected usage are given below.

(i) General Description

(a) Location and Arrangement. The loop and cell will be constructed in the reactor building basement, adjacent to the existing primary purification cell, which contains the cold trap and its ancillary equipment and piping. The loop will be shielded independently of the purification cell. The loop probably will be connected to the existing influent and effluent piping of the purification system in the shielded pipe trench (in the "depressed area" just below the operating floor) external to the purification cell. The improved surge tank and syphon-break system of the purification loop will also be used for the chemistry loop.

Isolation valves will be provided, and the arrangement of the pipe heaters will be such that either the chemistry loop or the purification loop can be isolated and frozen while the other loop remains in operation.

The current concept calls for the construction of three or four "cubicles" to contain sampling and analytical equipment. The shielding will be such that access can be attained to an individual cubicle, and work can be performed upon components in which the sodium has been frozen and radioactivity has decayed to permissible levels while the main loop and one

or more sampling branches (in other cubicles) remain in hot radioactive operation. This concept requires installation of the main loop piping within its own shield such that radiation levels in the cubicles from the sodium in the main loop are not excessive. The piping to carry sodium sample streams from the main loop to the cubicles must penetrate the main loop shield in such a manner that excessive radiation streaming cannot occur.

The current concept calls for access to the sampling cubicles through top hatchways with the possible exception of the cubicle containing distillation and freeze-line samplers [see Section (ii) below] which may be provided with an access door for convenience. An overhead I-beam and trolley will be required for handling shielding hatch covers and cubicle equipment items.

The cubicles will be made as large as space and arrangement permit so as to maximize equipment space and accessibility for installation and removal, repair, modification, etc.

The final loop and cell arrangement, and the number of sampling cubicles, will depend largely upon the size and feasible locations of the loop circulating pump and its power supply. The requirements, if any, for reinforcement of the basement floor to support the load of the cell shielding must be ascertained.

(b) Radiation Shielding and Containment of Sodium Leakage.

The general shielding arrangement is described in the preceding Sect. (a). Lead will be used wherever possible to allow a shielded space as large as feasible. Where lead cannot be used, high-density concrete will be used.

Steel cell liners and drip pans will be used in the main piping cell and in the sampling cubicles to contain sodium in the event of leakage from piping, valves, sampling and analytical components, etc. Equipment for extinguishing sodium fires will be provided for the cubicles, similar to that for the primary cold-trap cell.

Sodium-leak detectors will be included as part of the operating instrumentation for the loop [see Sect. (e) below].

(c) Pump and Piping. A direct-current electromagnetic pump will be used, similar to the purification-system pump. The highest flow rate attainable will be provided, consistent with the maximum size of pump and power supply which can be accommodated (but not larger than the existing purification-system pump). A high flow rate through the main loop piping is desirable in the interest of obtaining representative samples of the bulk primary sodium, of minimizing changes of sodium composition during transit through the loop, and of providing flexibility for potential future uses of the loop in component testing.

The pipe size for the main loop will be in the range from 1 to 2 in. IPS, depending upon the flow rate finally selected.

The necessary throttle valves will be provided to control pressure drops and flows in the main loop and in the sampling cubicle branches. Loop "root valves" will be provided for isolation of the loop from the primary purification system when necessary.

All joints in the main loop piping and all sampling branch-line connections to the main loop will be welded, in accordance with established specifications for sodium service.

Isolation valves will be provided in the sample stream supply and return lines for each sampling cubicle. These valves will be welded into the sample piping branches.

Connections of components to sampling lines inside the sampling cubicles will utilize nonwelded types of tubing and piping connections wherever feasible, for operational and experimental flexibility and convenience.

All main-loop and branch-line isolation and throttle valves, with the possible exception of the loop-isolation root valves, will be operated by reach rods or other remote manual operators which penetrate the shielding. Safety considerations may dictate that the loop-isolation root valves be closed automatically upon receipt of a signal from sodium-leak detectors (for example) or that the capability must be provided for root-valve closure initiated from a remote pushbutton station. These points will be resolved during detailed design of the loop and cell.

Electrical resistance-type pipe-heating circuits will be used for the main loop piping and the sampling branches.

(d) Cell Ventilation and/or Inert Gas Atmosphere. Ventilation of cubicles will probably be required for control of ambient temperatures during operation. The ventilation system will incorporate exhaust blower(s) and fire-resistant absolute filter(s) to prevent excessive release of radioactive sodium oxide smoke to the building atmosphere in case of sodium leakage and fire. This is the type of system currently in use for ventilation of the primary cold trap cell.

(e) Main-loop Instrumentation. Instrumentation will be provided for measuring and recording the temperatures of sodium, piping, and ambient cell atmospheres.

The pump discharge pressure and the pressure drop between the supply and return legs of the main loop will be measured and recorded. This pressure drop drives sodium through the parallel sampling branches in the cubicles.

The total flow in the main loop and sample stream flows in the branches to the cubicles will be sensed by electromagnetic flowmeters and recorded.

Gamma-radiation levels in the main-loop piping cell and in each of the sampling cubicles will be measured and indicated on monitor meters. Sodium-leak detectors will be provided in the main-loop piping cell and in each sampling cubicle. A leak detector will also be installed in the leak-containment enclosure of the DCEM pump, similar to the leak detector of the purification system pump.

(f) Control. As noted in Sect. (c) above, reach rods or other remote operators will be used for manual control of loop isolation and throttle valves (with the possible exception of the isolation root valves). The loop pumping rate can be varied by manual adjustment of the pump power supply. Thus, loop and sampling branch flows and pressure drops will be established and controlled manually.

Temperature controllers, with thermocouple temperature sensors, will be used with the resistance-heating circuits for the sodium piping. Heater sequences for melting of sodium in the loop will be initiated manually by the system operators.

(ii) Types of Sampling and Analysis Equipment To Be Utilized and Tested. Current plans call for installation in the sampling cubicles of the devices listed below. Since a separate shielded cubicle cannot be provided for each device, tentative combinations are shown, on the basis that four cubicles can be provided. If only three cubicles can be provided, the combinations will, of course, be revised.

<u>Device</u>	<u>Cubicle</u>	<u>Remarks</u>
Distillation Sampler	A	Consideration will be given to providing a shielded viewing window and manipulative capabilities in this cubicle for distilling radioactive sodium under visual observation, using equipment of the type being developed by Reactor Engineering Division, ANL.

<u>Device</u>	<u>Cubicle</u>	<u>Remarks</u>
Bypass Freeze-line Samplers	A	Plans envisage the use of several samplers in a parallel manifold to permit taking bulk samples for various analytical purposes. High-purity nickel (INCO-270) tubing will be used routinely.
Carbon Analyzer (United Nuclear Corp.)	B	According to the manufacturer, the inlet sample stream must be heated to 1400°F.
DC Resistivity Meter and/or Rhometer	B	
Oxygen Analyzer (United Nuclear Corp.) and Radiation Meter	C	According to the manufacturer, the sample stream temperature must be controlled near 600°F. Testing will determine the effect of gamma radiation upon instrument response.
"Pilot" Cold Trap and Impurity Saturation Detector	D	The first device is intended to provide information concerning the types and amounts of impurities being deposited in the purification-system cold trap. The second is to determine the deposition behavior of primary-sodium impurities as a function of temperature. It is essentially a length of tubing with a known gradient from the bulk-sodium temperature to a temperature somewhat above the freezing temperature of sodium. Deposition of impurities is determined by removal of the tubing from the dissolution or mechanical removal of frozen sodium, and examination of the tubing surface.
Sample Filters (for particulate impurities)	D	The purpose of this device is to determine the types and quantities of particulate impurities in the primary sodium. If satisfactory filter media of graded porosity can be found, it should be possible to obtain information concerning particle-size range also.
Continuous Automatic Plugging Meter	D	This type of plugging meter, devised by the British and the French, should be tested under plant operating conditions for comparison with the plugging meters in the existing primary purification system.

In addition to the foregoing devices, it probably would be possible to operate a "hydrogen meter" of the type undergoing development by APDA. However, the value of such operation in EBR-II seems questionable, either from the standpoint of characterizing the EBR-II coolant or from the standpoint of testing the device (because the hydrogen level in the EBR-II primary sodium is low and should remain low). No plans currently exist for installation of a hydrogen meter.

If a "nitrogen meter" is developed at some future time, it could probably be installed and operated in Cubicle C.

m. Oscillator Rod Mark II. The necessary parts for the modification to the sensing rod are essentially complete. The new lower drive assembly, including the gripper jaws and bellows assembly, has been received in Idaho. Preparations are being made to begin reassembly of the oscillator drive shaft and bearing-carrier housing so that the unit can be reinstalled as quickly as possible when the installation date is established.

4. Reactor Analysis and Testing

a. Relocation of Data Conditioning System. The relocation of data conditioning equipment for reactor kinetics experiments to the recently refurbished data acquisition room in the power plant building was accomplished and modification of the equipment was started. The modified equipment will permit greater flexibility and yield signals of improved quality.

b. Nuclear Analysis

(i) Reactivity Worths. The average reactivity worth of materials as a function of radial position in the core for Run No. 25 with stainless steel reflector in Rows 7 and 8 have been measured and calculated using the MACH I code. A comparison of the values is given in Table II.

TABLE II. Measured and Calculated Reactivity Substitution Values in Run 25D

Substitution		Position	Measured (Ih)	Calculated (Ih)
In	Out			
XO11	1/2-worth Subassembly	2F	-15	-12
Core Subassembly	Stainless Steel Blanket	6N1	+150	+101
Depleted Uranium	Stainless Steel	6N1	-.9 ^a	-.7.4
Total Worth of Control Rod		5N1	+135 (avg)	+142
Total Worth of Control Rod		5N3	+151 (avg)	+164

^aAverage value for Row 6.

c. Thermal and Hydraulic Studies

(i) Decrease in Reverse Subassembly Bowing. Calculations are underway to determine the effect of reducing the flow in the 8th-row stainless steel subassemblies to 1.7 gpm. Reducing the flow should sharply decrease temperature differences between opposite sides of Row 7 subassemblies and temperature gradients across the 8th row. Consequently, the reversed bowing effect of subassemblies in the 7th and 8th rows should be decreased.

(ii) Heat Transfer Calculations. Two generalized heat transfer programs, TIGER V and THTB, have been obtained and made operational on the CDC-1604 computer at the NRTS. Both may be useful, and a comparison is in progress to determine which will be most effective, in terms of ease of input preparation and machine time, for EBR-II heat transfer problems.

A consolidation of several heat-transfer-oriented programs has been started. At the present time the results of core physics calculations are used as input to a curve-fitting program which produces a continuous curve of fission rate versus radial position. The output of this program is used as input to the REFER program, which calculates a relative fission rate for every pin position in a $1/12$ (the minimum symmetry sector) section of the core. It also calculates the average relative fission rate for a subassembly and for each sector ($1/12$) of a subassembly, as well as breaking down the relative fission rate for each pin or portion of a pin in a sector into a relative fission rate for 19 heat transfer surfaces within the sector. The output of the REFER program is then used as input for the modified HECTIC II program.

It is planned that the various programs shall be combined so that it will be possible to use the input data from physics calculations, together with the number of effective subassemblies of each typical core location (e.g., 3N2, 6N3, 4) and obtain in one step the complete temperature distribution within the core.

d. Safety

(i) Power-coefficient Measurements. Near the end of Run No. 25, after a total power accumulation of 13,765-MW days, the reactor was shut down to permit the loading of two experimental subassemblies, XO23 and XO24, into positions 2B1 and 7D4, respectively. Upon startup, July 21, it was noted that while 12.5 lh of reactivity were necessary to bring the power from hot critical to 10 MW, only 0.3 lh were used in continuing upward to 20 MW. Since this behavior differed appreciably from that noted in previous Run 25 startups, the system was shut down to permit the formulation of plans for a systematic investigation.

Later, on July 21, the power was raised incrementally to 30 MW (the level required for the irradiation of XO23 and XO24). Power-coefficient measurements were made at 10 MW and in 2.5-MW increments thereafter. Since the operating behavior at 30 MW differed little from that previously experienced at 45 MW, the programmatic commitment of 30-MW operation to irradiate the experiments was completed. After 16 hr at 30 MW, the system was shut down, XO23 and XO24 removed, and the reactor brought to a power level of 45 MW in order to extend the range of power-coefficient information. Shortly thereafter power was reduced in approximately 10-MW steps. A comparison of final and initial critical positions revealed a net loss of less than one inhour. It may be inferred from the data that the presence of XO23 and XO24 in the core was not responsible for the "flattening" of the power coefficient curve in the 10-20-MW region since the increasing power data for July 21 and 22 were taken with and without the experiments, respectively.

Fig. 3. Differential Power Coefficient in Run No. 25, EBR-II.

An illustration of the change in the nature of power coefficient is given in Fig. 3, which compares earlier differential power coefficient data (April 1967) with those taken on July 22. Although the overall shapes are similar, it is nevertheless evident that the differential power coefficient has been reduced in the region from 10 to 20 MW. It is equally evident from the data of Fig. 3 that in the region of 45 MW the differential power coefficient is strong.

To provide additional information, rod-drop tests were conducted at the following levels of power: 500 kW; and 20, 30, and 41.5 MW [see Sect. (iii) below]. While the results of these tests are still being processed, a preliminary analysis of data taken at 20 and 41.5 MW reveals that no deterioration in the prompt negative power coefficient component has occurred.

(ii) Power-coefficient Model. Although the reduction of the power coefficient discussed above does not compromise the operation of the system in the operating power region, the possibility of a further reduction at intermediate levels of power is being investigated. Accordingly, effort has been directed at an understanding of the change in terms of thermal-mechanical-reactivity feedback effects.

At the present time the most promising postulate involves a gradual statistical ordering of clearances between blanket subassemblies. While definitive proof is still lacking, it seems reasonable to believe that inverse temperature gradients across Row 8 stainless steel reflector units effect a bowing situation in Rows 7 and 8 which is aptly described as a double-leaf "buggy spring" configuration. Hence, the effects of a power increase are reflected through increased spring action as a compression of core subassemblies and an outward "drift" of blanket units.

To explain the change in the nature of the power coefficient it is necessary to postulate the following: that on shutdown the effective radius of the inner boundary of the blanket has been increased slightly, i.e., some fraction of the normal clearances between blanket units remains closed. Vital to this model is the additional postulate that clearances between core subassemblies tend to randomize during and after shutdown. Although no proof currently exists, it seems reasonable that high coolant flow rates through core subassemblies may randomize core clearance systems while the much lower flow rates in the blanket are insufficient for restorative action. If such is true, it follows that the effect of a power cycle will be reflected by an increase in deformational freedom for all subassemblies lying within the first eight rows. Consequently, during the following power cycle, more room will be available for inward (positive) subassembly bowing effects.

Calculations have been made which indicate that a radial expansion of the core of 0.001 in. has a reactivity worth of 3 lh. Therefore the net blanket drift needed to explain the recent decrease in the overall power coefficient amounts to only 0.003 in., an amount credible in terms of existing blanket clearances.

To refine and to substantiate the validity of the model, two courses of action are planned: one is based on a comparison of expected behavior with existing power-coefficient data for a variety of conditions; the other is based on a proposed experiment in which the power coefficient is remeasured after a deliberate attempt is made to randomize blanket units with the fuel handling equipment.

If the blanket-drift model can be substantiated, it would explain the effects of the stainless steel reflector on the Run 25 power coefficient; the hysteresis noted in all (including Run 25) power-coefficient data; and, perhaps, establish the limit of additional power-coefficient changes which might be expected.

(iii) Rod-drop Analyses. Attempts are being made to place a quantitative evaluation on the results of rod-drop experiments performed during Run 25. The approach currently being used involves the following:

1. a recording of reactor power decay following a ramp reactivity (approximately 15-17 lh);;
2. an inverse kinetics calculation based on delayed-neutron parameters;
3. subtraction of zero-power kinetic effects from time-dependent reactivity feedback effects at the power level of interest;
4. a Fourier analysis of the reactivity feedback;
5. a polynomial fit to the Fourier analysis;
6. the construction of a feedback model;
7. substantiation of the feedback model.

At the present time, difficulty is being experienced in perfecting a computer program for polynomial fitting of the results of the Fourier analysis. It is expected that preliminary results will be available during the August reporting period.

(iv) Noise Recordings and Analyses. Noise recordings were taken at power levels ranging from 500 kW to 45 MW. Upon reduction of the data to rms values, a comparison of recent values with those measured last month was made. The results of the comparison indicated for both sets of data that below 35 MW the major noise component is one having a frequency of 10 cps. For levels above 35 MW the amplitude of the 10-cps oscillation decreases.

During rod-drop tests conducted on July 21, 22, and 23, it was noted that the amplitude of the 10-cps oscillation at lower levels of power was considerably decreased. Attempts to associate the origin of the 10-cps component with structural ordering effects are continuing.

(v) Power Coefficient--Considerations of Reduced Flow. As part of a program devoted to an understanding of EBR-II feedback, analytical expressions have been developed for the power coefficient as a function of coolant flow rate. Fundamental to the development were detailed temperature evaluations for specific core regions. Resulting from the mathematical development is a separation of two power-coefficient components: one, linear with respect to flow, and concerned with normal expansion effects; the other, nonlinear, and associated with subassembly bowing. The magnitude of the latter was established by subtracting calculated values of the linear component from the experimentally measured overall power coefficient. Such separations have been carried out for a variety of power and flow conditions at the beginning of Run 25, with the result that the nonlinear component appears to vary with the power-to-flow ratio. It has also been calculated

that at the beginning of Run 25 the nonlinear (bowing) component is negative from 0-16 MW, turns positive at 16 MW, and the overall component reaches a maximum value in the vicinity of 40 MW, where the differential value of this component is zero.

e. Experimentation

(i) Gas Release Tests in the IBC. The possibility of using the Interbuilding Coffin (IBC) as a fuel-element-failure locating device has long been suggested. Essentially the technique is based on transferring a suspect subassembly to the IBC, reducing the internal pressure, and monitoring the IBC gas for rare gas fission product activities released, presumably, through the defect.

The recent identification of Subassembly XO11 as responsible for release of fission product gas provided an opportunity to test the method with a subassembly known to contain a failed fuel element and capsule. Accordingly, subassembly XO11 was transferred to the IBC which, in turn, was sent to the interbuilding passageway in the usual manner. The tests involved reducing the internal pressure in 3-in. (Hg) increments between 25 in. (atmospheric pressure) and 5 in. Following each pressure reduction, 15 min were allowed to elapse before gas sampling. In no case was evidence of gas release noted.

From the readings of thermocouples located at the exit side of the subassembly and in the gas-circulation loop, it was apparent that bond sodium had frozen and was apparently sealing the defect. As a remedial measure, insulated heating wire, with a gross capability of 1200 W, was installed around the inlet recirculating pipe. While inlet gas temperatures rose eventually to 255°F, the outlet gas temperature never exceeded 140°F. Nevertheless, tests were conducted at pressures as low as 5 in. (Hg). In no case was gas evolution noted. It was surmised that bond sodium apparently has plugged the defect through freezing.

In a second series of tests, conducted in the reactor room, attempts were made to maintain the bond temperature above the freezing point of sodium. For 15 min prior to the transfer of XO11 from the FUM to the IBC, hot argon gas (200-350°F) from the FUM was purged through the IBC in an effort to preheat the inner skirt. Shortly after transfer, the pressure in the IBC was reduced directly to 2 in. (Hg), the system immediately backfilled with tank argon, and gas samples taken. No evidence of gas release was noted. Although the bond sodium was molten upon entry into the IBC (as indicated by thermocouple readings), the time elapsed between entry and final vacuum pulldown (15 min) may have been too long to prevent freezing.

The failure to release entrapped fission product gases through the defect tends to detract from use of the IBC as a failure-locating device. On the other hand, if sodium bonds were truly frozen at the time of pumpdown, the concept may still merit additional consideration.

(ii) Preliminary Proposal for Critical Experiments with EBR-II.

Current proposals for core modifications (Mark-II fuel) and reflector changes (nickel versus stainless) have prompted a study of the possible application of critical facility measurements. A preliminary proposal for a set of critical experiments has been prepared. Specific areas in which critical measurements could benefit analyses of EBR-II problems include the following:

1. evaluation of the relative merits of nickel, stainless steel, and uranium reflectors;
2. interpretation of material movements in terms of a reactivity effect; such information, as it pertains to subassembly bowing, structural expansion, etc., would contribute significantly to a better understanding of reactivity feedback effects;
3. core modifications associated with the Mark-II-fuel loading. The Mark-II loading will involve the following changes: an increased fuel enrichment, a larger gas reservoir, a less dense upper axial reflector, and the possible use of combination fuel-poison-loaded control rods. If more than a presently allowable number of control-rod positions are used for the installation of instrumented subassemblies, experimental information will be needed to redesign the remaining control rods to satisfy shutdown and operational reactivity requirements;
4. determination of neutron flux through the driver fuel, in the vicinity of experimental subassemblies, and in poison-loaded control rods. In order to place a more rigorous interpretation on burnup-damage effects, flux data with an accuracy of $\pm 5\%$ are desirable. The results of critical facility measurements for a variety of typical situations should facilitate a better understanding of flux-depression effects and the extent of interaction between experimental fuel specimens and driver elements.

A study is being made to weigh the usefulness of critical facility measurements in the areas outlined above compared with facility availability, alternative methods for obtaining the same information, and the application of more rigorous computational techniques.

5. Development of Methods of Nuclear Analysis

The preliminary computations for checking and debugging the burnup cycle code PYRO have been successfully completed. PYRO is a

CDC-3600 code which calculates the isotopic burnup in a two-dimensional x,y configuration as a function of time. It is structured to be used in conjunction with the Transport SNARG 2D Code. An option is included in the PYRO Code for the printout of the geometrically fine regional reaction rate distribution. This allows reaction-rate studies to be compared with experimental distributions determined from wire-foil irradiations.

A comparison has been made of the U^{235} and U^{238} fission-rate distribution between measurements made in Run 25 and those computed from a two-dimensional SNARG S-4 in x,y geometry. The preliminary computations assume a 4-fold symmetry of the Run-25 loading and utilize the EBR-II 6-group cross-section set 23606. The computed U^{235} fission rates were found to agree to within 20 percent of the experimental values near and in the reflector region. The agreement in the central core region was found to be within 10 percent. The U^{238} fission rates were in general predicted to about 10 percent of the measurements. The computed values used in the comparisons are rates averaged over a quarter-region representation of EBR-II subassembly.

An improvement in the reaction-rate distribution is expected to be made in subsequent studies. A two-fold symmetry of the core and reflector configuration would account for the local effects resulting from the nonuniform distribution of driver fuel and test assemblies. Another factor which is expected to improve the agreement is the addition of reflector-averaged cross sections to the 6-group set currently being used. Efforts in these areas of study have been initiated.

A series of test- and driver-fuel irradiation and replacement steps are being planned in support of the current EBR-II test-fuel irradiation program as well as in support of the EBR-II Improved Driver Fuel program.

6. Reactor System Testing, Surveillance, and Evaluation

a. Reactor Building Containment System. A 6-hr leak-rate test at approximately 2 in. of water pressure of the primary tank annulus indicated that measurable gas leakage from the annulus was essentially zero.

b. Fuel-handling Systems. A comprehensive engineering review of the preventive maintenance procedures for the control-rod drives, core gripper, and core holddown was made. The necessary maintenance jobs and their frequency of performance were reviewed. Several changes and recommendations were made. The revised procedures are presently being written for the permanent maintenance-procedure files.

Assistance was also given to the maintenance group in the ordering of the necessary materials required for the fabrication of a new

core-subassembly holddown mechanism lower assembly. Some of the components of this unit are long-lead time items, and it is desired to have the necessary materials in stock in the event of damage to the present unit.

c. Cooling Tower. A test of the main cooling tower, to determine its expected performance at full load during hot weather, indicates that at 62.5 MWt and with ambient wet bulb temperature of 60°F the temperature of cooling water to the tower will be somewhat less than 80°F.

An attempt was also made to schedule an ASME performance test of the tower under the supervision of a Marley Company test engineer. This test is intended to yield detailed measurements of the performance characteristics and capabilities of the cooling tower. This test should be valuable in pointing out the most expeditious route to follow, both from cost and performance standpoints, to enable the cooling system to handle higher loads.

So far attempts to schedule this test have not been successful. The test engineer was available only when the reactor was scheduled to be shut down in July. Efforts are being made to reschedule this test as soon as possible.

d. Startup Feedwater Pump. As a result of increased pressure drop across the suction strainer of the motor-driven feedwater pump, the pump was shut down. Subsequent removal and inspection of the suction strainer revealed it was plugged from the residue of packing from the startup feed pump. These packing shreds had apparently entered the condensate supply line, passed through the condenser hotwell, the condensate system into the No. 2 heater, and finally to the feedwater-pump suction strainer. To stop this debris near its source, a "Y" strainer has been installed in the line from the lantern ring chamber of the startup pump to the condensate line.

To protect the pump and connected piping from severe vibration when the packing fails, a vibration switch has been mounted on the leakoff line from the lantern ring chamber. Actuation of this switch will unload the pump when packing failure causes the line to vibrate excessively.

e. Power Plant Piping. To limit pipe movement, sway braces have been installed on the turbine feedwater pump leakoff line.

f. Storage-basket Sense Rods. The storage-basket sense rods were deactivated and clamped in an out-of-the-way position late in 1964 as part of a modification to the system. An evaluation of this situation revealed that it is impossible to reactivate the sense rods for operation. The present status of the sense rods is also a potential hazard to the reliable operation

of the storage basket. Interference caused by inadvertent slippage of the sense rods could damage the storage basket sufficiently to require its removal of the basket from the primary tank for repair.

A proposal to remove the storage basket sense rods from the primary tank has been prepared and submitted for review.

7. High Power Operation

EBR-II was originally designed for full-power operation at 62.5 MWt. Many changes have been made in the original core, such as shortening of the active height, replacement of some driver fuel with experimental subassemblies having different fuel composition and geometry, and the replacement of the first two rows of the blanket with stainless steel-reflector subassemblies.

a. 50-MWt Proposal. Work on the proposal for operation at 50 MWt continued. A study to determine the effect of the power increase on allowable fission rates in experimental irradiation subassemblies has been completed. The increase in power to 50 MWt will require the movement of some experiments to positions of lower flux. The experiments affected are those whose original locations were predicated on specific fission or heating rates at 45 MWt. Because movement of the experiment affects the overall fission-rate distribution as well as reactor reactivity, it is necessary to arrive at a determination of final core configuration through a series of iterations. Such a series of calculations have been made for a core that will accommodate the present experiments in the reactor, plus several more of similar design. For 50-MWt operation, experiments in Row 4 can be moved to Row 5. Two experiments in Row 6 will require relocation to Row 7. At the present time, only structural experiments are in Row 2, and they may remain there to take advantage of the higher flux.

A complete calculation of axial radial temperature distribution within the driver-fuel subassemblies has been made for operation at 50 MWt by the modified HECTIC II program. The power distribution used was that resulting from the MACH-I physics calculations in which experiments were relocated to the 5th row from the 4th row.

b. High-fuel-temperature Experiments. A revised proposal for the irradiation of Mark-IA and Mark-IB and/or Mark-IA subassemblies with reduced flow (73% of full flow) was submitted for review by the Idaho Division Irradiations Review Committee. The subassemblies are to be irradiated in 5th row positions after the reactor power has reached 50 MWt. The Committee has recommended approval of this experiment. The 70% enriched fuel experiment is being reviewed by the Laboratory Reactor Safety Review Committee.

A safety evaluation of reactor operation at 50 MWt is nearing completion. This evaluation reviews those aspects of the power increase which conceivably could affect operational stability and considers the consequences of fuel element failure. The detailed results of this work will be incorporated in the proposal for operation at 50 MWt.

c. Improved Core Orificing. A Row-6 core-type subassembly with six inlet holes enlarged to 0.314-in. diameter has been fabricated and loaded into the primary-tank storage basket. It will be installed under the outlet thermocouple in reactor location 6C4 to investigate the subassembly outlet temperature in Row 6. The calibrated flow of sodium at 800°F with a pressure drop of 41 psi is increased from 73 to 90 gpm.

A Plant Modification Proposal is being prepared for reorificing the driver subassemblies to increase the flow in Rows 4, 5, and 6. The intent of the proposed modification is to provide more uniform temperature distribution. Irradiations of reorificed subassemblies to test the modification will be included in this proposal.

8. Fuel-element Development and Metallurgical Assistance

a. Anomalous Swelling of EBR-II, Mark-IA Fuel. The problem (see Progress Report for March 1967, ANL-7317, pp. 8-12) has been split into 11 subproblems and will be reported in this format.

(i) Irradiation of MC-S-type Fuel. This type of fuel is being irradiated in three matched sets of subassemblies in EBR-II. The three sets reached 44, 40, and 32% of the target burnup at the end of reactor Run 25.

(ii) Encapsulated Irradiation of SL-type Fuel. These fuel elements (in irradiation test subassemblies XO17 and XO15) reached approximately 60 and 30% of the target burnup at the end of reactor Run 25.

(iii) Effect of Design Change. Two subassemblies, each containing fuel elements of Mark-I and Mark-IA designs, reached 45% of the target burnup at the end of reactor Run 25. A third subassembly containing fuel elements of the Mark-I, Mark-IA, and Mark-IB designs is being assembled.

(iv) Effect of Enrichment. Three subassemblies, each containing fuel enriched to 48 w/o and to 52 w/o, reached 64, 44, and 37% of the target burnup at the end of reactor Run 25. The first of these (C-283, being irradiated in reactor grid position 1-A-1) will reach target burnup about midway through reactor Run 26.

(v) Effect of Position and Orientation in the Casting Bundle.

One subassembly containing fuel pins identified with respect to their position and orientation in the injection-casting-mold array reached 44% of the target burnup at the end of reactor Run 25.

(vi) Effect of Trace-element Impurities.

Argonne, Illinois injection castings from Runs 409 and 410 and the ingots from which they were prepared were reanalyzed for ruthenium. The reported ruthenium content is now very close to 2.00 w/o and well within the specification. The high results (2.50 w/o) previously reported (see Progress Report for June 1967, ANL-7349, p. 24) appear to have been an analytical error. Pins from castings 409 and 410 have been processed into fuel elements at Idaho and loaded in subassembly C-2077. Irradiation will start with reactor Run 26 in reactor grid position 3-B-2.

The silicon specification for Melt II which is to be made in the FCF at Idaho from virgin metal has been changed from 500 to 300 ppm to bring it in line with Melts III and IV. A fifth melt, containing only silicon as an intentional addition, has been added to the series. The current composition specifications for these five FCF melts are given in Table III.

TABLE III. Trace-element Specifications--FCF-produced Melts

Melt No.	Trace Element (ppm by wt)				Identification Melt No.
	Carbon	Silicon	Aluminum	Iron	
I	^a	-	-	-	EUF
II	250	300	-	-	EUF
III	250	300	250	750	EUF-42S
IV	250	300	750	250	EUF-46S
V	-	300	-	-	EUF

^aA (-) in the table means that no deliberate addition of trace element has been made.

Two ingots (EUF-42S and EUF-46S) have been cast and are being analyzed for the trace elements.

(vii) Effect of Pressure Upon Swelling. The first capsule (P-1) containing specimens for pressure studies was inserted 22 May 1967 and has been accumulating burnup at a rapid rate. Removal from the reactor is now scheduled for some time in September 1967.

The second capsule (P-2) was inserted 20 July 1967 and is not expected to be removed until December 1967, because of an enforced down time of several weeks for the reactor during October. The objective for the second capsule is to provide some compensation for variation in microstructure and to confirm findings for capsule P-1.

A third capsule is in preparation in which use is made of the upper half of a cast pin adjacent to that being examined in the two previous tests. Recent metallography has demonstrated the uniformity of this portion of the casting. Irradiation of these specimens should provide further confirmation of the pressure-swelling relation.

(viii) Effect of Heat Treatment. Specimens cut from Pin 56 of Melt 409 [see Sect. (vi) above] have been heat treated according to schedule (see Progress Report for June 1967, ANL-7349, pp. 26-27). Start of irradiation in CP-5 is scheduled for the last week in August 1967.

The effect of heat treatment on the structure and swelling behavior of U-5 w/o Fs alloy containing trace-element impurities will be carried out on pins cast from one of the EUF-type melts (see Table III) being prepared at Idaho. Melts II and III are more typical of reactor fuel currently being produced than was Melt 410 prepared at Lemont. This EUF-type material will be somewhat radioactive, and more difficult to characterize and encapsulate, but the results will be more directly applicable to typical EBR-II reactor fuel.

(ix) Hot-lab Examination of Irradiated Fuel. Aluminum analyses reported by ORNL are not always self-consistent, and specimens from some pins have been resubmitted. Specimens from another seven pins, not previously analyzed, have been sent to ORNL for iron and aluminum determinations. Preliminary results for carbon on five more pins have been obtained. No new silicon results have been obtained this month.

(x) Analysis of Fuel-pin Swelling. In the original analysis of the swelling-burnup data, the data were classified into three groups (SL, MR, and MR-C types) on the basis of the source of the fuel pins. However, machine plotting of all of the available data revealed that: (1) the SL data fit a single smooth curve but with considerable scatter, (2) both the MR and MR-C data split into two groups at burnups beyond about 1% maximum burnup. In other words, while all of the SL-type fuel can be classified as high swelling, not all of the MR-type is high swelling nor is all of the MR-C type low swelling. A new criterion is therefore needed for grouping the FCF-produced fuel. All casting batches for which data were available from several different subassemblies and over a range of burnup were studied separately, and an attempt was made to develop a figure of merit that would be simultaneously independent of the burnup and would clearly distinguish high-swelling from low-swelling material. Of the six ratios examined, the most consistent seemed to be $Y/(Bu)_{\max}^2$, that is, the quotient consisting of the percent swelling Y divided by the square of the maximum percent burnup, Bu . This figure of merit is now used to distinguish high-swelling from low-swelling material rather than the somewhat arbitrary MR and MR-C designations (see Table IV).

TABLE IV. Classification on Basis of Swelling Behavior

Figure of Merit ($Y/(Bu)_{max}^2$)	Classification
<5	Low Swelling
5-7	Moderately Low Swelling
7-10	High Swelling
>10	Very High Swelling

melt) and 4032 (a MR-C melt). Pins from these casting batches are to be analyzed at Argonne, Illinois.

With this criterion for a figure of merit and the available chemical analyses, one can establish a tentative correlation between swelling and the silicon content, as seen in Table V. With perhaps two exceptions, the correlation is quite good in that if the silicon content is less than 200 ppm by wt, the swelling is high; if it exceeds 200 ppm by wt, the swelling is moderately low or low. The two exceptions are casting batch 4027 (a MR

TABLE V. Swelling Behavior and Trace Element Content

Casting Batch	Melt Refining No.	Figure of Merit ($Y/(Bu)_{max}^2$)	Trace Elements (ppm by wt)				Source of Analysis ^a
			Carbon	Silicon	Iron	Aluminum	
4001	22	8.91	107	88	470	295	IL & OR
4002	23	9.89	135	78	173	125	
4003	24	6.17	201	205	455	173	
4004	25	9.60	108	140	435	125	
4008	30	9.89	(237)	71	160	205	ID
4022	C-30	9.15	(207)	180	340	160	
4025	C-33	5.14	(237)	210	260	160	
4026	C-32	5.95	(222)	219	430	175	
4027	46	10.18	-	205	285	77	IL & OR
4032	C-36	6.16	238	150	375	161	
4033	C-34	4.34	204	243	365	153	
4035	43	4.36	178	250	665	121	
4036	C-40	4.33	188	270	330	20	ID
SL-20	-	8.67	146	88	148	225	
SL-21	-	8.62	153	87	155	200	
SL-29	-	9.86	148	109	153	160	
SL-30	-	10.60	217	108	255	140	ID
SL-42	-	11.58	-	78	193	243	

^aIL = Illinois for carbon and silicon; OR = Oak Ridge for iron and aluminum; ID = Idaho for carbon, silicon, iron and aluminum.

b. Mark-II Driver-fuel Development. Fifty-five Mark-II fuel elements have been encapsulated and sodium bonded. They have passed the X-ray and eddy-current tests for bonding, and have had tip extension sections welded to them. Of these, 37 were selected for the initial loading of a subassembly with specific reference to restrainer levels and sodium levels for varying degrees of fuel growth and pressure generation for evaluation at the earliest possible time, namely, one percent burnup, as well as at later intervals. Also, a replacement schedule for substitution of capsules after 1, 2, 3, 4, and 5 a/o burnup has been prepared.

The first group of 37 encapsulated Mark-II elements has been shipped to Idaho for assembly in a B-37 irradiation subassembly. An addendum to the Description and Safety Evaluation was completed and forwarded July 12, 1967.

Eighteen capsules need tip-extension sections to be welded to them. Twelve capsules need X-ray and eddy-current tests for bonding in addition to the welding of tip extensions.

9. Reactor and Primary Coolant System Equipment

a. Primary Cold Trap. All materials for this job are on hand; and all prefabrication of piping practical prior to installation has been accomplished.

Current plans are to install the modified cold trap and purification system during reactor operation after the new plugging meter is installed in the FERD loop. In this manner, the quality of the primary sodium can be monitored while the purification-system plugging meters are inoperative.

b. Primary Tank Pressure Relief. Analysis of the Primary Tank Pressure Relief system continued. A gas-storage tank attached to the discharge side of the pressure-vacuum relief device is required. This tank filled with argon would have two lines with check valves attached to it. One check valve would allow reactor building pressure to equalize inward to the primary tank while preventing reverse flow of radioactive gas into the building. The other check would allow relief of increased pressures in the primary tank to a system, such as the suspect exhaust system.

A tentative size of the gas-storage tank was estimated by assuming the reactor building atmosphere reaches its design maximum pressure of 24 psi. The tank must also contain sufficient argon to flow into the primary tank cover gas and equalize this pressure. To satisfy this criteria, a tank with a volume of about 225 ft³ is required.

Some other changes recommended for the primary tank relief system are as follows:

(i) Change the present +12 and -4 in. of water pressure limits to +12 and -5.65 in. This will keep the pressure limits within those of the floating head tank oil seals but minimize changes of inadvertent inward flow of air or gas into the primary tank.

(ii) Decrease the high-reactor-building-pressure alarm from the present +10 in. of water pressure down to +5 in. This would then give an alarm indication prior to activation of the primary tank relief system.

c. Engineering Consultation

(i) A detailed cost estimate is being prepared for the installation of the new secondary EM pump. The cost estimate is needed as part of the Davis-Bacon determination of this work.

(ii) Consultation was held with the Fast Breeder Reactor Study Group from the Edison Electric Institute to discuss the status, performance, problems, and plans for EBR-II as part of their overall study of fast reactors.

(iii) Consultation was held with members of the LMFBR Program Office Instrument and Control Section on the design, schedule requirements and capabilities of the new high-temperature nuclear instrument test facility planned for EBR-II.

d. Instrumentation. In reviewing the drawings for the trip amplifier for the replacement nuclear equipment and postulating potential hazards, it has been determined that a high-voltage spike (for example, lightning) could possibly enter the equipment and not necessarily be suppressed in the filtered power supplies. Failure of a control transistor, so that the control relay in the reactor protective circuit would not open, could then occur.

Because this mode of failure does exist, however remotely, a zener and quick-acting fuse have been added to the power-supply outputs and to the individual trip amplifiers for added protection. The zeners and fuses have been sized to limit the current to 60% of rated normal capacity and the voltage to 65% of rated breakdown voltage (PIV). The fuse will be greater than normal load current by a factor of three. The zener operating voltage will be approximately 120% of normal.

e. General Improvements

(i) FERD Loop Plugging Meter and Sampler. In prooftesting the plugging-meter flow components and associated instrumentation in the secondary sodium system, a drift in flow rate has been detected. This drift has been traced to a change in circuit impedance, specifically a change in reactance of the voltage step-down transformer. The reactance change has been isolated and identified as due to the temperature distribution within the transformer. Compensating components have been added, improving overall stability. Additional testing has been scheduled prior to final acceptance of the circuit.

Material necessary for this plant modification has been assembled and work has started on the panel fabrication.

(ii) New Plugging Meter for the Primary and Secondary Purification Systems. The design of an improved plugging meter for the primary and secondary purification systems has been completed. Installation drawings are being prepared. One plugging meter will be installed in the secondary purification system and two in the primary purification system.

(iii) Storage Basket. The rotational alignment of the storage basket with the transfer arm has been time-consuming in the past. A micrometer-type positioner and a position indicator have been designed that will greatly reduce the amount of time and manpower required for adjustment as well as improving the accuracy of alignment. The latter will give a direct readout of Row 3. Both items are being fabricated.

(iv) Primary Argon-purification System. Pressure gauges PI-572 and 573 were calibrated and the manometers removed. A safety valve, set to relieve at $1\frac{1}{2}$ psig, was installed in each argon blower box to replace the manometers which were acting as safety valves.

(v) Purification Cell and Sampling Cell Ventilation. An investigation was made to determine the reason for less-than-rated flow through the exhaust duct of the primary purification cell. Use of a test pressure gauge across the filter shows a very slight pressure drop. Therefore, although the filter has not been changed for sometime, it is not restricting the flow.

A proposal for plant modification was prepared to cover changes recommended for the ventilation system in the sampling cell. A new duct penetration should be made so that the temporary supply duct can be removed from the doorway. The exhaust ducting for the purification cell and inlet screen inside the cell will be checked for cleanliness the next time the cell can be entered.

(vi) Control-rod-drive Lifting Platform. Redesign of the mechanical interlocks for the control-rod-drive lifting platform has been initiated to improve reliability and reduce the required maintenance effort. The gears to change the travel of the platform locks have been received and will be installed when design is completed and the new locks are fabricated.

10. Secondary Sodium and Power System

a. Relief Valve for Main Stream Header. A study of the main stream header has indicated the desirability of relocating the new relief valve to a position in the Sodium Boiler Plant. The new location will enhance the quick dumping capabilities of the steam-generating system. Drawings for this change have been completed and are ready for review.

b. Inspection of Steam Generator. Discussions were held with a representative of the Gray Tool Company, Houston, Texas manufacturer of Grayloc Couplings. Information available to date strongly indicates that the Grayloc Couplings, properly applied, may be the best method of providing for occasional removal of the steam-generator riser. Preliminary design of a riser spool piece with these couplings has been initiated.

c. General Improvements. Work continued on several modifications during this period. The installation of a check valve in the interconnection of the 150-lb steam line with the 175-lb line was completed. This will prevent the flow of steam to the 175-lb (auxiliary steam) system in the event of overpressurizing of the plant 150-lb system.

Approval was received to proceed with the installation of automatic trips on the turbine feed and condensate pumps, and to provide for automatic tripping of the blowdown valve P5-VC-587. Design and procurement for this work are simultaneously underway.

Approval to proceed with moving the controller for No. 2 heater auxiliary steam from the first floor of the Power Plant to the control room was also received. Work on this modification is now underway. Both plant changes will enhance ease and dependability of plant operations during periods of unusual operational circumstances.

Design work for installing a new relief valve on the No. 2 heater is nearly ready for final submittal for approval and is awaiting completion of drawings. Design for installing a drain valve downstream of the feedwater line check valve is similarly awaiting completion of drawings.

Specifications for the new startup feedwater pump, WP 1723, were completed and bids for this machine will be invited.

11. Process Chemistry

a. Quality Monitoring and Control of Sodium Coolant

(i) Sampling of Primary Sodium. The waste sodium container in the purification cell became completely filled with sodium at the end of June. It was isolated from the sampling equipment, and another waste container was fabricated and installed in the primary sodium sampling room. During this installation, the sampling system was out of service. The new waste container has been shielded with lead brick. Installation was completed and the sampling system was placed back in service. During the period of changing waste containers, modifications were made to permit installation of a flow-through sampler between the PI-A (inlet to cold trap) and PI-B (outlet from cold trap) sample lines. This arrangement permits a primary sodium sample stream to flow through the sampling vessel under the driving force of the pressure drop through the primary cold trap and piping. With the revised arrangement, the sample stream flow must be stopped while plugging temperature determinations are being made. An extrusion vessel with improvised end fittings was used for the first flow-through sample taken.

(ii) Analysis for Carbon in Primary Sodium. The analytical equipment for carbon determination has been inoperative over the past four weeks due to high blanks from the analytical train. All efforts to discover and eliminate the source of these blanks have failed. Work is continuing on stepwise replacement of combustion tubes and purification tubes, and on leak testing and blank runs. Very pure quartz tubing, free drawn with very low carbon, has been ordered from two different vendors. The tubing now in use will be replaced with the new tubing if other efforts to eliminate the high train blank continue to fail.

The first flow-through sample of primary sodium in an extruder vessel was taken in July. This is the extruder vessel with improvised end fittings utilizing metal O-ring seals (see Progress Report for June 1967, ANL-7349, p. 83). This sample was not rapidly cooled.

(iii) Segregation of Radiocesium in Samples of Primary Sodium. Cesium segregation during cooling and/or freezing of primary sodium samples has been shown to occur (see Progress Report for June 1967, ANL-7349, pp. 37-39). Approximately 75% of the cesium was found in a layer of sodium ~0.002 in. thick on the walls of the sample vessel. To determine whether the segregated cesium would redistribute during remelting and subsequent heating, a sample of primary sodium was heated and held at temperature of ~2 hr. The sodium was poured from the cup and the $\text{Cs}^{137}/\text{Na}^{122}$ ratio determined in the bulk sodium and the sodium dissolved from the wall of the cup. Approximately 93.5% of the cesium was found in the bulk sodium poured from the cup, with 6.5% left with the sodium on the wall of the container. Verification of the observed redistribution as a function of time and temperature is planned.

(iv) Analysis for Oxygen in Primary Sodium. A special glovebox for the analysis of oxygen in primary sodium has been constructed and is being installed. Oxygen analyses have not been run by the amalgamation method pending installation of this glovebox.

A cyanide reduction method for the determination of oxygen in sodium is being investigated. Reaction products from two different runs have been sent to CPP for mass-spectrometric analysis.

(v) Analyses for Copper in Primary Sodium. The following analyses were provided by the ID Chemistry Laboratory:

<u>Sample Date</u>	<u>Time</u>	<u>Sample Point</u>	<u>Copper (ppm)</u>
6/28/67 (b)	1455	PI-A ^(a)	0.32
7/18/67	1400	PI-A	0.64

(a) Sampled from cold-trap inlet piping.

(b) Routine weekly samples could not be taken.

(vi) Fission Products in Primary Sodium. During Run 25 Cs^{137} was found in the primary sodium between April 24 and May 8 (see Progress Report for June 1967, ANL-7349, pp. 40-41). The Cs^{137} concentration as of May 19, 1967 was $1.5 \times 10^{-2} \mu\text{Ci/ml}$ and has remained constant since then. Since the mass-134 and -136 decay chains are both blocked by stable xenon isotopes, sodium samples collected prior to and following the May 24 fission product release were analyzed for Cs^{134} and Cs^{136} . Both cesium isotopes were identified after the release but were nondetectable prior to the release. The presence of these isotopes following the release can be explained only by direct volatilization and/or diffusion of cesium, rather than resulting from short-lived precursors, from the fuel into the bond sodium.

Following the fission product release on June 28, the primary cover gas was analyzed for Xe^{133} , Xe^{135} , and Kr^{85m} . The data, corrected for decay to release time, are given in Table VI.

TABLE VI. Fission Gas Release of June 28, 1967.

Sample Date	Time	Xe^{133} ($\mu\text{Ci/ml}$)	Xe^{135} ($\mu\text{Ci/ml} \times 10^3$)	Kr^{85m} ($\mu\text{Ci/ml} \times 10^3$)
6/28	0850	0.054	2.6	Not found
6/28	1445 ^a	0.33	5.9	10
	1515	0.36	4.5	12
	1550	0.35	4.1	11
	1700	0.31	2.5	11
	1850	0.37	3.7	10
	1943	0.37	3.9	11
	2135	0.37	3.9	9.1
	2200	0.37	3.8	12
	2243	0.38	3.9	10

^aAll values following the releases are corrected for decay to release time.

On July 18, 1967, a sample of argon cover gas was found to contain 1.2×10^{12} tritium atoms per liter of argon gas. Since the cover gas was renewed just prior to June 19, this tritium value represents the tritium discharged to the cover gas as the result of 412 MWd of operation plus the fission gas releases of June 19, and June 28, and should represent, therefore, an upper limit for tritium discharge to the cover gas under normal operations.

(vii) Primary Sodium Plugging Temperatures. Through July 20, the measured plugging temperatures of the primary sodium were all in the range from 220 to 240°F, with the bulk of the values around 225°F.

(viii) Primary Argon. Data from the recorder chart of the continuous gas chromatograph are summarized in Table VII. The gas chroma-

TABLE VII. Analysis of Primary Argon by Continuous Chromatograph

	Hydrogen (ppm)	Oxygen (ppm)	Nitrogen (ppm)
High	172	0	8,400
Low	4	0	4,000
Average	10	0	6,400

tograph shows increases in hydrogen content of the cover gas associated with operations involving the fuel-transfer port. Seven such increases were observed during the report period.

Routine measurements of Ar^{41} , Xe^{133} , and Xe^{135} activities in the primary argon continued.

(ix) Secondary Sodium Sampling and Analysis. The sampling equipment operated with only minor difficulties. Galling of the threaded fitting on the "Conoseal" fitting over the extrusion vessel chamber in the oven continues to be a problem. A filter was installed in the argon line to collect sodium vapor in an effort to solve the problem of sodium plugging the argon lines. The detector probes have not been installed for the oxygen analyzer because the cells cannot yet be maintained at the desired temperature.

A sample of secondary sodium taken on 4/5/67 was analyzed by the Chemistry Division (ANL-Illinois) for oxygen and hydrogen contents. Duplicate determinations for oxygen gave 9 and 6 ppm; for hydrogen, 3.5 and 4.8 ppm. Secondary sodium plugging temperature was maintained at about 225°F during this month.

(x) Secondary Argon. The hydrogen and nitrogen contents of the secondary cover gas sampled from the surge tank remained fairly steady. Table VIII summarizes readings from the chromatograph recorder chart.

TABLE VIII. Analysis of Secondary Argon by Continuous Chromatograph

	Hydrogen (ppm)	Nitrogen (ppm)
High	25	2,600
Low	8	1,600
Average	12	2,000

(xi) Expanded Sampling Capabilities for Primary and Secondary Sodium

(a) Secondary Sodium Sample Station and Oxygen Meter Loop. Checkout and debugging of the sample station were completed. Minor

difficulties with valve operators and plugged lines were corrected. Sodium containers for FCF were filled routinely, and sodium samples were taken in both extrusion vessels and aluminum tubes.

With the installed heater capacity of the oxygen meter loop, it was not possible to reach the recommended operating temperature of 600°F for the electrochemical oxygen detectors. To correct this deficiency, a regenerative heat exchanger was designed and fabricated for installation in the flow system of the detectors. A control system for loop heaters is being assembled which will control detector sodium temperature at $600 \pm 2.5^\circ\text{F}$ (the manufacturer recommends control within $\pm 5^\circ\text{F}$). Also, a recorder is being provided to record simultaneously oxygen-detector output and cell temperature to facilitate correction of indicated oxygen. Installation of the electrolyte in the detector housing and testing of the oxygen meters is planned after these modifications are completed and checked out in August.

Heater cable used on the sample station and oxygen meter lines is rated at a lower temperature than that experienced during loop operation. Tubular resistance heaters, Inconel clad, have been procured in anticipation of heater cable failure. Heaters on the oxygen meter loop will be replaced when the regenerative heat exchanger is installed. Others will be replaced if they fail.

(b) Sodium Sampling Devices and Hardware. The extrusion vessel presently used as an overflow sample cup in the secondary sample station is being modified to (1) provide a flow-through sample cup, and (2) facilitate rapid freezing of the sample. As a test of the flow-through principle, an extrusion cup has been installed in the primary sample station using metal O-rings to seal between the cup and end flanges. A sample was successfully obtained, but no inspection or analysis has been performed as yet.

A double-wall extrusion vessel which will permit forced cooling of the sample has been fabricated. End-closure flanges are being redesigned to adapt the double-wall vessel as a flow-through sampler. Reusable "chevron type" metal seal rings for this purpose have been received.

A storage vessel for liquid nitrogen, capable of pressurized service, is being fabricated. Liquid nitrogen will be forced from the vessel through the outer annulus of the double-wall extrusion vessel to quench the sodium sample. It is anticipated that nitrogen will not reach the extrusion vessel in the liquid state; however, the flow of cold gas through the cooling jacket should freeze the sample in a few minutes. The quench system will be thoroughly tested prior to installation in the sodium system. Completion of tests is anticipated next month. Installed in the primary and secondary sample stations, the sampler will be used to study the effect of cooling rate on the segregation of impurities in the sodium.

Modifications to the vacuum distillation samplers are in progress. The sample cup has been changed to eliminate the foil insert previously used to contain the distillation residue. This, in addition to changes in the configuration of the distillation cup heater and thermocouple, will provide better heating and temperature-monitoring capabilities during distillation. It is planned to reinstall the distillation sampling equipment with an all-metal vacuum system (the original system used glassware extensively). To facilitate this change, Pirani-type vacuum gauges have been ordered as replacements for McLeod-type gauges. These changes will eliminate or greatly reduce the former problems with incomplete distillation which were attributed to poor heating of the distillation cup and difficulty in maintaining proper vacuum.

(xii) Analyses of EBR-II Seal Trough Alloy Samples. Because of intervening commitments of higher priority, the chemical analyses of samples taken in June have not been completed as yet.

b. Water Treatment Monitoring and Control.

(i) Treatment Equipment. Control of pH in the condenser cooling water during the month was by manual control of acid added at the tower suction bay. The polyethylene tubing for the temporary acid line was received but has not been installed, pending the receipt of tubing fittings. The sulfonator, using sulfur dioxide to reduce the chromate ion in the cooling water blowdown, was operated on a test basis during the month. The data show that about 5 lb SO_2 are required to reduce one pound of CrO_4 . The pH of the treated water is roughly 3.4. Two representatives of Betz Laboratories, suppliers of the equipment, visited the plant and made some suggestions regarding the operation of the unit and results to be expected.

c. Carburization Potential Measurements with EBR-II Sodium. The literature is being reviewed to obtain information enabling the specification of the method to be used for EBR-II sodium work. A glovebox was selected for use in this work and procurement of equipment for insert atmosphere maintenance and other glovebox appurtenances was initiated. The location of the glovebox in the FCF and the space requirements are not yet decided.

12. Experimental Irradiations and Testing

a. Status of Experiments in EBR-II. The status of all experimental irradiations in EBR-II as of the end of July is shown in Table IX.

Two subassemblies, XO23 and XO24 (Mark B-7 type), were irradiated for 28.7 MWd of reactor operation. These subassemblies, installed in Row 2 and Row 7, contained capsules provided by ORNL for an experiment

TABLE IX. Status of EBR-II Experimental Irradiations, July 31, 1967

Subassembly	Date Loaded	Capsule Content and Number of Capsules ()		Experimenter	Approximate Accumulated Exposure (MWd)	Goal Exposure (MWd)
XG02	7/16/67	UO ₂ -20 w/o PuO ₂	(1)	GE	11,297	13,650
		Stainless Dummies	(18)			
XG03	7/16/67	UO ₂ -20 w/o PuO ₂	(2)	GE	11,297	19,450
		Stainless Dummies	(17)			
XG04	7/16/67	UO ₂ -20 w/o PuO ₂	(2)	GE	11,297	39,000
		Stainless Dummies	(17)			
XG05	9/3/65	UO ₂ -20 w/o PuO ₂	(9)	GE	10,720	14,750
		U-15 w/o Pu-10 w/o Zr	(1)	ANL		
		U-15 w/o Pu-10 w/o Ti	(1)	ANL		
		UC-20 w/o PuC	(3)	ANL		
		Structural	(5)	GE		
XA08	12/13/65	UC-20 w/o PuC	(8)	ANL	8,898	19,600
		Structural	(11)			
XO10	3/24/66	UO ₂ -20 w/o PuO ₂	(4)	GE	8,228	19,600
		Structural	(11)	ANL		
		Structural	(4)	PNWL		
XO12	8/10/66	UO ₂ -20 w/o PuO ₂	(19)	NUMEC	7,528	20,600
XO15	11/15/66	PuO ₂ -UO ₂	(11)	NUMEC	2,873	11,000
XO16	1/13/67	Structural	(9)	ANL	1,553	3,000
		Structural	(10)	GE		
XO17	11/15/66	PuO ₂ -UO ₂	(11)	NUMEC	2,873	6,500
		(Pu, U)C	(3)	UNC		
		Mk-IA (Metal)	(5)	ANL		
XO18	12/6/66	Structural	(3)	GE	2,183	21,300
		Structural	(1)	PNWL		
		Structural	(2)	ANL		
		Structural and Heavy Metal Fission Yield Samples	(1)	ANL		
XO19	1/13/67	UO ₂ -20 w/o PuO ₂	(7)	GE	1,553	12,500
		(U _{0.8} Pu _{0.2})C	(3)	UNC		
		Structural	(8)	PNWL		
		Graphite	(1)	PNWL		
XO20	1/13/67	Mixed Oxides	(9)	GE	1,553	25,000
		Mixed Carbides	(3)	UNC		
		Structural	(4)	PNWL		
		Structural	(2)	ANL		
		Graphite	(1)	PNWL		
XO21	2/27/67	Structural	(7)	PNWL	1,553	21,500
XO22	2/27/67	Structural	(7)	PNWL	1,553	18,000
XO23 ^a	7/21/67	Materials for Gamma Heat Test	(7)	ORNL	28.7	30
XO24 ^a	7/21/67	Materials for Gamma Heat Test	(7)	ORNL	28.7	30

^aSubassemblies loaded at end of Run 25 for a short exposure run at 30 MW to measure nuclear heating in Row 2 and Row 7. They were removed on July 23 and therefore were not in the reactor at month's end.

whose principal objectives are to measure the axial distribution of gamma heating and axial neutron fluxes in reactor locations to be utilized for materials irradiations. After irradiation the subassemblies were removed from the reactor and transported to TAN for disassembly. A representative of ORNL was present during the work at TAN.

Two Mark-A type experimental irradiation subassemblies, designated XO27 and XO30, and two Mark-B type subassemblies, XO25 and XO26, were assembled, flowtested, and will be installed in the reactor during the loading change for Run 26. The makeup of each of these subassemblies is as follows:

XO25 (Mark-B-19 type; Reactor Location 4E2)

19 structural materials capsules from General Electric Company

XO26 (Mark-B-7 type; Reactor Location 7D5)

7 structural materials capsules from Naval Research Laboratory

XO27 (Mark-A type; Reactor Location 4E3)

16 encapsulated mixed oxide fuel elements from GE (GE Group ElH);
3 structural materials capsules from Pacific Northwest Laboratory

XO30 (Mark-A type; Reactor Location 6E1)

2 capsules containing candidate FFTF thermocouple materials (Pacific Northwest Laboratory);
1 structural materials capsule (PNL);
3 capsules containing graphite (PNL);
13 stainless steel dummy capsules.

The GE Group ElH in XO27 comprised 19 capsules. However, two capsules were rejected on the basis of radiographic results (see Sect. II.A.12.d.iv). A third capsule was rejected because of a surface defect in the capsule tubing. The fuel element in this capsule also contained chips of fuel pellet material. The condition of the fuel in ElH capsules was reported to GE and to AEC-RDT. Following review of radiographic inspection and discussions with a GE representative, the EBR-II Irradiations Review Committee judged the remaining 16 capsules to be acceptable for irradiation.

Subassembly XO30 has an indefinite exposure goal. It will be installed on a space-available basis until such time as its reactor location may be required for an experiment of higher priority or until the sponsor requests removal.

b. Procurement and Inventory Control of Experimental Hardware. Tip elements and end plugs for the Mark-B-61 and Mark-B-7 experimental capsules were received from Illinois. The remaining experimental capsule hardware being transferred from Illinois is expected in Idaho during August. Also, 13 graphite-containing capsules were received from Pacific Northwest Laboratory, and 37 Mark-II fuel-bearing capsules were received from Metallurgy Division.

c. Other Work. Hardware for the Mark-B-37A prototype irradiation subassembly, which will accommodate unencapsulated mixed-oxide fuel elements (of 0.250-in. OD) of GE and Pacific Northwest Laboratory, is available except for the grid to which the elements are attached. So as not to delay the obtaining of flow-test results, the prototype subassembly will be assembled with a grid of the B-37 design. This will be satisfactory to obtain flow-test data representative of B-37A subassembly installation in Rows 1 through 8. It is expected that flow testing will be performed early in August.

d. Nondestructive Testing

(i) Antimony-Beryllium Neutron Source. The antimony oxide gamma source in EBR-II has now been activated by power run (Run No. 25). At the end of Run 26 this source will be removed from the reactor for a proof-test of the neutron source.

The criticality review for installing the source in the FCF Argon Cell is in progress. Additional information on the neutron-flux intensity and energy distribution was supplied to the Idaho Division Criticality Representative. A decision is expected in August.

Construction of handling fixtures and other needed hardware for the neutron radiography facility in FCF has not been started because of delay in the availability of construction drawings. It is estimated that three months from receipt of drawings will be required for completion of the facility.

A sample of essentially pure antimony tetraoxide (see Progress Report for June 1967, ANL-7349, p. 50) was heated to approximately 900°C in a 304 stainless steel capsule. The capsule remained at temperature for 6 hr, and was then cooled and cut open. There was a considerable change in the physical appearance of the material. The bulk of the material was in a solid plug which occupied about 2/3 of the original volume. The plug had a black coating, but the material in the center was white. There were small metallic beads on the side of the capsule and whisker-like crystals on both the capsule wall and the outside of the plug. There was no evidence of overly severe attack of the stainless steel capsule, nor was there any visible reduction in the capsule wall thickness.

X-ray diffraction and spectrochemical analyses were performed on the substances exhibiting different physical forms. The results are as follows:

1. Metallic beads--Antimony-nickel alloy
2. Whisker-like crystals--Antimony trioxide (Sb_2O_3)
3. Black coating on plug-- Sb_2O_3 and Fe_3O_4
4. White material from center of plug-- Sb_2O_3

It appears that the antimony tetraoxide was largely reduced to the trioxide under the test conditions.

The major purpose of the capsule test was to determine whether deleterious reactions between the antimony oxides and stainless steel are liable to occur. The test temperature (900°C) was considerably greater than the calculated maximum temperature of an antimony oxide-stainless steel source in Row 7 of EBR-II (namely, about 450°C). To provide additional assurance of the integrity of these sources in EBR-II, an oxide source which was irradiated for about 1200 MWd of reactor operation in Row 8 will be destructively examined in FCF or TAN. This examination will reveal the condition of the oxide (originally antimony tetraoxide) and the stainless steel after irradiation at temperatures around 800°F .

Hardware is available for the fabrication of four additional antimony oxide-stainless steel sources. These, together with the source presently undergoing irradiation, will be used in the FCF neutron-radiography source.

Plans call for the fabrication of a source-storage thimble for Row 7 of the reactor to permit source irradiation in this location.

(ii) Evaluation of Isotopic Neutron Source. Negotiations continued with General Atomics concerning use of their GAM-II program on their computer to generate group cross sections for an accelerator source of neutrons.

(iii) Other Nondestructive Tests. The pulsed eddy-current system has been completed. Work on the tubing conveyor is in progress. Delivery of the conveyor is expected in August.

(iv) Inspection of Experimental Capsules. General Electric Group-ElH fuel capsules (see Sect. II.A.12.a) were radiographed. The radiographs showed that some movement of fuel had occurred in the fuel elements contained within the capsules. It was concluded that inadequate packing and possible rough handling during shipment were the causes for

the movement. The fuel separation was largely corrected, with the consent of a GE representative, by dropping the capsule from a 1-in. elevation onto a surface of such material that damage to the capsule end piece could not occur.

One capsule from this group had a surface defect whose depth exceeded 10% of the specified capsule wall thickness, and two capsules had lower-end welds that showed poor penetration. These were rejected for irradiation.

Eleven of the thirteen graphite capsules from Pacific Northwest Laboratory were not accepted for irradiation because of defective welds between capsule wall and end piece.

(v) Examination of Irradiated Capsules from Subassembly XO11.

A work plan and schedule for examination of the capsules from Subassembly XO11 are being formulated. Disassembly will be by the Vertical Assembler-Dismantler (VAD) in the FCF Air Cell. Neutron radiography in the TREAT facility is also being considered.

Detail drawings are being made for the shield and handling equipment necessary for the neutron radiography at TREAT. A high priority has been given to the fabrication and installation of equipment for this purpose.

e. Handling and Examination

(i) Materials Surveillance Subassembly. Flux wires from the first EBR-II materials surveillance subassembly (designated SURV-1), removed from position 12B1 after 11,541 MWd of irradiation, were cut into 1-in. sections and analyzed by gamma spectrometry. The neutron reaction for each of the wires is given below:

Copper Wire: $\text{Cu}^{63}(\text{n},\alpha)\text{Co}^{60}$

Iron Wire: $\text{Fe}^{54}(\text{n},\text{p})\text{Mn}^{54}$

Nickel Wire: $\text{Ni}^{58}(\text{n},\text{p})\text{Co}^{58}$

Titanium Wire: $\text{Ti}^{46}(\text{n},\text{p})\text{Sc}^{46}$

Metallographic examination has been completed on 10 hardness samples from SURV-1. Work is progressing on the metallographic examination of the unirradiated control samples of the material similar to that loaded into SURV-1. Macrohardness and microhardness readings have been taken on several samples but the data have not yet been compiled. Spring rate measurements have been made on the 18 Inconel-X spring samples and the data are presently being processed.

Shipment No. 1 to ANL, Illinois containing the tensile, impact, and bend-test specimens plus the lower adapter and a section of hex tube from SURV-1 has been completed. Shipment No. 2 is presently being prepared and should be shipped within two weeks.

(ii) Testing of Procedures for Subassembly Sodium Removal

(a) Metallographic Examination of Capsule Wall Material (Type 304 Stainless Steel) from Capsules Irradiated in Subassembly XG06.

The metallography, which was to be performed by GE, has not been started because of intervening work of higher priority and work previously scheduled ahead. In order to avoid undue delay in obtaining this information which is needed in support of the planned EBR-II interim examination and reirradiation program, specimens of capsule tubing have been requested from GE. These specimens will be subjected to metallographic examination at the EBR-II site.

(b) Cyclical Testing of Types 304L and 304 Stainless Steel Tubing Exposed to Sodium at 1100°F, Washed in Water, and Stored in Air.

This work consists of four phases. The first phase, which was completed in May, provided sensitization test data for stainless steels exposed to sodium and then subjected to one cleaning operation. The remaining phases involve further exposure to sodium of previously washed stainless steels, with determination of the effects of different storage periods in air. The data and observations from these tests may or may not be representative of effects of similar handling at EBR-II. The applicability of these results to EBR-II plans for interim examination and reirradiation is subject to further study.

Observations and data from Phase II were completed in June. Selected tentative conclusions from this phase of the program are as follows:

1. A single 10-day storage period in air of samples exposed to sodium at 1100°F has small adverse effects on 304L stainless steel.
2. Re-exposure to 1100°F sodium of 304L samples previously exposed and stored (10 days in air) produces significant changes in grain structure. Microstructures show highly sensitized grain boundaries.
3. Data from Strauss-reagent and electrical-resistivity tests indicate that 304L becomes very susceptible to intergranular attack to a depth of about 0.004 in. after completing a sodium exposure-washing-air storage-sodium exposure cycle.

The sodium used for these tests is being chemically analyzed. Phase III of the testing program has been completed and a report is being prepared.

Comparisons will be made among the as-received, Phase-I, Phase-II, and Phase-III conditions for ultimate and yield strengths, microhardness and ductility. The mechanical tests are in progress.

(c) Cyclical Testing of Stressed Capsules (Type 304L Stainless Steel at EBR-II). This work involves metallographic comparisons of as-received Type 304L stainless steel tubing with samples from pressurized capsules which will be exposed to EBR-II primary sodium, cleaned by standard EBR-II procedures and equipment, stored in air, and returned to the primary tank. Four cycles, into the storage basket in the primary tank and out again, are planned, followed by a fifth cycle which will involve installation of the capsules in Row 7 of the reactor for exposure up to 6×10^{21} nvt. Specimens will be examined after each wash cycle. Some capsules have been pressurized to 730 psig at room temperature; the calculated hoop stress in these capsule walls during the washing operation will be about 3680 psi. Other capsules were pressurized to 1500 psig; their hoop stress will be about 7560 psi. The remaining capsules were not pressurized. The first sodium exposure is planned in August.

(d) EBR-II Subassembly Recycle Test. To provide further information about the behavior of stainless steel cladding which has undergone repeated cycles into sodium and through the standard EBR-II washing procedure, a test is being considered which would involve recycle of an irradiated driver subassembly of about 1.2 a/o fuel burnup through five cycles of sodium exposure and washing. After the last washing, the subassembly would be dismantled and some number of elements would be examined for evidence of deleterious effects. This examination would involve metallography of cladding specimens.

f. Safety Review of Experiments. During July the following experiment proposals were reviewed:

(i) a proposal from Naval Research Laboratory (NRL) for irradiation of seven capsules of structural materials in a Mark-B-7 experimental subassembly--XO26;

(ii) a proposal from General Electric Company for irradiation of 19 capsules of mixed-oxide fuel, Group ElH, in a Mark-A experimental subassembly--XO27;

(iii) a proposal from ANL-ID to irradiate a driver subassembly containing modified driver-fuel alloy, the modifying elements being silicon, aluminum, and carbon;

(iv) a proposal from ANL-ID to irradiate 5 experimental driver subassemblies in which the coolant flow is orificed to simulate operation at 62.5-MW reactor power;

(v) a proposal from Pacific Northwest Laboratory (PNL) to irradiate six materials capsules in a Mark A subassembly--XO30.

Two capsules contain candidate thermocouple materials for FFTF: one contains structural materials, and three contain graphite. The remaining 13 locations in the subassembly are occupied by stainless steel dummy capsules.

13. Fuel Cycle Facility

a. Operations

(i) Hot-line Operation. Routine operations were carried on throughout the month. To supplement the irradiated-fuel inventory, alloy-preparation runs were made with unirradiated feed stock. Because the in-cell storage capacity is now fully utilized, subassemblies being fabricated with unirradiated fuel are being stored in the vaults.

Process operations are summarized in Table X.

TABLE X. Production Summary

	7/1/67 through 7/31/67	Total This Year
1. Subassemblies received:		
Core Control, Safety (for processing)	0	40
Other (XO23 and XO24 to TAN)	2	117
2. Subassemblies dismantled (for processing); includes one for reassembly in new hardware	0	45
3. Subassemblies dismantled (for examination, etc.); includes 2 refabricated subassemblies with bad hardware	0	31
4. Subassemblies fabricated (includes those made with unirradiated fuel)	15	106
5. Subassemblies transferred to reactor	3	71
6. Elements decanned:		
From irradiated subassemblies	134	4296
Rejects	210	694
Other	0	0
Total Decanned	344	4990
<u>Melt Refining</u>		
	<u>Irradiated Fuel</u>	<u>Recycle Material</u> <u>New Fuel</u>
7. No. of runs	2	7 16
8. Average pour yield, %	95.0	94.6 96.4

TABLE X (Contd.)

	<u>Processing</u>	
	7/1/67 through 7/31/67	Total This Year
9. Injection-casting runs (total number)	24	114
10. Elements processed:		
Accepted	1692	9709
Rejected	239	1280
11. Elements welded:	1796	9847
Rewelded:	3	12
12. Elements leak tested:		
Accepted	1686	9140
Rejected	30	166
13. Elements bonded (including recycle)	1913	11,411
14. Elements bond tested:		
Accepted	1617	9272
Rejected	128	1448
15. Elements to surveillance	None	None
<u>Waste Shipments</u>		
16. Cans to burial ground	6	68
17. Oxide and glass scrap to ICPP	10	45

Subassembly XO11 was transferred from the reactor to the FCF to attempt to verify a cladding break in a fuel capsule (see Sect. II.A.4.e).

The manufacturer of the melt-refining crucibles has confirmed that some of the crucibles are low in density and that this may be the cause of the cracking reported previously (see Progress Report for May 1967, ANL-7342, p. 44). He has had problems in manufacturing of the present order of new crucibles, but was able to obtain 20 acceptable crucibles which will constitute a 6-week supply. Manufacturing of a new batch is underway and should be completed before these 20 crucibles have been used.

(ii) Cold-line Operation. Installation of the alloy-preparation furnace has been completed and this unit is under operational testing. Five runs were completed using depleted uranium. For one of these runs, chopped depleted uranium-fissium alloy pins were melted and cast into an ingot. Maintaining the integrity of the immersion and top reference thermocouples through a complete run has been the only major problem encountered. Use of coated graphite crucibles appears satisfactory, but its success cannot be measured until injection cast pins are available and analyses for carbon pickup are complete.

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

(a) Injection-casting Furnace and Pin Processing. Installation of the injection-casting furnace is complete except for wiring in the control console and induction power supply. The system has been successfully vacuum tested to 200 μ pressure, and pressure tested to 35 psig. Pin-processing equipment and instrumentation is ready to setup and test as soon as building modifications are completed.

(b) Glovebox and Element Assembly. The glovebox for sodium loading and element assembly is complete except for leak testing.

(c) Fuel-element-process Equipment. Testing of the leak detector is complete. The equipment is operational. Bonders and the bond tester are ready to be installed.

(d) Assembly Machine. Drawing of details for the cold-line assembly machine are in Central Shops. Most items are in the process of being fabricated; some, such as the drive motor and gear reducer, are already on site.

(e) Installation. The building addition on the ITF for the degreaser is complete and the degreaser itself in operation.

(iii) Maintenance and Repair. Routine maintenance and repair is required to keep the existing equipment in operating condition and hence to prevent production interruptions.

The pickup brushes were replaced on the Air Cell crane carriage by transferring the carriage through the Air Cell roof hatch and using direct maintenance. This solved the problem of a poor connection at the south end of the crane bridge.

The computer program for equipment maintenance has been worked out and is being initiated for some of the equipment as a test of this program.

(iv) Operator Training. On-the-job training continued for technicians who have been given new work assignments.

(v) Operating Manuals and Reports. In the past month, two new sections and six revised sections of the FCF Operating Manual have been completed and submitted for review and acceptance. Ten sections, revisions, and parts of sections have been issued as accepted.

The criticality rules and procedures for operation of the cold-line within the FCF building have been submitted to the Laboratory Director's Office for approval.

(vi) Product Analysis

(a) Chemical Analyses. The number of fuel production analyses, together with the average values and the range, are tabulated below. All values fell within the normal production range.

<u>Production Subassemblies</u>			
<u>Analyzed for</u>	<u>Number</u>	<u>Average Value</u>	<u>Range</u>
U (Total and Isotopic)	21	94.43 w/o	94.19-94.76
U ²³⁵		52.23 w/o	52.11-52.32
Mo	24	2.51 w/o	2.38-2.63
Ru	20	1.93 w/o	1.82-2.05
Pd	26	0.169 w/o	0.15-0.19
Rh	13	0.288 w/o	0.26-0.30
Zr	23	0.092 w/o	0.07-0.15
Al	32	223 ppm	43-517
Si	21	190 ppm	90-620
Fe	25	300 ppm	98-750
Cu	<u>1</u>	55 ppm	
Total 206			

Installation of the atomic absorption spectrophotometer adjacent to the analytical caves for analysis of both irradiated and nonirradiated samples continued during the month. A small hood was fabricated and installed above the burner, fuel and oxidant lines were completed, and the instrument was assembled in place. Final alignment of the optical system will be performed by the manufacturer's service engineer.

(b) Postirradiation Analysis of EBR-II Fuel. Mark-IA fuel elements from 20 subassemblies sent to the Fuel Cycle Facility between reactor power Runs 24 and 25 have been tested and increases in fuel pin volume calculated. The average increase of fuel volume for this group of subassemblies was lower than the volume increases of comparable Mark-IA elements from subassemblies that have been tested in the past. Possible reasons for the lower increases are being investigated.

Subassemblies that will be removed from the reactor at the end of Run 26 will be tested to determine if the lower volume increase is common to all fuel batches or if the fuel that was represented in the latest subassemblies is performing differently than the fuel examined previously.

Experimental driver-fuel subassemblies are being planned and fabricated to help determine the causes for variations in increase of fuel volume between Mark-I and Mark-IA designs and variations between different injection-casting batches.

Five Mark-IB-fueled subassemblies are being planned and should be available for Run 27.

Five special ingots have been prepared which contain varying amounts of carbon, silicon, aluminum, and iron. Postirradiation data obtained from elements cast from these ingots will be used to determine the specifications of these trace elements in future fuels for EBR-II. Special handling procedures will be employed in the casting of these ingots into fuel pins in order to aid in determining the effects of injection-casting parameters on radiation damage to the fuel.

The ICPP in-house corrosion tests are still in progress. The tests are being carried out to determine if, after in-reactor exposure of the fuel element to sodium, the corrosion rate of the stainless steel cladding is accelerated during storage in the water basin. ICPP personnel have reported that the five fuel elements selected for the corrosion studies were tested for cladding defects after one-month storage under water (in the ICPP basin) and none was found. Underwater storage of the five elements continues.

c. FCF Equipment Improvements

(i) Processing Equipment: The installation of the vertical assembly and dismantling machine (VAD) is complete in the Air Cell of the FCF. Testing of the electrical and pneumatic leads and of the mechanical working of the machine was completed. Production operations with the VAD were begun on July 28.

Purchase orders for procurement of fabricated items for the pneumatic-transfer-tube system used to transfer samples between the FCF and the analytical laboratory have been initiated. The project is about one month behind schedule, having been delayed by projects of higher priority. Completion is now forecast for the end of FY 1967.

Checking of the drawings for the neutron-radiography equipment is proceeding. It is anticipated that the first drawings will go to the shop during the coming month. Based on present evaluation, this project should be completed during FY 1968.

Design efforts have been initiated on an improved straightness and tensile-testing machine. This machine would replace the two now employed in the Air Cell for final inspection of refabricated subassemblies. In addition to requiring less cell floor space, the new combined machine would save operation time now required to transfer a subassembly from one machine to the other, and make the operation less complicated.

(ii) Auxiliary Equipment. As a means of decreasing the interdependency of the schedules of the FCF, 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 operations. An interim-storage facility providing for a two-month holdup capacity would insure compensation for credible interruptions.

It was previously reported (see Progress Report for May 1967, ANL-7342, p. 53) that owing to insufficient storage capacity and shielding problems, it would not be feasible to utilize the existing storage pits in the passageway between the reactor building and the FCF. An alternative approach for providing the additional interim-storage capacity involves constructing a new wing in which up to 69 complete subassemblies would be stored in shielded and air-cooled, floor storage pits. The new wing would be located near the FCF passageway. Cost estimates for the construction of this storage facility are being developed.

A shielded cask employing bottom loading and discharging of a subassembly will be used (1) to transport subassemblies to and from the new facility, (2) to charge and remove a subassembly from a storage pit, and (3) as an alternate cask for the transfer of subassemblies from the FCF to the ICPP for storage and subsequent reprocessing. The cask will handle one subassembly. Design concepts of this cask are being coordinated with those of the new storage facility.

Tests are continuing on the filter system for use with the recirculating gas in the Argon Cell (see ANL-7342, p. 52).

A new drive has been obtained for the shielding plug which is being added as part of the transfer cell between the Air and Argon Cells.

(iii) Repair and Decontamination. Work continued on the engineering of this facility. Detailed design and drawings of the Process Equipment Decontamination and Repair Facility are progressing. The engineering is 50% complete and the entire project 25% complete.

Detailed drawings of the Manipulator Decontamination and Repair Facility have been started and the job is estimated to be 15% complete.

14. Hot Fuel Examination Facility (HFEF)

The Prospectus to be used by the Project and architect-engineering firm as a guide for the preparation of a Cost and Feasibility Study and Title I Report was completed and submitted to AEC for approval. This Prospectus provides most of the conceptual design requirements for the entire facility as well as a suggested schedule to permit completion of the construction effort by late 1970.

The first meeting of the architect-engineering selection board was held on July 14, 1967, on schedule. The Prospectus will be submitted to qualified Architect-Engineer (A-E) firms who display an interest in the work, as soon as approval of the Prospectus has been received from AEC. The Project schedule calls for an A-E to be selected and under contract by October 16, 1967.

Steps were taken to obtain a consultant-engineering firm to assist the Project in the establishment of a reasonable cost estimate for submission to AEC in November 1967. Review of the qualifications and final selection is scheduled for early August, with active participation expected by August 14, 1967. This cost estimate is to be independent of the Cost and Feasibility Study performed by ANL and the A-E (due February 1, 1968).

Drawings of the Mark-E4A electric master-slave manipulator and rudimentary specifications were sent to four manufacturers to get outside cost estimates. These will only be preliminary estimates since the Mark E4A must have several modifications before it can be installed in the HFEF. The estimates are due at ANL by September 1, 1967.

Two configurations based on the transfer-tunnel approach were further evaluated. Considerable attention is being given to a configuration in which the low-level area is butted directly to the end of the high-level area. A shadow shield between these two areas permits thinner shielding walls in the low-level area. A common gas atmosphere would be used.

B. Physics Development

1. ZPR-3

a. Assembly 50. Assembly 50, the third assembly in the series of simple plutonium-fueled assemblies being studied in ZPR-3, was loaded to critical and contained an uncorrected mass of 218.5 kg of $\text{Pu}^{239} + \text{Pu}^{241}$ in the core. Assembly 48, the first assembly in the series, contained a core loading similar to that expected in a fast power reactor plus extra graphite to soften the neutron spectrum. Assembly 49 was the same as Assembly 48 except that all of the sodium was removed so that the effect of sodium on the various reactor parameters could be studied. Assembly 50 is like Assembly 48 except that graphite now has replaced the sodium in the core loading. Assembly 50 has the smallest core and the softest neutron spectrum of the three assemblies studied.

Experiments completed this month on Assembly 50 include the measurement of the worth of core drawers at the edge and the worth of control and safety drawers as compared to core drawers so that the corrections for these various items can be made to the critical mass. Analysis of data from Assembly 50 as well as data from previous assemblies is continuing.

b. Assembly 48B. Experiments with Assembly 48B were completed in March 1967. Additional experiments have been analyzed and the results are presented here.

Assembly 48B was a variation of Assembly 48 in which a central zone in the core contained plutonium fuel with a higher Pu^{240} content than the fuel used in Assembly 48 (see Progress Report for March 1967, ANL-7317, pp. 33-42; and for June 1967, ANL-7349, Fig. 8, p. 64).

Central perturbation measurements were made in the high Pu^{240} zone with a group of fissile and nonfissile materials. The results of these measurements are summarized in Tables XI, XII, and XIII.

Small-sample reactivity traverses at the reactor midplane were made at selected points in the high Pu^{240} zone and in the low Pu^{240} zone surrounding it. The results of these measurements are given in Table XIV.

c. Assembly 49. Experiments with Assembly 49 were completed in June 1967. Several of the experiments run on Assembly 49 have now been analyzed and the results are presented here.

TABLE XI. Central Perturbation Measurements with Fissionable Materials in Assembly 48B of ZPR-3

Sample Type	Thickness or Diameter (in.)	Measured Worth (Ih/kg)
Pu (4.5% Pu ²⁴⁰)		
Annular foil	0.010	423 ± 7
Annular foil	0.020	428 ± 7
Pu (22% Pu ²⁴⁰)		
Annular foil	0.005	350 ± 6
Annular foil	0.010	356 ± 6
Annular foil	0.020	365 ± 6
Enriched Uranium ^a		
Annular foil	0.010	310 ± 3
Depleted Uranium ^b		
Annular foil	0.022	-22.2 ± 0.7

^aEnrichment range 93.10 to 93.30%.^b0.40% U²³⁵.

TABLE XII. Central Perturbation Measurements with Boron and Tantalum Samples in Assembly 48B of ZPR-3

Sample Type	Thickness or Diameter (in.)	Measured Worth (Ih/kg)
Boron (Enriched in B ¹⁰) ^a		
Annular	0.010	-7341 ± 50
Annular	0.020	-7145 ± 50
Annular	0.040	-6651 ± 45
Tantalum		
Annular	0.010	-153 ± 10
Annular	0.015	-147 ± 10

^a92.8 w/o B, 7.2 w/o impurities; boron = 92.1 w/o B¹⁰.

TABLE XIII. Central Perturbation Measurements with 0.42-in.-dia Cylindrical Samples of Selected Nonfissile Materials in Assembly 48B of ZPR-3

Sample Type	Measured Worth (Ih/kg)	Sample Type	Measured Worth (Ih/kg)
Fe	-13.2 ± 0.6	C	-7.4 ± 1.8
SS	-13.1 ± 0.2	Cr	-9.2 ± 0.7
Ni	-17.4 ± 0.4	Na	-6.3 ± 3.0

TABLE XIV. Radial Traverses of Reactivity Samples
in Assembly 48B of ZPR-3

Radius (cm)	Reactivity Coefficient (lh/kg)					Stainless Steel
	Plutonium (5% 240)	Enriched Uranium	Depleted Uranium	Enriched Boron	Tantalum	
	± 2	± 3	± 0.6	± 30	± 2	± 0.6
-11.11	402	289	-20.1	-6723	-134	-11.7
0.00	437	310	-23.7	-7371	-137	-16.1
+11.11	392	277	-20.4	-6571	-133	-14.6
+22.23	287	207	-9.9	-4625	-88	-4.6
+33.34	161	117	-1.6	-2062	-37	+4.4

The core and blanket outline of Assembly 49 are shown in Fig. 4. The fuel mass loaded on the lattice was 281.0 kg Pu^{239} ($282.9 \text{ kg } \text{Pu}^{239} + \text{Pu}^{241}$). Preliminary calculations, correcting for control- and safety-rod spiking, roughness of core outline, and interface gap, yielded a corrected heterogeneous critical mass of 293.5 kg Pu^{239} . The corresponding core radius was 43.30 cm, with radial and axial blanket thicknesses of 32.75 and 30.05 cm, respectively. The core length was 76.35 cm. The composition of the core for this assembly is given in Table XV. The reflector compositions were the same as for Assembly 48.

Fig. 4. Arrangement of Assembly 49 of ZPR-3

TABLE XV. Core Composition of Assembly 49 in ZPR-3

Isotope	Composition (10^{22} atoms/cm ³)	Isotope	Composition (10^{22} atoms/cm ³)
Pu ²³⁹	0.1644	Fe	1.0083
Pu ²⁴⁰	0.0106	Cr	0.2508
Pu ²⁴¹	0.0011	Ni	0.1121
Pu ²⁴²	0.00004	Mn	0.0105
Mo	0.0206	Si	0.0123
Al	0.0109	U ²³⁵	0.0016
C	2.0766	U ²³⁸	0.7406

Measurements of sodium worth were carried out by incremental substitution of sodium in a 37-central-drawer cross section shown in Fig. 5. A total of 20.6 kg of sodium was added. A preliminary estimate of this sodium addition worth is $133 \times 10^{-5} \Delta k/k$.

Fig. 5. Sodium Zone in Assembly 49 of ZPR-3

The data from central-drawer foil scans and perturbation and Doppler coefficient measurements are being evaluated.

2. ZPR-6

Measurements of physics parameters are being made with Assembly 5 of ZPR-6. The importance of fission neutrons as a function of radial position was measured with the Cf^{252} source technique.⁴ The results are shown in Fig. 6 together with the calculated distribution which was obtained with an MC²-averaged set of cross sections for Assembly 5.

Fig. 6. Importance of Fission Neutrons as a Function of Radius in Assembly 5 of ZPR-6

The necessary parameters for evaluating the normalization integral of the perturbation expression experimentally were also measured, and these data are now being processed.

The self-shielding and self-multiplication in 1/16-in.-thick U^{235} plates which are used in the loading of Assembly 5 were studied by replacing the 2 x 2 x 1/16-in. plate of U^{235} with an equivalent mass composed of a stack of eighteen 2 x 2 x 0.004-in. U^{235} foils. These foils were irradiated; the activities of the fission fragments in 3/8-in.-dia foils which were punched out of the center of each of the foils were determined. Determination of the activities at the center of each foil eliminated any edge effect that may be present. The same procedure was used for 1/8-in.-thick U^{238} plates. In the case of the U^{238} foils, the activity of Np^{239} resulting from neutron capture in U^{238} as well as activity of the fission fragments were counted. These data are being analyzed.

⁴Karam, R. A., et al., Measured Physics Parameters in a Zoned Fast Uranium Carbide Core, Assembly 4Z, ANL-7281 (1966).

3. ZPR-9

A series of Doppler effect measurements were made with Assembly 18 of ZPR-9, with different material environments around the Doppler sample:

- (a) 1-in. natural UO_2 Doppler sample surrounded by 1 in. of depleted uranium;
- (b) 1-in. natural UO_2 Doppler sample surrounded by $1/2$ in. of depleted uranium;
- (c) 1-in. natural UO_2 Doppler sample surrounded by 1 in. of stainless steel;
- (d) 1-in. natural UO_2 Doppler sample surrounded by $1/2$ in. of stainless steel.

A series of small-sample sodium-worth measurements using the Doppler oscillator were also made. In addition, measurements designed to check for possible experimental uncertainties are also scheduled to be made.

Fission-ratio and fission-traverse measurements are currently being made.

In Table XVI are listed the results of the Doppler-effect measurements in Assemblies 13, 14, 16, and 17, which were designed to have progressively softer spectra. The material compositions and sizes of these assemblies are given in Table XXXVIII, p. 68, of the Progress Report for June 1967, ANL-7349.

4. ZPPR

a. Construction. Good weather has aided the continued construction of the ZPPR Facility. Following the placement of the cable sleeves in the cell ring beam forms, concrete for the ring beam, which has a cross section $4\frac{1}{2}$ ft high by 7 ft wide, was poured on July 1. On July 5, radial and circumferential cracks on the top surface of the ring beam were discernible in the pattern of the top reinforcing steel. Six exploratory holes were made down to the rebar, and it appears that there are cracks between the concrete and the top layer of rebar. ANL has recommended to the Commission that the necessary repair and replacement of the top section of the ring beam by acceptable means be made.

The catenary array consisting of 80 cables has been pulled. The sag of cables has been adjusted in preparation for installation of the cable clamps. Prior to stringing the cables, the cell-crane components were placed in the cell.

TABLE XVI. Experimental Doppler Results with ZPR-9

Assembly ^(a)	Element ^(b)	Type ^(c)	Average Temperature, Average Reactivity Worth, Standard Deviation of the Average Reactivity Worth									°K	lh/kg U ^x	±lh/kg U ^x
			°K	lh/kg U ^x	±lh/kg U ^x	°K	lh/kg U ^x	±lh/kg U ^x	°K	lh/kg U ^x	±lh/kg U ^x	°K	lh/kg U ^x	±lh/kg U ^x
13	N1	FE	293	0.0	0.010	393	-0.276	0.009	498	-0.590	0.010	715	-1.160	0.009
			877	-1.518	0.005	1076	-1.916	0.010						
13	HE-8	FE	294	0.0	0.004	420	+0.131	0.005	517	+0.247	0.003	668	+0.407	0.002
			881	+0.601	0.002	1074	+0.765	0.005						
13	HE-9	NNE	293	0.0	0.004	389	+0.084	0.003	443	+0.141	0.003	488	+0.181	0.003
			590	+0.283	0.003	692	+0.386	0.004	893	+0.532	0.004	1070	+0.634	0.007
14	N1	FE	288	0.0	0.005	390	-0.365	0.005	499	-0.715	0.004	681	-1.217	0.004
			898	-1.170	0.003	1073	-2.082	0.002						
14	HE-8	FE	291	0.0	0.004	428	+0.159	0.002	515	+0.238	0.004	668	+0.375	0.003
			899	+0.552	0.002	1078	+0.672	0.003	378	+0.106	0.006			
14	HE-9	NNE	292	0.0	0.005	397	+0.116	0.004	456	+0.160	0.004	507	+0.211	0.004
			589	+0.286	0.003	703	+0.382	0.007	911	+0.513	0.004	1057	+0.601	0.004
16	N1	FE	292	0.0	0.002	380	-0.365	0.006	480	-0.730	0.009	688	-1.368	0.004
16	N1	FE	291	0.0	0.003	458	-0.640	0.003	593	-1.072	0.002	882	-1.855	0.005
			1079	-2.314	0.004									
16	HE-8	FE	291	0.0	0.004	409	+0.082	0.002	501	+0.153	0.004	588	+0.212	0.002
			710	+0.284	0.003	903	+0.386	0.003	1076	+0.474	0.002			
17Z	HE-8	FE	292	0.0	0.005	429	-0.200	0.004	592	-0.361	0.004	813	-0.525	0.005
			1089	-0.702	0.003									
17F	HE-8	FE	293	0.0	0.003	405	-0.176	0.003	495	-0.283	0.003	596	-0.384	0.004
			709	-0.483	0.005	898	-0.612	0.003	1091	-0.732	0.005			
17F	HE-10	NNE	293	0.0	0.007	402	-0.200	0.007	472	-0.310	0.007	521	-0.379	0.007
			609	-0.467	0.007	747	-0.593	0.008	900	-0.707	0.009	1048	-0.796	0.010
17F	N1	FE	294	0.0	0.003	385	-0.634	0.002	497	-1.302	0.004	616	-1.937	0.007
			755	-2.605	0.005	902	-3.251	0.008	1067	-3.936	0.008			

(a) Assembly 13: 1% $\Delta k/k$ = 437.85 lh.
 Assembly 14: 1% $\Delta k/k$ = 435.69 lh.
 Assembly 16: 1% $\Delta k/k$ = 428.07 lh.
 Assembly 17Z: 1% $\Delta k/k$ = 415.19 lh.
 Assembly 17F: 1% $\Delta k/k$ = 391.78 lh.

(b) N1 1.26629 kg UO₂; 1.11623 kg U; 1.10818 kg U²³⁸; 0.00799 kg U²³⁵;
 HE-8 1.03758 kg UO₂; 0.91328 kg U; 0.01012 kg U²³⁸; 0.89187 kg U²³⁵;
 HE-9 0.91219 kg UO₂; 0.80419 kg U; 0.00891 kg U²³⁸; 0.78534 kg U²³⁵;
 HE-10 0.91560 kg UO₂; 0.80710 kg U; 0.00894 kg U²³⁸; 0.78818 kg U²³⁵;
 HE elements w/o U: U²³⁴, 1.026%; U²³⁵, 97.656%; U²³⁶, 0.209%; U²³⁸, 1.108%
 N1 elements w/o U: U²³⁴, 0.0057%; U²³⁵, 0.7158%; U²³⁸, 99.2785%.

U^x--Inhours per kg of predominate uranium isotope.

(c) FE--Freely expanding UO₂ element 1 in. x 12 in. pellet cylinder.

NNE--Non-Non-expanding UO₂ element 1 in. x 12 in. powder cylinder (retained both axially and radially).

The installation of instrument and control wiring is continuing. All of the instrumentation wires except the coaxial cables for the nuclear instruments have been pulled from the seal pit to the cable-routing room. The seal frames for the cable through the access corridor are being re-worked to provide a flat sealing surface.

The roof of the support wing has been completed, and work on finishing the inside walls and doors is continuing. The subcontractor has commenced installing insulation of piping and air-handling ducts. All of the air-handling equipment has been installed in the fan loft of the support wing.

The subcontractor has processed the sand for the cell roof and the emergency filter. The gradation of the sand appears to be satisfactory; it is estimated that the quantity of sand, about 230 cubic yards, is not sufficient and about 40 more will be required. The contractor is in the process of evaluating what will be done to obtain the additional sand.

The Final Safety Analysis Report has been reviewed by the Laboratory and was transmitted to the AEC for review and approval on July 28.

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

Personnel Shield Tractor Drives: The two units are scheduled for shipment on August 1.

Control Rod Drive Mounting Plates: The two units are scheduled for machining, but no work was done in July due to vacations. Work is scheduled to start again on July 31. ANL representatives visited the plant on July 10 and are scheduled for another visit on August 3.

Fuel Control Rod Drawers: The new drawers were received and inspected. The drawers were again out of tolerances and rejected. The vendor has received the returned drawers and is now attempting to repair them.

Boron Carbide Plate Fabrication: Approximately 50 plates have been completed. An ANL representative is scheduled to visit the vendor's plant on August 2.

Alpha Air Monitors: A new development in monitor counting technique is being evaluated and the bid package for the monitors will be sent out early in August.

b. Fuel Procurement. Procurement of ZPPR fuel and experimental materials is being carried out by subcontract for (1) preparation of plutonium and fertile seed materials, (2) manufacture of jacket components, (3) fuel-element fabrication, (4) fertile-element fabrication, and (5) manufacture of elements containing sodium and other nonnuclear materials.

The plutonium available for ZPPR-element production ranged widely in isotopic composition; for example Pu^{240} ranged from approximately 8 to 15%. This material was blended to 11.5% Pu^{240} content and cast by Isochem Inc. as 1.1-kg ingots.

Isochem Inc. completed 1337 kg of ingots and submitted analyses for materials cast during March, April, and May. The Isochem isotopic analyses taken from melt samples for materials ingoted during these three months is as follows:

Isotope (w/o)	Relative % 2 σ Confidence	March, n = 93		April, n = 116		May, n = 91	
		Range	Average	Range	Average	Range	Average
Pu ²³⁸	±102.7	0.067-0.011	0.032	0.094-0.022	0.039	0.028-0.076	0.041
Pu ²³⁹	±0.36	87.158-86.034	86.796	87.147-86.227	86.710	87.215-85.978	86.527
Pu ²⁴⁰	±0.25	11.972-11.264	11.525	11.922-11.257	11.536	11.990-11.007	11.527
Pu ²⁴¹	±3.75	1.752-1.412	1.493	1.863-1.433	1.570	1.971-1.500	1.703
Pu ²⁴²	±7.06	0.177-0.103	0.130	0.194-0.120	0.149	0.254-0.135	0.184
Am ²⁴¹	±8.8	1181-105	475	1506-<50	338	728-<50	0.314

The results of isotopic analyses obtained from the Dow Chemical Company of ingots were slightly different than those obtained on the melt samples by Isochem. The difference appears to be insignificant for purposes of meeting the specification requirements, but is of concern from a nuclear-materials-management standpoint because of the large quantity of plutonium involved. A program has been set up to establish whether the differences of analysis result from:

1. differences or inhomogeneities between samples from two laboratories;
2. differences in analytical techniques;
3. differences in "standards" used for control of the analytical techniques.

Preliminary meetings were held between ANL, Dow, and Isochem to study the problem. A one-day symposium was organized between mass spectrographers from the above organizations and from the NUMEC-designated laboratories to discuss the mass-spectrometric analysis. A meeting will be held at ANL on August 11, 1967.

A contract was signed with NUMEC on July 28, 1967 for fabrication of 1000 kg of plutonium into 5379 ZPPR fuel elements, with additional options for 3091 (575 kg of plutonium) and 3366 (625 kg of plutonium). The contract calls for a partial delivery of 2151 fuel elements by February 29, 1968, and the complete delivery of the contract required 5379 fuel elements by August 31, 1968. The option amounts may be produced sequentially after delivery of the first 1000 kg of plutonium.

A preproduction run of fuel elements made at Rocky Flats showed that additional work is required on the production molds and casting techniques proposed by them. The other operations: machining, assembly, welding, leak detection, counting, and inspection, functioned satisfactorily. Steps are being taken to change furnaces and molds to duplicate the casting parameters that were previously found to be satisfactory.

Jacket materials, i.e., stainless steel tubing and formed wire, were received for ZPPR jacket fabrication. Approximately 1000 jacket sleeves were stretch formed by the wedge-mandrel method. The dimensions were as specified. The fuel-element assembly and welding operations required that the end plugs be a light push fit into the jackets and that gaps in the corners be minimized. The close-fitted end plug reduces the risk of contamination during assembly and welding, and yields better uniformity of the welds. Some difficulty was experienced with the initial tooling for production manufacturing of the end plugs, and it was necessary to supply the initial requirements by tool-room methods. New tooling was designed to shave simultaneously the ends and four radii on the end plugs. The shaving dies have been tested, found to work satisfactorily, and are now in use.

5. Burnup Analysis and Fission Yields for Fast Reactors

a. Development of Analytical Procedures for Fission Product Burnup Monitors. In the determination of burnup of fast reactor fuels by fission product analysis, radiochemistry, mass spectrometry (isotopic dilution), and spectrophotometry have been the techniques most widely used. However, each technique has very definite limitations. Cesium-137 is the only active fission product which has a sufficiently high and dependable yield and a sufficiently long half-life to warrant consideration as a burnup monitor using a radiochemical technique. However, it diffuses out of both metal and oxide fuels during irradiation. Mass spectrometry is applicable to the analysis of a number of fission products but the equipment is very costly and difficult to operate, the separation techniques which must be used to prepare the samples for analysis are generally intricate and therefore time-consuming, and the technique cannot be applied to elements which are monoisotopic, e.g., yttrium, technetium, rhodium, and praseodymium. Spectrophotometry can be applied to the determination of fission products but it too is limited in its application. Of the fission products which can be considered useful burnup monitors, spectrophotometry is applicable to only molybdenum and technetium. Both these elements diffuse out of oxide fuels. A program is therefore being carried out to investigate the feasibility of using other analytical techniques for the determination of burnup in reactor fuels.

(i) Photometric Titration of Total Rare Earths. The status of the investigation of a method that uses an EDTA titration of the total rare earth group to monitor burnup was reported in the Progress Report for March 1966, ANL-7317, p. 47. In that report, low results were attributed to the polymerization of cerium during the separation procedure.

Further tests of the entire procedure using cerium that had been tagged with cerium-144 confirmed this conclusion. Although the tracer indicated that the recovery of cerium in the separation procedure was complete, the amount of EDTA consumed in the titration was 30% low. This

incomplete reaction of cerium with EDTA has now been overcome by changing from a direct titration with EDTA to a back titration. In the back titration, a known excess of EDTA is added to the cerium solution, the solution is allowed to stand for 30 min to insure that the depolymerization is complete, and the excess EDTA is titrated with standard lanthanum solution. Allowing the cerium solution to stand in the presence of excess EDTA for an extended period apparently provides the conditions necessary for destroying the cerium (IV) polymers. The back titration was tested on mixed rare earth solutions which had been treated under conditions which had given low results in the forward titration. When these results were compared with back titration results obtained with untreated rare earth solutions, they agreed within experimental error ($\pm 0.5\%$).

The effectiveness of the proposed procedure in separating the rare earths from other fission products was tested using a sample of irradiated uranium. It was found that the anticipated quantitative separation from zirconium was not achieved; the amount of zirconium accompanying the rare earths would cause an error of about 5% in the rare earth analysis.

The separation of rare earths from zirconium has been effected by replacing the initial step in the separation procedure, a coprecipitation of the rare earths on calcium fluoride, with an anion exchange separation in 12M hydrochloric acid. Uranium, plutonium, zirconium, and many other fission products are absorbed onto the column while the rare earths and the group I and II fission products pass through. (The separation of rare earths from yttrium must still be effected by means of an anion exchange separation in a nitric acid-methanol mixture.⁵)

The entire procedure was tested with three aliquots of a mixture of rare earths (500 μ g). Results of the titrations showed recoveries of 100.1, 99.3, and 99.6%. Further tests of the effectiveness of the separation procedure will be made using irradiated fuel material to verify that there are no other major interferences.

(ii) X-ray Spectrometric Determination of Rare Earth Fission Products. The feasibility of measuring burnup by using X-ray spectrometry for the determination of fission products in fast reactor fuels is also being considered. X-ray spectrometric analysis has a very important advantage when compared to radiochemistry, mass spectrometry, and spectrophotometry in that the separation procedure does not have to provide a separated fission product of high chemical purity. The procedure must only provide (1) a separation from active fission products that is sufficient to protect the analyst from overexposure and (2) a separation from gross amounts of uranium, plutonium, and the other fission products.

⁵Chemical Engineering Division Semiannual Report January-June 1966, ANL-7225, p. 229.

Since the present emphasis in the fast reactor fuel program is on oxide fuels, and since it appears that the rare earths will not diffuse out of the fuel during irradiation, the initial application of X-ray spectrometric analysis to fission product analysis is directed toward the rare earths. The following general procedure is being considered:

1. Quantitative isolation of the rare earth fission products from uranium, plutonium, and high specific activity fission products, e.g., zirconium and ruthenium.
2. Addition of an internal standard, i.e., a known amount of an element whose chemical properties are similar to those of the fission product being determined. For the rare earth fission products, lutetium, which is not a fission product, could be used as an internal standard.
3. Transfer of all or a portion of the solution onto a metal plate and evaporation to dryness.
4. Assay of the plated sample in the X-ray spectrometer for the internal standard and the fission product.
5. Determination of the atom ratio of the fission product (F.P.) to the internal standard (I.S.) from a calibration curve in which $\text{concentration}_{\text{I.S.}} / \text{concentration}_{\text{F.P.}}$ has been plotted against the X-ray intensity ratio, $I_{\text{I.S.}} / I_{\text{F.P.}}$.

The X-ray spectrometric work can be divided into three areas of inquiry:

(a) The Effect of Fission Product Radioactivity on the Operation of the Spectrometer. Of the 10 rare earth fission products which have yields greater than 1%, the isotope whose activity is most likely to affect the X-ray detectors on the spectrometer is 285-d cerium-144. It is the only long-lived rare earth fission product which is a significant gamma emitter; its daughter, 17-min praseodymium-144, is the principal beta emitter in the rare earth group. To test the effect of cerium-144 on the detectors, 2.5 μg of cerium-144 (the amount which would be present in 50 μg of rare earth fission products) was inserted into the spectrograph. A small but inconsequential increase in background was observed.

(b) Choice of Spectra. Both the K-spectra and L-spectra are available for use in rare earth X-ray spectroscopy. The most attractive feature of the K-spectra is simplicity--only two principal emission lines per element. On the other hand, the K-spectra exhibit several disadvantages. High excitation potentials are necessary to excite the K-spectra and adequate

spectral dispersion is difficult to attain. Line-to-background ratios are poor; this in turn affects the ultimate sensitivity attainable.

The L-spectra, on the other hand, exhibit more attractive line-to-background ratios and greater sensitivity can be attained with their use. The L-spectra have one distinct disadvantage; they have a large number of emission lines, and in a multielement situation line conflicts are quite commonplace. More experimental work is necessary before a decision can be made as to which spectra give the more reliable results.

(c) Analytical Accuracy. The technique of presenting the separated rare earths to the X-ray spectrometer as a solution provides the desired accuracy, $\pm 1\%$, but the amount of fission product which would have to be separated from the fuel, 5 to 10 mg, would be excessive. (One gram of 1% burnup fuel contains about $300\ \mu\text{g}$ of a fission product having a 6% yield.) Presenting the separated rare earths to the spectrometer as a solid on a metal plate enhances the sensitivity markedly but the geometric reproducibility, $\pm 5\%$, of the sample mount is not good enough to obtain the desired accuracy. The precision of the measurements can probably be improved by incorporating either an internal standard technique or a standard addition technique. Both techniques will be investigated.

C. Component Development

1. Reactor Mechanism and Instrumentation

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 has 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.

(i) Electrical Insulators. The electrical resistivity of such various forms of high-temperature insulators as thoria, alumina, beryllia, and hafnia are being measured up to temperatures of $\sim 2800^\circ\text{C}$ to determine their potential usefulness; mechanisms of failure are being investigated.

The two coaxial resistivity probes TR-1 and TR-2, described previously (see Progress Report for June 1967, ANL-7349, pp. 70-71), yielded the results shown in Fig. 7. The reasons for the slightly different resistivities above 2000°C are not immediately apparent, but will be systematically investigated by testing additional thoria samples.

The small anomalies shown may be caused (a) by the differing and uncontrollable line and point contacts between the insulator and

Fig. 7. Resistivity of Vitified Thoria (99.9% Pure)

date, as was discovered when alumina was similarly tested. This effect will also be investigated further to determine its significance to the successful use of thoria as thermocouple insulators.

(ii) Thermoelement Properties. Long-term thermoelectric stability and failure mechanisms of various thermoelement combinations and forms are being investigated experimentally.

The first thermoelectric stability test (BTC-1) of W-3% Re/W-25% Re thermocouple wire had to be terminated after 24 hr at 2510°C because the apparent thermoelectric output decreased by more than 10%. The reduction in emf output was found to be caused by contact of the thermocouple wires with the tantalum support tube at places along the hot zone other than the hot junction. In an attempt to correct this mechanical malfunction, a slightly different bare-wire thermocouple design is being fabricated and will be tested as soon as the furnace schedule permits.

electrodes, as well as the small gas gaps that necessarily exist in the construction of a vitrified-type resistivity probe or thermocouple, or (b) by slightly different stoichiometric compositions. Both possibilities are being investigated.

The resistivity values of 99.9%-pure vitrified thoria measured at a given temperature are somewhat larger than those obtained from 99.8%-pure crushable thoria but somewhat smaller than the apparent resistivity of either helium or vacuum. The resistivities measured for thoria, therefore, followed similar patterns established previously from vitrified and crushable alumina.

The "freezing-in" of high-temperature defects after quick cools to lower temperatures was not, however, evident in the case of the thoria samples tested to

(iii) Material Compatibility. The compatibility of typical thermocouple candidate sheath materials with fast-reactor fuel materials is being determined.

A metallographic comparison of the various compatibility samples tested thus far confirms that there has not been a reaction between the tantalum and the UO_2 under any of the test conditions previously described. Additional fueled samples are being fabricated for test.

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 Magnetic Materials. The effects of temperature on the magnetic and mechanical properties of various magnet configurations are being measured and evaluated; methods for "stabilizing" magnets for use in a flowmeter at 1200°F are being established.

Thermal tests of the Alnico alloys are continuing. Fabrication of the circuits for the B-H tester also is continuing.

Calculated values of the "too-close" errors, when the magnet length or the magnetometer probe length is nonnegligible compared to the distance d separating them, are significantly larger than those determined experimentally by making moment measurements with various probe-magnet spacings on the same magnet sample. A further study is being made to determine the cause of the discrepancies between the theoretical and experimental results.

The "too-close" errors, of course, cancel when determining the changes in any given magnet sample due to either temperature or nuclear radiation and are, therefore, relatively unimportant. It is, however, desirable to keep the moment measurements as accurate as possible so that discrepancies in absolute values of samples of different lengths can be explained and so that the results from these experiments can be correlated with other possible future work.

(ii) Magnetic-flowmeter Prototype. Prototype magnetic flowmeters are being designed, fabricated, and tested at sodium temperature up to 1200°F to determine calibration sensitivity, impedance, and field-noise signal levels; temperature compensation schemes will be developed if needed. As a final proof test, it is planned to install a prototype magnetic flowmeter in EBR-II.

The flowmeter sensitivity change has been reconfirmed by another calibration at 700°F. The sensitivity change is now postulated to have been caused by extremely heavy welding currents being conducted through the flowmeter assembly during the installation of new heaters on the loop. These currents could have caused a further permanent demagnetization of the flowmeter magnets. A simulation of this condition of heavy welding current will be performed with an out-of-loop flowmeter assembly to either verify or disprove this postulate.

After the successful operation of the recirculating loop and flowmeter at 700°F, the loop operating temperature was being raised to 1000°F. Upon reaching approximately 850°F, an expansion bellows near the electromagnetic pump failed, causing a sodium leak and consequently resulting in the loop being shut down. The bellows is being replaced and the loop will be restarted. The flowmeter time at various temperatures is summarized in Table XVII, a usable signal was extracted for a total of about 880 hr at temperatures above 400°F.

TABLE XVII. Temperature History of Flowmeter at the End of July 1967.

Temperature (°F)	Time (hr)
200-300	10.7
300-400	45.5
400-500	246.5
500-600	59.7
600-700	345.8
700-800	163.7
800-900	18.0

(ii) Magnet Irradiation Tests.

To determine performance after irradiation in fluxes simulating the EBR-II environment, rod magnets of Alnico-5, -6 and -8 are being prepared for installation in EBR-II irradiation subassemblies.

The experiment description and hazard analysis has been drafted. The recent core-configuration modification in the EBR-II has changed the gamma- and neutron-flux distribution; new flux-distribution data will be factored into the irradiation plans.

The leaks in the magnet-sample capsule (see Progress Report for June 1967, ANL-7349, p. 73) were not actual leaks; the incorrect indications were caused by faulty leak-testing techniques. All samples fabricated were helium leaktight.

The detailed design of the irradiation capsule is awaiting the results of final heat-transfer calculations. The mockup of the hot cell and manipulators is almost complete.

c. Fission-gas Pressure Transducer. To obtain an indication of the fission-gas pressure buildup in individual fuel pins while they are

operating in the reactor, commercially available pressure transducers are being reviewed and evaluated for FFTF conditions, and will be tested out-of-pile and in-pile. Three basic types of pressure transducers hold greatest promise for being useful in 0.25-in. diameters and in 1200°F sodium: a NaK-filled capillary system, a strain-gauge-type sensor, and a null-point system.

(i) NaK-filled Capillary System. Various versions of a bellows-operated NaK-filled pressure transmitter, consisting of a long capillary tube connected to a bellows and a strain gauge, are being modified and tested to determine their suitability.

The prototype pressure transducer system has successfully completed the second month of the 3-month life test (see Progress Report for June 1967, ANL-7349, p. 74).

The 1/8-in.-OD subminiature NaK-filled system was completed to the point of NaK filling when the bellows failed due to overstrain. Repair is being attempted; if the repair is not successful, a new one will be made. The first assembled system was found to be plugged. A second assembly (including the bellows and 10 ft of 0.012-in.-ID capillary tube) was cleaned with trichloroethylene, acetone, and ethyl alcohol, and dried with argon. Then the capillary tube was passed into a 9.75-ft-long, 1/16-in.-ID protective tube which was welded to the bellows end fitting and silver-soldered to the 0.012-in.-ID capillary tube via a fitting. A 10-ft-long, 1/16-in.-ID capillary tube was then silver-soldered to the fitting to continue the capillary tube to a total length of 20 ft, the end of which is connected to a strain-gauge pressure transducer and fill tube. At this point and just prior to filling, the bellows was flexed unintentionally and broke. A more substantial protective jig will be used in the next system to attempt to eliminate this problem.

d. Signal-lead Connector. Tests on the prototype connector are continuing. The female portion of the connector has been replaced with a unit having 12 drain holes around the gasket. The male portion of the connector has now been used in four testing periods, exposed to heating and cooling cycles and to sodium immersion totalling several months.

Better pressure measurements were obtained (a) after a thermocouple was added at the upper end of the main bellows and by normalizing observed pressures according to the temperature variations measured at this thermocouple, and (b) by a reduction in the internal volume of the system. The temperature variations at the bellows were also reduced by adding shrouds which reduce air current affecting the bellows.

e. Failed-fuel Locator. A detectable helium leakrate for the test system of $22.89 \times 10^{-10} \text{ cm}^3/\text{sec}$ was reported previously (see Progress Report for June 1967, ANL-7349, p. 75). The volume of helium added to the experimental sodium system at 1200°F was 25 cm^3 for a tank of $208,538 \text{ cm}^3$, or a volume ratio of 120×10^{-6} . The residual helium has been removed from the experimental apparatus. A device to produce a smaller helium leakrate has been made. Calibration tests are in progress; when they are completed, smaller volume ratios of helium will be injected and leakrate response time and quantity will be measured.

f. Physicochemical Mechanics of Metals. Experiments seeking to determine and understand the nature of the grain-boundary-penetration-type attack observed on numerous stainless steel specimens that have been exposed to sodium are being conducted.

An investigation is in progress to observe the penetration of sodium into the grain boundaries of Type 304 stainless steel. The material for these studies consists of "as-received" EBR-II fuel-element tubing. The nominal 0.009-in.-wall tubing is being fabricated into 12 capsules which will be partially filled with sodium. These capsules will be placed, in vacuum, in a furnace and will be heated to 1200°F. At intervals of one week, two capsules will be removed from the furnace, cooled, and prepared for analysis.

The analytical procedure will consist of placing the sample horizontally on the microscope stage of the laser microspectrophotometer. When the laser is fired, a spectrograph will be recorded on Polaroid film. Successive exposures will be made by refocusing the microscope cross-hair on the base of the crater made by the laser in the wall of the tube. The spectrographs will be recorded on the plate and will be compared to a reference spectrum of sodium consisting of the 5890-5895.5 Å doublet. Positional measurement of the microscope stage will be made by means of a dial indicator attached to the microscope. As the tube wall is progressively penetrated, the dial-indicator readings will be correlated with the spectrographs and the appearance of the sodium lines will show the depth to which sodium has penetrated, if such penetration occurs.

In preliminary tests, the tubing wall was penetrated in 18 steps with an average power setting of the laser. Each penetration averaged 0.00047 in., with an average positive deviation of 0.00025 in., an uncertainty factor of 53%. The latter is expected to be improved by the elimination of the mylar covering used to protect the microscope objective.

g. Sodium Effects on Wear Properties. To develop material combinations for load-bearing mechanism components for use in hot sodium, the effects, interrelationships, and controlling relationships of bearing performance, friction coefficients, sodium lubricity, wetting characteristics,

and surface hardness and finish are being investigated for various bearing compositions in weartests in a gas and in static and dynamic sodium.

The facility to perform the dynamic work, the Component and Material Evaluation Loop (CAMEL), has undergone complete electrical, electronic and data-logging checkout.

2. Fuel Handling, Vessels, and Internals

a. Core Component Test Loop (CCTL). The CCTL sodium facility is being prepared and modified for engineering tests of materials, instrumentation, controls, and core components for the FFTF and overall LMFBR Program. To determine experimentally the adequacy of piping stress-analysis methods for large high-temperature sodium systems, of which the CCTL is an example, strain-gauge readings on the operating system will be evaluated.

The original design of the CCTL loop omitted a flow-control valve because pump-motor speed would be adequate to control flow in the tests planned. However, for tests now contemplated in support of FFTF, a 4-in. Annin Co. globe, angle, bellows flow-control valve will be installed inside the pressure vessel in the 4-in. pump-discharge line; it will be actuated by an external handwheel through an angle-gear drive. The change requires modifying the vessel internals to accommodate the valve.

Work completed on this facility includes: (1) construction of deck cover over pit, including elevator-shaft enclosure, (2) installation of ductwork for the heat exchanger, (3) raising and semipermanently mounting the hood which was over the pressure vessel, (4) turning the test-vessel spool piece 90° for throttle-valve installation, (5) replacement of the gasket between pressure vessel and spool piece, (6) relocation of the thermo-couples on spool piece, and (7) vacuum check of the system after new installation.

In a 120-hr vacuum leaktest of the empty loop, the vacuum was 29.34 in. Hg at the start (atmospheric pressure 29.39 in. Hg) and 27.88 in. Hg at the conclusion (at which time atmospheric pressure was 29.10 in. Hg); room temperature stayed the same. The test affords confidence in the seal on the very large test vessel.

Work in progress includes: (1) taking samples of sodium from the supply tank for chemical analysis prior to filling loop, (2) continued installation of pressure-vessel internals, and (3) training personnel for operation of the loop.

Calibration of the strain-gauge test oven is continuing. It has been determined that the time needed to establish equilibrium inside the

oven depends on whether the control temperature is on its way up or down; the equilibration time is shorter when the control temperature is proceeding upward.

It takes longer to stabilize the testing frame protruding through the oven than to stabilize the oven itself. Quantitative data are being compiled to construct calibration graphs.

To correlate future strain readings and the effective stress in the material, tests have been initiated to determine the properties of piping material at high temperatures. The stress-strain diagrams are being sought at room temperature, 400, 800, and 1200°F. Soaking time in air has been specified as 1, 10, and 100 hr. Test samples may be exposed to sodium in later tests.

3. Sodium Technology Development

With the ultimate objective of developing standardized techniques that can be applied in commercial LMFBR power stations, methods of measuring and controlling sodium purity are being established and improved. The program proceeds from development of sophisticated procedures useful in the Laboratory to the evolution and verification of simpler industrial methods. As the program progresses, it also supports Laboratory programs involving loop tests, sodium environmental tests, EBR-II operation, and FFTF development.

a. Sodium Quality Measurement. Both laboratory and in-line methods are being developed for determining the content and form of various impurities, particularly oxygen, carbon, and hydrogen. The laboratory-techniques program involves developing and evaluating methods for taking samples from sodium systems, for transporting and processing the samples, and for measuring the content and form of various impurities. Measurement methods being investigated involve vacuum distillation, spectrophotometry, and activation analysis. In-line oxygen-in-sodium monitoring methods and devices are being evaluated and compared on a specially controlled Sodium Analytical Loop. The devices include plugging meters, absolute resistivity meters, electrochemical cells, in-line bypass-tube vacuum-distillation samplers, and vacuum-distillation samplers.

A residual-gas analyzer tube has been installed in a sodium vacuum-distillation apparatus; this arrangement provides the capability for gas analysis during a distillation run. The vacuum system was modified to prevent exceeding the safety cut-out pressure of 10^{-4} Torr if gas bursts (hydrogen from the decomposition of any NaH present in sodium, release of dissolved gases, hydrocarbon release, etc.) occur during distillation. Tests are determining optimum resolution-sensitivity settings for

various gases within the mass range of 1 to 50 amu. Information initially generated is expected to provide only a qualitative indication of gas or gases released, as well as some information on decomposition temperatures and release rates.

The Sodium Analytical Loop was filled with sodium and circulation started. Flow was achieved in all three side loops; temperature scans for 600°F operation are in progress to determine control settings for the system. Good wetting of the sodium to the metal walls was achieved in all flowmeters after a few days. Observation through the viewing port of the main vessel showed a clean mirror surface on the sodium, with slight agitation created by the recirculating loop; the exit spout of the analysis loop can be seen as it pours sodium into a standpipe, where the ensuing overflow creates a ripple on the surface of the sodium.

Numerous plugging runs have been made with good results on an adjustable-range two-pen recorder.

The hydrogen content of the blanket gas is being reduced by evacuation. Hydrogen is being analyzed qualitatively by a residual-gas analyzer. Chemical analysis of the sodium will be made for record purposes after the system has operated for at least a month.

b. Sodium Quality Control. Methods are being evaluated for removing detrimental oxygen, carbon, hydrogen, and gross fission products from sodium by means of cold traps, hot traps, and solution gettering. Purification capabilities are being improved to meet requirements as they become defined; removal methods are being simplified. To eliminate the need for piping systems to convey sodium from the primary tank to external cold traps, a cooled-mesh-type cold trap that can be submerged in the large primary tank is being developed.

A model unit of a submersible sodium oxide collector is being constructed. This device will be tested in a very large sodium tank to determine its capabilities to remove sodium oxide. A concurrent program is underway to devise methods to determine the oxide concentration in sodium and sodium oxide mixtures. Direct reaction with water, with subsequent measurement of hydrogen evolution, is being tested. This system, when applied to reactor primary sodium systems, could eliminate radioactive sodium piping and simultaneously reduce the probability of malfunctions.

c. Characterization of Nitrogen-bearing Species in Sodium:
Solubility of Nitrogen in Liquid Sodium. Information concerning the nature and concentration of nitrogen-bearing impurities in sodium is sparse. Occasional nitridation of structural materials in contact with sodium systems at elevated temperatures, however, has been observed. Because of

the importance of understanding and controlling nitridation, the solubility of nitrogen in sodium is being measured.

The solubility, which is very low, is determined by a procedure that involves saturating liquid sodium at a preselected temperature with N^{15} -enriched nitrogen gas at a known pressure, stripping the dissolved gas with helium, trapping the stripped gas, and assaying it with a mass spectrometer. The enriched nitrogen is used to preclude errors caused by in-leakage of atmospheric nitrogen into the system. In several experiments, it was demonstrated that known quantities of N^{15} -enriched nitrogen gas introduced into the system could be recovered quantitatively (95 to 100%).

A preliminary solubility experiment has been completed at 500°C and 8.2 atm pressure. The mole-fraction solubility of N_2 per atmosphere, i.e., the Henry's law constant, was found to be $1.5 \times 10^{-10} \text{ atm}^{-1}$. Further solubility measurements will be made between 300 and 600°C and at nitrogen pressures of 2 to 10 atm.

D. Fuel Development

1. Metallic Fuels

a. Performance of U-Pu Alloy Fuel. Irradiation of two fuel elements (of U-15 w/o Pu-10 w/o Zr and U-15 w/o Pu-10 w/o Ti) with V-20 w/o Ti jackets in subassembly XG05 is continuing in EBR-II. These fuel elements with current accumulated burnup are identified in Table XVIII.

TABLE XVIII. Status of Metal-fuel Irradiations (Clad in V-20 w/o Ti) in Progress in Subassembly XG05 in EBR-II

Spec. No.	Fuel Composition (w/o)	Design Parameters			Operating Conditions			
		Effective Density (%)	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
							a/o	fiss/cc x 10 ⁻²⁰ ^a
ND 24	U-15 Pu-10 Zr	63.6	0.208	0.015	10.0	535	5.4	12.4
NC 17	U-15 Pu-10 Ti	66.7	0.204	0.016	10.0	540	5.6	12.9

^aBased on effective density.

Group M-3 (see Table XVIII, p. 70, of Progress Report for May 1967, ANL-7342), consisting of fifteen U-15 w/o Pu-10 w/o Zr fuel elements is ready to be put into EBR-II for irradiation to a maximum burnup of 10 a/o. Calculations indicate a maximum jacket temperature of 630°C. These elements will be irradiated in subassembly XO28.

The jacket materials and capsule parts for Group M-4 (see Progress Report for June 1967, ANL-7349, p. 76) have been obtained.

2. Oxide Fuels

a. Mixed-oxide Fuel and Cermet Performance. The status of elements being irradiated in EBR-II are listed in Tables XIX and XX. Reactor operating conditions, calculated from data published in the Guide for Irradiation Experiments in EBR-II, have been adjusted in accordance with revised EBR-II flux values, as well as a number of analytical burnup results on experimental rods. The best-known values, at the present time, for linear heat-generation rates and cladding temperatures are listed.

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

Spec. No.	Design Parameters				Operating Conditions			
	Effective Density (%)	Cladding Composition	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding 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	605	3.7	7.6
SOV-7	85.1	304 SS	0.296	0.021	21.3	605	3.6	7.5
SOV-1	79.9	304 SS	0.296	0.021	20.0	590	3.6	7.1
HOV-10	84.8	Hastelloy-X	0.295	0.016	22.8	605	3.5	7.4
HOV-15	79.8	Hastelloy-X	0.295	0.016	21.4	590	3.5	6.9
HOV-4	80.0	Hastelloy-X	0.295	0.016	22.9	605	3.6	7.2
TVOV-1	76.8	V-20 w/o Ti	0.298	0.022	19.6	575	3.6	6.8

^aBased on effective density.

TABLE XX. Status of Type 304 Stainless Steel-clad, Coextruded Cermet-fuel Irradiations in Progress in Subassembly XO11 in EBR-II

Specimen No.	Design Parameters				Operating Conditions			
	Fuel Composition (w/o)	Effective Density (%)	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
							a/o	fiss/cc x 10 ⁻²⁰ ^a
5P-9	SS-40 PuO ₂	98.0	0.301	0.015	10.4	480	4.3	3.2
5P-12	SS-27 PuO ₂	98.7	0.294	0.015	6.7	440	4.3	2.1
5U-14	SS-27 UO ₂	97.9	0.298	0.013	5.8	430	3.2	1.7

^aBased on effective density.

Subassembly XO11 which contained the elements listed in Tables XIX and XX was removed from the reactor core 28 June 1967. The subassembly will be dismantled. The capsules will be reinserted in another subassembly, and the irradiation will eventually be continued.

The postirradiation examination of element SOV-6 (see Table XL in Progress Report for June 1967, ANL-7349, p. 77) is in progress. A section from the center of the rod has been removed from which small cores will be obtained for electron-microprobe studies.

3. Carbide Fuels

a. Mixed-carbide-fuel Irradiations in EBR-II. The status of 15 elements being irradiated in EBR-II is given in Table XXI.

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

S/A No.	Specimen No.	Design Parameters						Operating Conditions			
		Fuel Composition (w/o)	Fuel Form ^a	Effective Density (%)	Cladding Composition	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding 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	675	5.9	16.5
XG05	HMV-5	UC-20 PuC	VIPAC	80.0	Hast.-X	0.295	0.015	26.6	685	6.2	16.2
XG05	NMV-11	(U _{0.8} -Pu _{0.2})C	VIPAC	82.9	Nb-1 w/o Zr	0.284	0.012	25.5	635	6.2	16.9
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	3.7	9.9
XO08	NMV-4	UC-20 PuC	VIPAC	80.0	Nb-1 w/o Zr	0.283	0.013	27.0	645	5.5	14.5
XO08	NMV-7	UC-20 PuC	VIPAC	80.0	Nb-1 w/o Zr	0.283	0.013	26.0	630	5.4	14.3
XO08	NMV-12	(U _{0.8} -Pu _{0.2})C	VIPAC	85.5	Nb-1 w/o Zr	0.284	0.014	26.7	645	5.5	15.4
XO08	HMV-1	UC-20 PuC	VIPAC	80.0	Hast.-X	0.295	0.015	25.2	660	5.0	13.2
XO08	HMV-4	UC-20 PuC	VIPAC	80.0	Hast.-X	0.295	0.016	27.2	690	5.4	14.2
XO08	HWMP-1	(U _{0.8} -Pu _{0.2})C	PELLET	81.4	Hast.-X + W	0.296	0.020	16.9	565	3.6	9.8
XO08	HWMP-1	(U _{0.8} -Pu _{0.2})C	VIPAC	82.9	Hast.-X + W	0.295	0.018	26.4	685	5.4	14.8
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	1.1	2.9
XO15	NMV-3	UC-20 PuC	VIPAC	80.0	Nb-1 w/o Zr	0.284	0.013	26.0	635	1.6	4.3
XO15	HMV-2	UC-20 PuC	VIPAC	80.0	Hast.-X	0.295	0.015	27.0	685	1.6	4.2
XO15	TVMV-1	(U _{0.8} -Pu _{0.2})C	VIPAC	83.7	V-20 w/o Ti	0.298	0.021	23.4	640	1.5	4.0

^aVIPAC = Vibratorily Compacted into Cladding; PELLET = Pressed and Sintered Pellets.^bBased on effective density.

Metallographic examinations of four specimens removed from element number SMP-1 (see Table XLIII in Progress Report for June 1967, ANL-7349, p. 80), containing pellets in Type 316 stainless steel cladding, disclosed structural similarities between the two elements containing the same solid-solution fuel but in different initial forms. During irradiation, sintering occurred over as much as 65% of the diameter of the vibratorily compacted solid-solution fuel element (VMV-1) producing a structure similar to that of the solid-solution pellet fuel.

4. Fuel Cladding and Structure

a. Fabrication of V-15 w/o Cr-5 w/o Ti. A production size (3.8-in.-dia) ingot of V-15 w/o Cr-5 w/o Ti had been consolidated and extruded for use in the development of secondary fabrication operations (see Progress Report for May 1967, ANL-7342, pp. 77-79), and a means of working the alloy at room temperature was being sought.

The secondary extrusion of V-15 w/o Cr-5 w/o Ti sheet-bar, rod and tube blanks, from ingot AM-103, has been completed.

High yields of semifinished tubing (of 0.51-in. OD and 0.44-in. ID) have been obtained from the tube-reduction schedule shown in Table XXII. The annealed⁶ tube-reduced stock is then mandrel drawn in three passes (~51% reduction) to a 0.38-in.-OD, 0.34-in.-ID tube size. Following an intermediate anneal, two mandrel and three plug passes (~48% reduction) are used to reach a typical finished tube size (of 0.290-in. OD and 0.260-in. ID). A sizeable quantity of high-quality finished tubing has resulted from this schedule.

TABLE XXII. ANL Tube-reduction Data for V-15 w/o Cr-5 w/o Ti As-extruded Tube Blanks^a
(Starting Size: ~0.860 OD x ~0.630 ID x ~36 long)

Pass Schedule	Reducer Roll Diameter	Mandrel Diameter	Working Temp	Reduced Tube Size		Total Reduction between Anneals (%)	
				OD	ID	Area	Wall
1 ^b	0.875 → 0.750	0.581	200°C	0.760	0.587	32	24.7
2	0.750 → 0.625	0.500	200°C	0.638	0.507	56.2	43
3 ^c	0.625 → 0.562	0.484	Room Temp	0.575	0.491	40.2	35.8
4	0.562 → 0.500	0.437	Room Temp	0.510	0.442	56.7	48

^aAll dimensions are in inches.

^bExtruded tube blanks are annealed at 850°C for 2 hr prior to tube reduction.

^cTube-reduced blanks are annealed at 800°C for 1 hr prior to pass schedules 3 and 4.

The cold working and recrystallization behavior of the V-15 w/o Cr-5 w/o Ti alloy is being studied as a conjunctive phase of the above fabrication development. Hardness changes with cold working are shown in Fig. 8. The effect of isochronal annealing (one hour at temperatures ranging from 300 to 1500°C) upon the hardness of the cold worked alloy is illustrated in Fig. 9. Above 800°C there is little significant difference between the two reductions (50 to 70%): A minimum hardness is reached at ~900°C, remaining unchanged up to 1500°C. A metallographic evaluation and grain-size determination is planned.

(i) Tube-burst Tests on V-15 w/o Ti-7.5 w/o Cr Alloy Tubing. A series of tube burst tests at 550°C have been made on eight specimens from a V-15 w/o Ti-7.5 w/o Cr alloy tube irradiated in EBR-II. The results of the tests are shown in Table XXIII where they are related to calculated fluences. Irradiation temperatures were calculated to be in the vicinity of

⁶Annealed in the long-tube induction furnace at 900°C while moving at 0.5 in.

600°C. Strength tends to increase with increasing fluence. Some non-conforming data values for strength are evident. All values for ductility as determined from diameter measurements are low.

Fig. 8. Hardness Change of V-15 w/o Cr-5 w/o Ti as a Function of Percent Reduction

Fig. 9. Hardness Changes of Cold-rolled V-15 w/o Cr-5 w/o Ti after One-hour Anneals at the Indicated Temperatures

TABLE XXIII. Burst Tests on Irradiated V-15 w/o Ti-7.5 w/o Cr Alloy Tubing at 550°C

Calculated Fluence, $n/cm^2 \times 10^{-20}$	Max Tangential Strength		% $\Delta D/D$ (Avg)
	psi	kg/mm ²	
0	100,100	70.4	a
0	101,300	71.3	a
3-4	110,500	77.7	2.8
4-9	92,500 ^b	65.0	1.0
9-18	112,700	79.2	1.4
18-27	116,000	81.6	1.5
27-49	122,200	85.9	0.4
32-47	113,800	80.0	1.4
47-56	128,000	90.0	2.1
49-56	118,200	83.1	1.2

^aSpecimen shattered.

^bPossible defect in specimen.

b. Development of Refractory-metal Alloys for Service in Oxygen-contaminated Sodium. Vanadium-base alloys were reported (see Progress Report for June 1967, ANL-7349, pp. 80-81) to lose weight in static low-oxygen sodium at 450°C and to gain weight at higher temperatures. After about three months of static exposure of V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr, and V-15 w/o Cr-5 w/o Ti in sodium at 450°C the static tests are essentially complete. Weight losses were recorded for systems cold-trapped at 110°C and 175°C, respectively. Of the alloys tested, V-20 w/o Ti

incurred the smallest weight losses in both systems, just as was the case at 550°C. The corrosion film, being amorphous, was not amenable to X-ray analysis and was loosely adherent.

In dynamic tests V-15 w/o Cr-5 w/o Ti in sodium cold-trapped at 110°C was reported (see ANL-7349, p. 81) to gain slightly in weight. Since then the dynamic system was opened in an inert atmosphere long enough for a single inspection. Both gains and losses (thus far less than $<0.2 \text{ mg/cm}^2$ for 27.6 days) occurred among the various alloys.

Initial design of a facility for tests of corrosion mass transfer combined with tests of mechanical properties of vanadium alloys has been completed. Component parts have been selected. Their procurement and construction will continue through December.

c. Irradiation of Jacket Materials. Tensile tests were conducted on specimens of V-20 w/o Ti and V-15 w/o Ti-7.5 w/o Cr at temperatures of 400 and 550°C. Additional tests at 650°C are in progress. The specimens were irradiated in EBR-II subassembly XO13 to a nominal fluence of $5 \times 10^{21} \text{ n/cm}^2$ in the temperature range from 510 to 704°C. The V-20 w/o Ti alloy specimens had grain diameters of 11, 20, and 35 μ . The V-15 w/o Ti-7.5 w/o Cr alloy specimens had grain diameters of 11, 25, and 35 μ . The results of the tests are being evaluated.

d. Analysis of Oxygen and Carbonate in Sodium-Nickel Mass-transfer Loop. A pumped-sodium loop for mass-transfer study of nickel-base alloys has been constructed and successfully operated. The loop (see Fig. 10) has the unique feature that the test section for high-temperature ($\geq 350^\circ\text{C}$) sodium is constructed of essentially inert refractory material, while the components in contact with low-temperature ($< 350^\circ\text{C}$) sodium are made of Type 304 stainless steel. The test section consists of two concentric pipes fabricated of molybdenum and TZM (molybdenum-base alloy). This loop is so designed that purified (free of test-alloy constituents) sodium is continually supplied to the test section. The parameters for mass flow (0.5 to 3 gpm), linear velocity (0.5 to 10 m/sec), temperature (350 to 750°C), temperature gradient Δt (100 to 300°C), and oxygen concentrations (1 to 30 ppm by wt) can be precisely measured and controlled. Supplying purified sodium (from the cold trap via the sodium reservoir) to the inert test section will make it possible to evaluate the "downstream" effect observed in loops constructed completely of austenitic stainless steel.

However, before any preliminary mass-transfer experiments with nickel specimens at 650°C and nickel crystallizers at 350°C (to study thermal-gradient effect) or titanium crystallizers (to study chemical-gradient effect) can be performed, precise determinations of metallic and nonmetallic impurities in the flowing sodium are required. Special effort

is needed to develop the analytical techniques for these measurements, since there are no standard, generally acceptable procedures available. In the pumped-sodium loop, a nickel cup is dipped into the sodium reservoir at 325 to 345°C and transferred to the Pyrex tube chamber in an argon atmosphere. When the cup is cooled to below 120°C, the argon atmosphere is evacuated to 1- μ pressure before applying induction heat to distill the sample sodium. For a 12.7-g sample the total distillation time is about 2-1/2 hr, with approximately 1-1/4 hr at 345°C and another 1/2 hr at a maximum temperature of 385°C to insure complete distillation.

Fig. 10. Nickel Mass-transfer Loop

The distillation residue is hydrolyzed with 15 ml of water, which has been boiled and adjusted to pH 8.0. Then the solution of soluble residue is titrated with 0.001N HCl to pH 8.0 and pH 4.5 for determining sodium oxide and carbonate, respectively. Finally, the titrated solution is analyzed for total sodium and metallic impurities by flame spectrophotometric technique. The total oxygen determined is plotted against the emf generated by the electrochemical cells (United Nuclear Corp. oxygen meters), located at the exit of the sodium reservoir. In addition, the carbonate and carbide content

is separately determined by hydrolyzing the distilled residue with 2N sulfuric acid and measuring, by gas chromatography, the carbon dioxide and acetylene liberated. Preliminary test results for oxygen and carbonate in the loop sodium at various cold-trap temperatures are summarized in Table XXIV. These results are discussed in the light of (a) distillation temperature, (b) oxygen-meter calibration, and (c) carbon dioxide pickup.

TABLE XXIV. Results of Sodium Analyses for Oxygen and Carbonate

Run No.	18	16	15	14	13	8	7	6	5	3	2	1	12	11	
System															
Reservoir, °C	320	325	325	329	325	325	325	348	345	345	340	345	328	330	
Cold Trap, °C	180	187	185	180	151	150	140	142	140	192	139	139	145	155	
UNC Cell #1, V	1.129	1.130	1.128	1.126	1.145	1.143	1.145	1.143	1.145	1.136	1.148	1.142	1.145	1.146	
UNC Cell #2, V	1.130	1.131	1.129	1.127	1.145	1.143	1.143	1.143	1.141	1.132	1.143	1.139	1.144	1.146	
Cell Flow, gpm	0.55	0.55	0.55	0.50	~0.40	0.60	0.50	0.40	0.3	0.3	0.3	0.3	<0.3	<0.3	
Distillation															
Max Temp, °C	← 385 →				420				540	← 500 →				385	385
Residue Atmosphere	← Argon →								Air						
Solubility Data															
Na ₂ O, ppm	21.7	30.0	26.3	21.8	8.9	8.1	5.8	5.8	5.8	31.8	5.4	5.4	6.2	10.1	
Titration															
Na ₂ O, ppm	28.8	39.0	-	26.1	5.3	7.3	-	9.8	0	22.2	5.2	7.1	5.9	11.6	
NaHCO ₃ , ppm	21.3	19.9	-	70.5	34.2	35.8	-	38.2	-	47.7	32.0	37.2	53.5	74.5	
Total O, ppm	9.2	12.0	-	13.4	5.1	5.3	-	6.1	-	10.3	4.3	5.3	10.2	10.1	
Total Na, ppm	26.5	34.4	-	38.6	13.7	15.0	-	17.8	-	29.6	12.6	15.4	29.2	29.2	
Carbonate C, ppm	2.7	2.7	-	-	-	-	-	-	-	-	-	-	-	-	
Bicarbonate C, ppm	2.7	2.8	-	(10.1)	(4.8)	(5.1)	-	(5.5)	-	(6.8)	(4.6)	(5.3)	(7.6)	(10.6)	
Flame Absorption															
Total Na, ppm	a	a	a	40.8	13.4	15.2	-	20.6	6.0	-	-	-	27.8	27.9	
Gas Absorption															
Carbonate C, ppm	-	-	2.6	8.7	2.5	-	3.1	-	-	-	-	-	3.1	-	

^aTo be analyzed.

(i) Distillation Temperature: At 2 to 3 wt ppm carbonate carbon in sodium, there is no significant change in total oxygen at maximum distillation temperatures from 385 to 500°C. Table XXV shows that when the maximum temperature is raised to 540°C there appears to be a reduction in total oxygen. Similar observations were observed by Sinclair, Drummond, and Smith,⁷ who reported decomposition of carbonate to carbide and graphite plus soluble carbon at 550°C.

TABLE XXV. Effect of Distillation Temperature on Apparent Oxygen Concentration of Sodium

Run No.	Max Distillation Temp (°C)	Oxygen Meter Avg Voltage (V)	Total O (ppm)	Total Sodium (ppm)
5	540	1.143	2.1 ^a	6 (Flame)
2	500	1.145	4.3 (Titration)	12.6 (Titration)
6	420	1.143	6.1 (Titration)	20.6 (Flame)
8	385	1.143	5.3 (Titration)	15.2 (Flame)
13	385	1.145	5.1 (Titration)	13.4 (Flame)

^aCalculated from flame sodium.

⁷Sinclair, V. M., Drummond, J. L., and Smith, A. W., TRG Report 118(D), Dounreay, England.

(ii) Oxygen-meter Calibration: Figure 11 is a plot of the total oxygen concentration (ppm) versus emf (V) for the oxygen meter at 315°C. The sensor employs a thoria-base solid electrolyte and a copper-cuprous oxide reference electrode. The emf produced can be represented by the equation⁸

$$E = E_0 - \frac{RT}{nF} \ln \frac{C}{C_0},$$

where

- E = cell potential in volts;
- E_0 = standard potential in volts;
- R = 8.31 volt coulomb (g-mole)⁻¹°K⁻¹;
- T = absolute temperature, °K;
- n = 2 g-equivalent g-mole⁻¹;
- F = 96,500 coulomb g-equivalent⁻¹;
- C = concentration of oxygen in ppm;
- C_0 = the solubility of sodium oxide at 315°C in ppm oxygen.

Fig. 11. Comparison of Predicted and Observed Performance of UNC Oxygen Meters

⁸Kolodney, M., Minushkin, B., and Steinmetz, H., *Electrochem. Tech.* **3**, 244 (1965).

The quantity E_0 , calculated from thermodynamic data, is 1.072 V at 315°C. The value C_0 is obtained from solubility data. Thus the final equations are

$$E = 1.198 - 0.058 \log C \text{ from ANL Data:}^9 \quad C_0 = 151$$

$$E = 1.196 - 0.058 \log C \text{ from UNC Data:}^{10} \quad C_0 = 130$$

$$E = 1.206 - 0.058 \log C \text{ from LA Data:}^{11} \quad C_0 = 200$$

Note that the scatter of the experimental data is comparable to the scatter of these theoretical curves. The discrepancy for points of Runs 11 and 12 are attributed to low flow (<0.3 gpm).

(iii) Carbon Dioxide Pickup: Sodium in the distilled residue can absorb carbon dioxide to form sodium carbonate in air, or sodium bicarbonate in air and water. Table XXVI shows the erratic carbonate carbon concentration obtained in runs in which the distillate residue was exposed to air as compared with the consistent carbonate carbon concentrations obtained in runs in which the distillate residue was exposed to argon. These runs show there is about 2.5 ppm carbonate carbon in the loop sodium.

TABLE XXVI. Effect of Air Exposure on Apparent Carbonate Concentration of Sodium

Run No.	Residue Atmosphere	Oxygen-meter Avg Voltage (V)	Carbonate C (ppm)
15	Argon	1.129	2.6 (Gas Absorption)
18	Argon	1.130	2.7 (Titration)
16	Argon	1.131	2.7 (Titration)
14	Air	1.127	8.7 (Gas Absorption)
7	Air	1.144	3.1 (Gas Absorption)
13	Air	1.145	2.5 (Gas Absorption)

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

⁹Smith, D., and Kassner, T., ANL-7335 (to be published).

¹⁰Goldman, K., United Nuclear Corp., Elmsford, N.Y., private communication.

¹¹McPheeters, C.C., and Williams, J. M., LA-DC-7743 (SM-85/29) (1966).

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).

(i) Pilot-plant Operations: Plutonium Salt-transport Processes.

Preliminary experiments are being performed prior to the pilot-plant operations to develop more fully the technology of unit operations with the liquid metal-molten salt system. The first plutonium salt transport experiment (see Progress Report for June 1967, ANL-7349, pp. 85-86) has been completed with the exception of the steps in which the plutonium and uranium products are recovered by solvent-metal evaporation. A high-vacuum furnace is being prepared to carry out these final process steps.

Preliminary chemical analyses of the experiment show that 99.4% of the plutonium charged to the salt-transport step was transported in 14 cycles. After the salt-transport step the Zn-Mg-Pu alloy contained ~1 g of copper. After the alloy had been treated in the subsequent oxidation-reduction step for copper removal, only 20 ppm copper remained in the alloy. This value corresponds to the copper present in the reagents used to make the alloy, indicating essentially complete copper removal.

Two of the process steps in the current pyrochemical flowsheet were demonstrated during this period. In the first, the uranium wash step, the Cu-33 w/o Mg-U alloy from the plutonium salt-transport step was washed three times with liquid magnesium at 700°C. During each wash the Cu-Mg-U alloy was contacted with magnesium for 0.5 hr, the magnesium solution was allowed to settle, and the resultant Mg-Cu supernatant was pressure siphoned away from the uranium precipitate through a heated transfer line.

The other process step demonstrated was the plutonium-strip step, in which plutonium was removed from the 47.5 m/o MgCl_2 -47.5 m/o CaCl_2 -5 m/o CaF_2 reduction salt. Calcium fluoride salt and lithium and calcium metals were added to the reduction salt to change its composition and thereby lower the plutonium distribution coefficient (see Progress Report for April 1967, ANL-7329, p. 52). The salt was contacted with a Cu-36 w/o Mg alloy from the strip step of a previous run without plutonium. The contact was made at 700°C for 4 hr with 250-rpm agitation. The salt was then separated from the alloy and contacted with a fresh Cu-36 w/o Mg alloy to complete the strip step. Results of these two steps will be presented upon receipt of analytical data.

The experiment has progressed most satisfactorily. There have been no difficulties with equipment and no problems regarding containment of plutonium within the process vessels.

(ii) Process Chemistry of Molten-salt Systems. In support of the plutonium salt-transport process demonstration experiment, the distribution coefficient (ratio of w/o Pu in salt phase to w/o Pu in metal phase) of plutonium was measured between the reduction salt and the Cu-Mg donor alloy. The distribution coefficient of plutonium between the 47.5 m/o MgCl_2 -47.5 m/o CaCl_2 -5 m/o CaF_2 salt and the Cu-33 w/o Mg alloy at 700 and 800°C is about 0.3.

An oxide-reduction system consisting of a molten CaCl_2 - CaF_2 salt mixture and a liquid Cu-33 w/o Mg alloy containing calcium as the reductant is being considered for use in processing fast breeder reactor fuels. Because the salt in this system would be disposed of after one use, it is important to know to what extent various fertile, fissile, and fission product elements distribute to the salt. The distribution coefficients of rubidium and cerium between 775 and 850°C were studied in this system. Calcium oxide was added to the salt in amounts up to 20 w/o to represent the product of oxide reduction. Some scatter of the rubidium distribution data was noted. This scatter is attributed to salt contamination of metal samples when most of the rubidium is in the salt phase. The best estimate of the value of the distribution coefficient of rubidium is about 500, which is more than adequate to insure disposal of the rubidium in the waste salt stream. The cerium distribution coefficient varied from ~0.015 at 850°C to 0.008 at 775°C when the alloy contained 1 w/o calcium. Increasing the calcium content to 5 w/o decreased the distribution coefficient to 0.008 at 850°C, while adding CaO to the salt seemed to increase the distribution coefficient slightly. A comparison of the values obtained for cerium with theoretical estimates suggests that the cerium which is dissolved in the alloy reacts with CaF_2 in the salt to produce CeF_3 in the salt and calcium in the alloy.

A MgCl_2 - NaCl - CaCl_2 salt mixture and a Cu-33 w/o Mg alloy are being considered for the separation of plutonium from the rare earth fission products in the current process. In order to design a multistage extraction process, it is necessary to set the distribution coefficient for plutonium to a definite value according to the number of stages to be used and the desired plutonium recovery. This may be accomplished by adjusting the MgCl_2 content of the salt. Measurements were made of the distribution coefficient of plutonium between a 1:1 NaCl - CaCl_2 salt mixture with varying amounts of MgCl_2 and a Cu-33 w/o Mg alloy at 600°C. The MgCl_2 content of the salt was varied from 8 to 61 m/o. It was found that the distribution coefficient (K_d) of plutonium may be represented by the relation $K_d = 1.37 (x_{\text{MgCl}_2})^{3/2}$, where x_{MgCl_2} is the mole fraction of MgCl_2 in the salt phase.

The distribution coefficient of neptunium between a molten 50 m/o MgCl_2 -30 m/o NaCl -20 m/o KCl salt mixture and various Cd-Mg alloys was measured to obtain data for neptunium separation processes. Measurements were made at 650 and 700°C with alloy compositions

ranging from 0.7 to 81 w/o magnesium in cadmium. The value for the distribution coefficient ranged from about 3 to about 0.15 from the lowest to the highest magnesium concentration, respectively. The value at 700°C was slightly higher than the value at 650°C at each composition.

(iii) Process Chemistry of Liquid Metal Systems. The activity coefficients of cerium in liquid Mg-Ce alloys were computed utilizing data previously reported (see Progress Report for February 1967, ANL-7308, pp. 36-37) for the distribution of cerium between 50 m/o MgCl_2 -30 m/o NaCl-20 m/o KCl and liquid Zn-Mg alloys. The values computed are 6.5×10^{-3} at 700°C and 2.0×10^{-2} at 800°C. The values are based on a cerium concentration of ~0.07 a/o.

E. 1000-MWe Plant

Argonne is technically directing design studies by five industrial firms to identify the research and development needed for successful application of large liquid-metal-cooled fast breeder reactors for central-station power plants.

1. Contract Management and Liaison

The Laboratory is coordinating the activities of the contractors to assure that all studies are being conducted in accord with the established scope and common ground rules and background information.

a. Westinghouse (DEL). The contract with Westinghouse was established; it provides that work be done on a cost-plus-fixed-fee basis between January 30, 1967 and August 30, 1968. Westinghouse began work on the 1000-MW(e) Follow-on Program on January 30, 1967. During the pre-contract period, Westinghouse completed the requirements of project planning and administration and has been performing Task-I studies. The Westinghouse project planning and administration procedures have been reviewed to assure compliance to schedule, fiscal, and technical requirements of the study. Westinghouse has completed two technical notes concerning fuel-handling system concepts and one technical note concerning selection of fuel materials.

b. Atomics International. The contract with Atomics International was established; it provides that the work be done on a cost-plus-fixed-fee basis between June 15, 1967 and December 30, 1968. Methods of accomplishing and reporting the work were reviewed. The company has supplied first-draft PERT diagrams for Task-I activities. Work packages have been defined and detailed package descriptions have been drafted.

c. Combustion Engineering. The first quarterly technical report for the period from February 9 to May 9, 1967 was issued and is being reviewed. The major area of effort was the thermal and hydraulic analysis of the modular and nonmodular cores. This information will be used in evaluating the effects of small design changes and in establishing a reference design core for this study.

d. Babcock & Wilcox. A draft of the Task-I report has been received and is being reviewed. This report is scheduled for formal issue August 31. The B&W work scope for architect-engineer assistance was reviewed and approved.

e. General Electric. A meeting was held with the Byron-Jackson Pumps, Inc., to obtain information concerning large sodium pumps suitable for application in 300-1000-MWe plants and to provide B-J with large reactor system requirements for pumps. Present at the meeting were representatives from Byron-Jackson, General Electric, Atomics International (LMEC), Combustion Engineering, Argonne National Laboratory, and AEC.

F. General Research and Development

1. Fast Reactor Parameter Study

To provide guidelines for the critical evaluation of large fast breeder reactors and to provide uniform methods (such as fuel-cycle cost codes, and thermal and hydraulic analysis codes) for evaluating reactor systems, the physics and engineering characteristics of large oxide-, carbide-, and metal-fueled LMFBR cores of various geometries are being tabulated on a consistent design and analysis basis. Later, additional studies will be undertaken on the same consistent basis for steam- and gas-cooled reactor systems.

Single-core-zone carbide-fueled reactors have been analyzed with self-consistent diffusion theory, as was done earlier with oxide-fueled reactors. In both reference cases, 1.75 full-power years of operation was assumed. In the case of the carbide cores, however, additional calculations were made for 2.50 full-power-years to duplicate more closely the flow characteristics of the 1.75 full-power-year oxide reactors. Void coefficients were computed by k-calculations for 40% uniform removal of sodium from the 2.50 full-power-year cores. A corresponding self-consistent diffusion-theory analysis of metal fueled-reactors is underway.

A breeding ratio has been calculated from the output of a two-dimensional diffusion problem for the case of a reactor with a core height-to-diameter ratio (H/D) of 0.2 and a core power density of 400 kW/liter. In this problem, the corner area was made a separate region so that the

breeding ratio could be calculated both with and without the contribution of this corner region. The results could then be better compared with the values obtained using one-dimensional analysis. The two-dimensional core conversion ratio was 0.6804 versus 0.6748 for the one-dimensional analysis. The reactor breeding ratio without the corner contribution was 1.257 versus 1.248 for the one-dimensional analysis. These figures agree very well. The reactor breeding ratio including the corner contribution, obtained from two-dimensional calculations, was 1.306, or about 4% higher. This is to be expected because the one-dimensional analysis did not account for the corner region.

As a check of the one-dimensional results, a sodium-void coefficient has been obtained using two-dimensional diffusion theory for the case of the reactor with core H/D of 0.2 and core power density of 400 kW/liter. The value of k with full sodium was 1.00136 and with 40% of the sodium uniformly removed from the core was 1.00521. The difference gives a sodium-void coefficient of 0.00385. The one-dimensional result (using a k -calculation with a transverse buckling correction) was 0.00400. The discrepancy is less than 4%.

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Thermal-neutron Cross Sections for $\text{Ne}^{21}(\text{n}, \alpha)\text{O}^{18}$

Abramov and Yutkin¹² have measured the $\text{Ne}^{21}(\text{n}, \alpha)\text{O}^{18}$ cross section for thermal neutrons and suggested that this reaction would be useful for fast-neutron spectroscopy. In addition, an interest exists in the possible contribution of this reaction to excitation of a nuclear-pumped helium-neon laser. Consequently, this cross section has been remeasured.

Abramov and Yutkin detected pulses in a neon-filled ionization chamber which they attributed to the $\text{Ne}^{21}(\text{n}, \alpha)\text{O}^{18}$ reaction. Their calculations showed that the nuclear masses were such that this was the only exothermal reaction with emission of charged particles for neon. The observed reaction energy of 0.696 ± 0.019 MeV confirmed the calculated value of $Q = 0.704$ MeV. By comparison with $\text{He}^3(\text{n}, \text{p})\text{H}^3$, they obtained a cross section of 96 ± 33 b for thermal neutrons.

This reported cross section is inconsistent with a thermal-neutron absorption cross section of 32 ± 9 mb for neon, measured¹³ by the pile-oscillator technique. From the measured 96 ± 33 b for the $\text{Ne}^{21}(\text{n}, \alpha)$ reaction and the 0.257% isotopic abundance of Ne^{21} , a contribution of 247 ± 85 mb to the thermal-neutron absorption cross section for neon would be expected. This is higher by a factor of 8 than that observed.

The cross section for the $\text{Ne}^{21}(\text{n}, \alpha)\text{O}^{18}$ reaction has been determined by a comparison with the $\text{N}^{14}(\text{n}, \text{p})\text{C}^{14}$ reaction in a hydrogen-recoil proportional counter.¹⁴ The counter gas was dominantly neon, at a total pressure of 2 atm, with 15 cm CH_4 present as a quenching agent. The detector was exposed to the neutron flux in the thermal column of Argonne's Thermal Source Reactor, both with and without 0.4 cm N_2 present in the filling gas.

The location of peaks observed for both fillings occurred at essentially the same value of total ionization. The one containing nitrogen provided pulses equivalent to protons of energy about 615 keV. This is the more intense of the two distributions. The second, without nitrogen, produces a line about 0.27 as intense as the first. If the source of this line is the $\text{Ne}^{21}(\text{n}, \alpha)$ reaction, an upper limit of 1.5 b is calculated, based on

¹²Abramov, A. I., and Yutkin, M. G., The $\text{He}^{21}(\text{n}, \alpha)\text{O}^{18}$ Reaction on Slow Neutrons, J. Expt. Theoret. Phys. (USSR) **41**, 1023 (1961) [Transl. in Soviet Phys.--JETP **14**, 728 (1962)].

¹³Carre, J.-C., Robin, M., and Vidal, R., Measurement of the Absorption Cross Section of Natural Neon for Thermal Neutrons, Compt. rend. **252**, 3973 (1961).

¹⁴Bennett, E. F., Fast Neutron Spectroscopy by Proton-recoil Proportional Counting, Nucl. Sci. Eng. **27**, 16 (1967).

the value of 1.81 ± 0.05 b for the $N^{14}(n,p)C^{14}$ reaction with thermal neutrons, the number of nitrogen and neon atoms present in the two fillings, and the isotopic abundance of Ne^{21} . This corresponds to a contribution of ≤ 4 mb to the total thermal-neutron absorption cross section of 32 ± 9 mb for neon.

The calculated¹⁵ Q values are 699 keV for the $Ne^{21}(n,\alpha)$ reaction and 626 keV for $N^{14}(n,p)$. Since the difference is about three times the resolution of the counter, observation of the peaks at the same ionization is surprising. The average energy loss per ion pair formed would not be expected to be sufficiently different for alpha particles and protons to account for this. It is possible that the neon gas used here, and earlier, contained nitrogen as a contaminant. Only 0.1% N_2 present in the neon is required to account for the total reaction rate observed. This does not affect the upper limit of 1.5 b for the $Ne^{21}(n,\alpha)O^{18}$ reaction with thermal neutrons established by these measurements. This upper limit is less than 2% of the previously reported value.

The thermal-neutron cross section of ~ 5300 b for the $He^3(n,p)H^3$ reaction makes insignificant any contribution from the $Ne^{21}(n,\alpha)$ reaction in a neutron-pumped He^3 -Ne gas laser. Furthermore, the low cross section for the $Ne^{21}(n,\alpha)$ reaction makes its use for fast-neutron spectroscopy unattractive, compared with $He^3(n,p)$, even though it would lead to a four-fold improvement in the neutron energy at which recoil nuclei begin to interfere with the pulses from the reaction products.

2. Calculation of Doppler-broadened Line Shapes

Takano's suggestion¹⁶ for calculation of the Doppler-broadened line-shape function ψ has been tested with a simple FORTRAN routine on the CDC-3600. The results show that this method is competitive with the best FORTRAN methods in use. The test program yielded an estimated time to calculate and print a value of ψ of 9 msec. It is reasonable to expect that this time can be shortened by attention to programming details.

This method is suitable for producing a line profile and will be much slower if single values at different energies not in sequence are required.

Takano has modified Amster's suggestion¹⁷ by noting that the differential equation satisfied by $y = \psi \exp(\theta^2 x^2/4)$, where

¹⁵Mattauch, J. H. E., Thiel, W., and Wapstra, A. H., 1964 Atomic Mass Table, Nucl. Phys. 67, 1 (1965).

¹⁶Takano, H., The Computation of Doppler Broadened Functions, J. Nucl. Sci. and Technology (Japan) 4, 154-5 (1967).

¹⁷Amster, H. J., A Differential Equation for Calculating Doppler Broadened Resonances, Nucl. Sci. and Eng. 11, 343 (1963).

$$\psi(\theta, x) = \theta \int_{-\infty}^{\infty} \left\{ [\exp\{-\theta^2(x - y^2)/4\}] / (1 + y^2) \right\} dy / 2\sqrt{\pi},$$

is just $y'' = \theta^4 \{\exp(\theta^2 x^2/4) - y\}$ and can be solved by a predictor-corrector method.

3. Evaluation of Cross-section Data

Revisions were made in the ENDF/B cross-section data for titanium and vanadium following the suggestions of the Data Testing Subcommittee of the Cross Section Evaluation Working Group. An outline of these revisions follows.

a. Titanium. A $1/v$ capture cross section was assumed up to 100 eV based on $\sigma_{n\gamma} = 6.1$ b at 0.0253 eV,¹⁸ rather than the original value of 5.8 b. From 100 eV to 22.5 keV, the capture cross section was calculated from resonance parameters¹⁸ with an assumed Γ_{γ} of 0.75 eV, along with the $1/v$ contribution necessary to give agreement at 0.0253 eV; the single-level Breit-Wigner formula without Doppler broadening was used. From 22.5 keV to almost 1 MeV, the capture cross section was read from the smooth curve on the graph on p. 22-0-7 of Ref. 18. At higher energies, the values from the original evaluation were used. In the region of the resonance calculations, the capture cross section is in general quite different from that in the original compilation, which was based on Ref. 19. This is because the values in Ref. 19 are based on old resonance parameters, which have been greatly revised.

The total cross section up to 100 eV was revised by adding the original scattering cross section to the new capture cross section. The original σ_{nT} was used from 100 eV to 2.5 keV. From 2.5 to almost 220 keV, new values of total cross section were read from the graphs on pp. 22-0-2, 22-0-3, and 22-0-4 of Ref. 18. The original values were used at higher energies, except from 2.4 to 15 MeV, where values were read from the figures on p. 22-0-5 of Ref. 18.

The elastic-scattering cross section was revised above 100 eV by subtracting all other cross sections from the total cross section.

b. Vanadium. The capture cross section was taken as $1/v$ up to 1.5 eV, based on a value of 5.06 b at 0.0253 eV.¹⁸ The original evaluation assumed $\sigma_{n\gamma} = 4.98$ b at 0.0253 eV. From 1.5 eV up to 22.5 keV, the

¹⁸Goldberg, M. D., et al., Neutron Cross Sections, BNL-325, Second Edition, Supplement No. 2, Volume IIA, Z=21 to 40 (Feb 1966).

¹⁹Tralli, N., et al., Fast Neutron Cross Sections for Titanium, Potassium, Magnesium, Nitrogen, Aluminum, Silicon, Sodium, Oxygen and Manganese, UNC-5002 (Jan 1962).

capture cross section was calculated from parameters for the first two resonances in V-50 and the first six resonances in V-51, as given in Ref. 18, along with the $1/v$ cross section required to agree with the experimental cross section at 0.0253 eV. Values of 0.60 and 0.75 eV were assumed for Γ_γ for the V-50 and V-51 resonances, respectively. These values were suggested by Kapchigashev²⁰ to give agreement with his broad-resolution measurements of capture cross section shown on p. 23-0-3 of Ref. 18. Above 22.5 keV and up to 2 MeV, the capture cross section was read from the smooth curve for V-51 on p. 23-51-5 of Ref. 18. The values of the original evaluation were used at higher energies.

The total cross section below 100 eV was calculated as the sum of the new capture and scattering cross sections. The original values were used from 100 eV to about 2 keV. Above 2 and up to 220 keV, σ_{nT} was read from the graphs on pp. 23-0-3 and 23-0-4 of Ref. 18. Original values were then used up to about 0.4 MeV, followed by values from the smooth curves on pp. 23-0-5 and 23-0-6 of Ref. 18 at higher energies, except for the range from 1.3 to 2.0 MeV in which the original values were used.

New elastic-scattering cross sections were determined above 100 eV by subtracting all other cross sections from the total cross section.

A new $\sigma_{n\alpha}$ tabulation was read from the smooth curve on p. 23-51-8 of Ref. 18, with the low-energy portion being estimated. This smooth curve essentially follows later data from Hamburg rather than the Hamburg data used in the original compilation.

Since the inelastic cross section at high energies was found by subtracting the cross sections for other reactions from the calculated optical-model reaction cross section, the new $\sigma_{n\alpha}$ forced changes in σ_{nn} above the (n,α) threshold.

4. Calculation of Resonance Integrals in the Unresolved Region

An investigation is being carried out on the effect of using multilevel-multichannel analysis in the calculation of resonance integrals in the unresolved region. Cross sections are being generated using the multilevel formalism of Reich and Moore,²¹ which assumes one neutron and two fission channels. Comparison of the resonance integrals calculated using these cross sections and the usual Breit-Wigner single-level cross sections will be made.

²⁰ Kapchigashev, S. P., Cross Sections of V, Zr, Zr⁹⁰, Zr⁹¹ and Zr⁹⁴ for Radiative Capture of Neutrons with Energies 1-50,000 eV, Soviet Atomic Energy 19, 1212 (1965).

²¹ Reich, C. W., and Moore, M. S., Multilevel Formula for the Fission Process, Phys. Rev. 111, 929-933 (1958)

The various resonance parameters will be selected from chi-squared distributions of degree one for the reduced neutron widths Γ_n^0 and for each of the two fission widths Γ_{f1} and Γ_{f2} . The level spacings D will be determined from chi-squared distributions with two degrees of freedom.

Several sets of resonance parameters will be used so that uncertainties in using randomly generated parameters in the calculation of multilevel and single-level resonance integrals may be ascertained.

B. Reactor Fuels and Materials Development

1. Behavior of Reactor Materials

a. U-S-O System. An investigation of the UOS solid-solution area had indicated (see Progress Report for April 1967, ANL-7329, p. 63) that nearly 70 m/o of UO_2 is soluble in UOS at 1650°C , but that more UO_2 results in a second cubic phase having approximately the UO_2 parameters. The solubilities of sulfur in UO_2 or of US_2 in UOS are slight. UOS has a tetragonal PbFCl structure with atoms in planes normal to the c-axis stacked in an O-U-S-S-O-U-S-S... sequence. Conversion to the UO_2 (fluorite) structure not only involves approach of the UOS d-spacings of the $\{110\}$ and $\{101\}$ planes to the d-spacings of the UO_2 $\{200\}$ and $\{111\}$ planes, as the sulfur is replaced by oxygen, but requires an $a_0/2$ translation of the new oxygen layers. Increase of the UO_2 solute in the UOS at room temperature from 50 m/o to an amount less than 70 m/o changes the compound from a semiconductor to an insulator. Silver coatings, baked on the specimens at 610°C , reduce the interfacial resistance of probes used for conductivity measurement.

In a continuation of these tests, compositions were prepared at 6 m/o intervals in the range 50 m/o UOS-50 m/o UO_2 to 100 m/o UO_2 better to characterize this end of the UOS- UO_2 binary system. These were pelletized and sintered at 1750°C in vacuum for 3 hr. The first appearance of faint UO_2 reflections in the X-ray powder patterns indicated the maximum solubility of UO_2 in UOS to be 67.5 m/o. The lattice parameters for this terminal tetragonal solid solution were determined to be $a_0 = 3.860 \text{ \AA}$ and $c_0 = 5.895 \text{ \AA}$. This lends credence to the above solubility limit since, if the transition from the UOS structure ($a_0 = 3.844 \text{ \AA}$ and $c_0 = 6.694 \text{ \AA}$) to the UO_2 structure (cubic, $a_0 = 5.4711 \text{ \AA}$) were linear with oxygen substitution, parameters $a_0 = 3.860 \text{ \AA}$ and $c_0 = 5.878 \text{ \AA}$ would be expected at the 67.5-m/o composition.

The lattice parameter for UO_2 in equilibrium with the terminal solid solution remained essentially constant in the range of $a_0 = 5.4711 \pm 0.0002 \text{ \AA}$, suggesting negligible solubility of sulfur and

and near stoichiometry. (This is in agreement with earlier work done on the US- UO_2 system where, at 1800°C , UO_2 maintained a constant parameter of $a_0 = 5.4715 \text{ \AA}$ in equilibrium with US across the whole system, which appeared to be simple eutectic.)

The electrical resistivities of the above solid solutions increased only moderately from $2.5 \times 10^2 \Omega\text{-cm}$ for the 50 m/o UO_2 composition to $4.8 \times 10^{-2} \Omega\text{-cm}$ for the terminal solution containing 67.5 m/o UO_2 . In the two-phase region, the transition to the higher resistivity level of UO_2 ($2.16 \times 10^2 \Omega\text{-cm}$) occurred continuously over the range where 40 to 70 m/o UO_2 was in equilibrium with the terminal solid solution.

To investigate the homogeneity range of the US phase, ten compositions were prepared in the range $\text{US}_{0.925}$ to $\text{US}_{1.038}$ at $\text{S}_{0.0125}$ intervals. Carefully analyzed uranium and US_2 powders containing about 0.2 w/o oxygen were blended, pelletized, and sintered, some at 1750°C and some at 1850°C , for 3 hr in vacuum. Specimens were sintered in individual tungsten crucibles having solid lids, which kept weight losses down to about 0.1% and the compositions thus essentially as mixed. The object of this work is to relate systematic and quantitative changes in phase assemblage and lattice parameter to the phase boundary of the US area.

Both the grain size and bulk density increased considerably with increasing S:U ratio, and the appearances of the pellets varied markedly. This, of course, bore direct relation to the character of the secondary phases in equilibrium with the US phase. Presently known phase relations dictate that there are seven different possible combinations for these secondary phases. The grain-size and density ranges were 7 to 110μ and 8.65 to 9.16 g/cc (79.8 to 84.6% of full density), respectively, for the 1750°C series and 9.07 to 9.88 g/cc (83.7 to 91.2% of full density) for the 1850°C series. Bulk densities were determined quite accurately by Archimedes' method in water, using a thin film of Apiezon stopcock grease (later dissolved in CCl_4) to seal off the surface of the porous pellets.

Determinations of lattice parameter for the 1750°C series yielded somewhat puzzling data which suggest that US has only a short homogeneity range on the hypostoichiometric side. All of the compositions except two (which were in the range from $\text{US}_{0.98}$ to $\text{US}_{1.00}$) had almost identical parameters falling in the range of $a_0 = 5.4870 \pm 0.0003 \text{ \AA}$. The two exceptions showed significantly higher parameters of 5.4889 and 5.4884 $\text{\AA}$. Polished sections showed these two compositions contained only UOS solid solution as a secondary phase, while compositions below $\text{US}_{0.98}$ had uranium and UO_2 as secondary phases; those above $\text{US}_{1.00}$ contained combinations of stoichiometric UOS and U_2S_3 . A point-count method is being used to determine these secondary phases quantitatively, and chemical analyses are in process. Other series from the $1500\text{--}2000^\circ\text{C}$ range must be similarly examined, as well as a few samples with lower oxygen content, before the homogeneity range can be satisfactorily defined.

It was initially hoped to use electrical-resistivity measurements as a tool in defining the homogeneity range. The four-point probe method, however, could not be made sufficiently precise to pick up small resistivity trends, at least for the relatively porous and small specimens from the 1750°C sintering. The trouble seemed to be in establishing the most effective end contact for the current leads. The smallest value obtained in numerous independent trials for the same specimen was therefore assumed to be most correct since it represented the largest effective conducting area for the sample. On this basis, the resistivities (corrected for porosity) for the 1750°C series pellets were found to vary from 170 to 200 $\mu\Omega$ -cm, somewhat lower values than those usually reported for US.

b. Corrosion Inhibition by Dissolved Getters in Liquid-sodium Environment. A dissolved getter loop for isothermal pumping of sodium at 650°C was completed in May and placed into operation at that time.

Preparation of test sodium for the dissolved getter (DG) loop has continued. Concentrations of metallic impurities in stock sodium have been determined, and various purification techniques have been applied. The results of cold-trapping techniques have been determined. The results of analysis of sodium purified by distillation are still to be received.

The results of spectroanalysis of stock sodium samples have been received. The accuracy is stated to be within a factor of two. The samples were prepared from various sources: (1) a one-pound quantity of stock reagent-grade sodium; (2) a large drum of DuPont reactor-grade sodium; (3) a five-gallon-pail quantity of DuPont reactor-grade sodium; and (4) previously used sodium from the "model tank" in Building D-206. All stocks except the last were found to contain several hundred weight ppm of calcium and somewhat greater than 100 wt ppm of potassium, as well as 25 to 50 wt ppm of strontium and smaller amounts of silicon. Concentrations of other metallic impurities of present interest were either below 10 wt ppm or below detection limits. Model-tank sodium was relatively pure, e.g., 3 wt ppm of calcium, except for 90 ppm of iron.

Results have been received of similar analyses made of sodium samples withdrawn from corrosion-testing systems which employed cold traps containing excess oxygen, through which sodium is circulated to control the oxygen concentration in the test sodium. The results show that such traps can effectively lower the above concentrations of calcium and strontium to less than 10 ppm; the potassium level was not greatly affected.

Although sodium thus purified is considered acceptable for the DG experiments, a sufficient quantity is not available. Preparation of suitable quantities of acceptable sodium for various programs is proceeding along several concurrent lines: (1) an oxygenation cold trap and recirculation-pump fixture is being prepared for attachment to drum quantities of stock sodium; (2) small-batch purification (from calcium) by oxygen addition to sodium in a portable container appears feasible after an initial experiment and is being investigated further; (3) distillation of model-tank sodium to reduce the iron concentration has been carried out both in a tantalum still presently being put into service and (separately) in a small stainless steel still made recently for the DG program. Distillate samples have been submitted for spectroanalysis.

The distilled product is expected to be suitable for DG loop loading. If the desired selective improvement is not obtained, use of model-tank sodium without further treatment is being considered. The sodium selected for the DG program will be further analyzed before use for nonmetallic impurities of interest.

c. Studies of Corrosion in Lithium at Elevated Temperatures.

A protective layer forms on tantalum in Li-1.5 a/o Si at 1200°C even when the bath was highly contaminated by air (see Progress Report for May 1967, ANL-7342, p. 91). Samples have been prepared for studying the mechanism of formation of the protective Ta_2Si layer that forms on the tantalum in lithium-silicon solution. Present equipment is being modified and a new high-vacuum furnace is being procured.

d. Irradiation of High-temperature Materials. A series of irradiations is in progress on ceramic fuel materials being developed under the high-temperature materials programs. The materials under irradiation presently include UC-20 w/o PuC, 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 XXVII.

TABLE XXVII. Status of Ceramic-fuel Irradiations in Progress 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 ^{-20b}
56-11	MV-2	UC-20 PuC	79	0.281	0.012	470	8.0	20.6
56-8	MV-3	UC-20 PuC	81	0.281	0.012	715	8.5	22.5
56-8	MV-5	UC-20 PuC	80	0.281	0.012	705	8.2	21.5
56-11	MV-6	UC-20 PuC	80	0.281	0.012	480	8.5	22.3
56-13	Z-4	UC-20 PuC	79	0.174	0.015	665	3.4	8.8
56-13	Z-5	UC-20 PuC	79	0.174	0.015	585	3.4	8.8
56-13	Z-7	UC-20 PuC	79	0.174	0.015	570	3.4	8.8
56-13	C-45	PuC	84	0.174	0.009	700	3.4	9.3
56-8	S-7	US	80	0.281	0.012	535	6.8	13.2
56-8	S-8	US	89	0.281	0.012	725	8.8	19.0
56-8	S-9	US	76	0.281	0.012	750	8.8	16.2
56-8	S-10	US	91	0.281	0.012	690	8.8	19.4
56-11	S-15	US	82	0.281	0.012	380	5.9	11.7
56-11	S-16	US	90	0.281	0.012	510	7.8	17.0
56-11	S-17	US	88	0.281	0.012	500	6.5	13.8
56-11	S-18	US	77	0.281	0.012	610	8.3	15.5

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

^bBased on effective density.

2. Chemistry of Irradiated Fuel Materials

a. Distribution of Gamma-emitting Fission Products in Irradiated Mixed-oxide Fuel. Cross-sectional cuts of a UO_2 -20 w/o PuO_2 fuel element (SOV-6²²) irradiated in EBR-II to an estimated burnup of 2.9 a/o (U + Pu) have been radially sampled to determine the distribution of the gamma-emitting fission products across the fuel element. The fuel element is 0.254 in. in diameter and has a central cavity approximately 0.070 in. in diameter.

The cuts were made 4, 6, and 9 in. from the bottom of the 12-in.-long fuel element. A total of five samples were taken along a radius of each cut by a 0.010-in.-dia microdrill. The drillings were collected by vacuum on Millipore filter papers and assayed for gamma-emitting fission products by means of a germanium-lithium drifted crystal spectrometer. The results of the measurements for the various fission products in the samples were normalized to cerium-144 and are given in Table XXVIII.

TABLE XXVIII. Ratio of Various Fission Product Activities to Ce^{144} Activity in Irradiated Mixed-oxide Fuel (SOV-6)

Sample No.	Cross-section Location	Distance ^a (in.)	$\text{Ce}^{141}/\text{Ce}^{144}$	$\text{Ru}^{103}/\text{Ce}^{144}$	$\text{Ru}^{106}/\text{Ce}^{144}$	$\text{Zr}^{95}/\text{Ce}^{144}$	$\text{Nb}^{95}/\text{Ce}^{144}$	$\text{Cs}^{137}/\text{Ce}^{144}$
1	6 in. from bottom	0.074	0.306	0.104	0.0421	0.200	0.95	<0.002
2		0.060	0.306	0.103	0.0352	0.182	0.79	<0.002
3		0.043	0.308	0.135	0.088	0.186	0.76	0.0021
4		0.020	0.314	0.129	0.074	0.191	0.84	0.0317
5		0.009	0.303	0.136	0.081	0.194	0.78	0.080
6	4 in. from bottom	0.081	0.326	0.107	0.0437	0.205	1.04	<0.002
7		0.057	0.303	0.097	0.0345	0.185	0.77	<0.002
8		0.043	0.314	0.130	0.084	0.186	0.77	0.0031
9		0.025	0.310	0.133	0.081	0.189	0.82	0.0158
10		0.006	0.307	0.135	0.079	0.196	0.86	0.145
11	9 in. from bottom	0.072	0.305	0.112	0.0364	0.197	0.93	<0.001
12		0.0495	0.302	0.137	0.071	0.179	0.74	0.002
13		0.0375	0.306	0.114	0.059	0.197	0.83	0.0036
14		0.023	0.312	0.135	0.072	0.195	0.85	0.0294
15		0.009	0.311	0.140	0.082	0.185	0.81	0.064

^aDistance from the periphery of fuel element.

Samples 1 and 5 (see Table XXVIII) were analyzed for uranium and plutonium mass spectrometrically by an isotopic dilution technique. Sample 1 was taken near the central cavity of the element, whereas Sample 5 was taken near the periphery of the element. The counts per minute per milligram of heavy metals (U + Pu) for the various fission products in the two samples are given in Table XXIX. The ratios of the counts per minute per milligram of the fission products in Sample 1 to those of the fission products in Sample 5 indicate that no migration of cerium or zirconium has taken place. The ratio for ruthenium shows some enrichment in the peripheral region. The difference in the ruthenium-103 and ruthenium-106 ratios can be predicted from the known diffusion of ruthenium and the difference in the half-lives of the two isotopes. Cesium-137

²²Data on the fabrication and metallographic examination of this fuel element were reported in Progress Reports ANL-7308, p. 28, and ANL-7317, p. 52.

was below the limit of detection in the inner sample (Sample 1), but was found in the outer sample (Sample 5). A slight enrichment of niobium in the inner region of the cut was observed.

TABLE XXIX. Comparison of Fission Product Content of Two Samples Taken from Cross Section of a Mixed-oxide Fuel Element (SOV-6)

Cross section cut 6 in. from bottom of fuel element

Sample 1 taken 0.074 in. from periphery of element

Sample 5 taken 0.009 in. from periphery of element

Fission Product	Counts/min/mg U + Pu		Ratio Sample 1/Sample 5
	Sample 1	Sample 5	
Cerium-141	4.90×10^4	4.71×10^4	1.04
Cerium-144	1.61×10^5	1.56×10^5	1.03
Ruthenium-103	1.67×10^4	2.11×10^4	0.78
Ruthenium-106	6.8×10^3	1.27×10^4	0.54
Zirconium-95	3.21×10^4	3.02×10^4	1.06
Niobium-95	1.53×10^5	1.22×10^5	1.25
Cesium-137	$<5 \times 10^2$	1.24×10^4	<0.04

The remainder of the samples will be analyzed for uranium and plutonium. Additional information on the distribution of fission products across the fuel element will be presented when these analyses are completed.

3. Thermodynamics of Fuel Materials

a. Vaporization of Uranium Monophosphide. The mass-spectrometric Knudsen effusion study of the high-temperature thermodynamic properties of uranium monophosphide (UP) has been completed. The experimental technique and preliminary results were reported previously (see Progress Report for November 1966, ANL-7279, pp. 47-51).

In the temperature range of this study, 2073-2423°K, phosphorus was found to vaporize preferentially from UP, leaving behind the two-phase system $U(l, \text{satd with P})-UP_{1-x}(c)$. The basic quantities measured were the partial pressures of U, P, and P_2 in equilibrium with this two-phase system. The two-phase boundary compositions of hypostoichiometric UP could not be determined accurately. However, from combined metallographic and chemical analyses, the boundary at 2277°K was found to have a P/U atom ratio between 0.98 and 0.95.²³ Accordingly, an average value of $UP_{0.965 \pm 0.015}$ was taken as the composition of the hypostoichiometric boundary in the temperature range of this study.

²³Bowman, M., Los Alamos Scientific Laboratory, reports that the hypostoichiometric range may extend down to $UP_{0.96}$ at 2173°K (private communication).

The partial-pressure equations derived for the range 2073-2423°K from the recent measurements are

$$\log p_U(\text{atm}) = (5.6767 \pm 0.0851) - (25,898 \pm 192)/T; \quad (1)$$

$$\log p_P(\text{atm}) = (8.0623 \pm 0.1501) - (31,831 \pm 340)/T; \quad (2)$$

$$\log p_{P_2}(\text{atm}) = (10.3186 \pm 0.6505) - (38,454 \pm 1480)/T. \quad (3)$$

From the uncertainty in the experimental data, the partial pressures are known (95% confidence level) to within 18, 20, and 44%, respectively. The corresponding apparent partial molar enthalpies of vaporization are (in kcal/mole) $\Delta H(U) = 118.5 \pm 0.9$, $\Delta H(P) = 145.7 \pm 1.6$, and $\Delta H(P_2) = 176.0 \pm 6.8$.

b. Enthalpy of Sublimation of Uranium. It is of interest to combine the enthalpy of the reaction

as calculated for 298°K from the high-temperature equilibrium data of the aforementioned study, with the recent calorimetrically determined enthalpy of formation of uranium monophosphide,²⁴ to obtain a value for the enthalpy of sublimation of α -uranium [$\Delta H_f^\circ(U,g)$]. Numerous papers have dealt with this important thermochemical value. Rauh and Thorn's²⁵ data yield $\Delta H_{f298}^\circ(U,g) = 117.2 \pm 0.1$ kcal/mole; however, higher values have been reported by Leitnaker and Godfrey,²⁶ Storms,²⁷ and Drowart *et al.*,²⁸ namely, $\Delta H_{f298}^\circ(U,g) = 123.7 \pm 1.3$, 125.7 ± 1.0 , and 125.5 ± 2.5 kcal/mole, respectively.

Calculations of the enthalpy of reaction 4 at 298°K by the second- and third-law methods yielded values of 264.0 ± 3.5 and 262.2 ± 4.7 kcal/mole, respectively. The auxiliary thermodynamic data for $U(g)$ and $P(g)$ used in the calculations were taken from Stull and Sinke.²⁹ The larger standard deviation of the third-law value compared to the second-law value reflects the uncertainty in the heat capacity for UP . [This uncertainty is, in fact, the greatest source of error in the overall calculation for $\Delta H_f^\circ(U,g)$.] An average value for the heat capacity of UP

²⁴O'Hare, P. A. G., Argonne National Laboratory (private communication).

²⁵Rauh, E. G., and Thorn, R. J., *J. Chem. Phys.* **22**, 1414 (1954).

²⁶Leitnaker, J. M., and Godfrey, T. G., *J. Nucl. Mater.* **21**, 175 (1967).

²⁷Storms, E. K., in *Thermodynamics*, Vol. I, IAEA, Vienna (1966), p. 309.

²⁸Drowart, J., Pattoret, A., and Smoes, S., *J. Chem. Phys.* **42**, 2629 (1965).

²⁹Stull, D. R., and Sinke, G. C., *Thermodynamic Properties of the Elements*, Advances in Chemistry Series, No. 18, American Chemical Society, Washington, D.C. (1956).

over the temperature range of integration (298-2300°K) was estimated to be 13.9 ± 0.8 cal/deg/mole. This estimate was obtained by combining the low-temperature (10-300°K) heat capacity measurements of Counsell et al.,³⁰ the intermediate-temperature (1073-1473°K) measurements of Brugger,³¹ and an extrapolation to higher temperatures. It was assumed in the extrapolation that the heat capacity curve for UP would be similar to those for UN³² and UC.²⁶ The value $S_{298}^{\circ}(\text{UP}_{0.965}) = 19.01 \pm 0.31$ eu was calculated by taking Counsell's³⁰ entropy value, $S_{298}^{\circ}(\text{UP}) = 18.71$ eu, and adding to it an estimate of the entropy of mixing of the phosphorus vacancies.

The enthalpy of sublimation of α -uranium at 298°K was calculated from a combination of the following values (in kcal/mole): the averaged enthalpy of reaction 4, $\Delta H_{298} = 263.1 \pm 4.7$; the enthalpy of formation²⁴ of $\text{UP}_{0.965}$, $\Delta H_{298}^{\circ} = -74.8 \pm 0.7$; the enthalpy of sublimation²⁹ of 0.965 P(red V), $\Delta H_{298} = 77.0 \pm 0.6$; and the enthalpy of the transition³³ 0.965 [P(red V) $\rightarrow$ P(α)], $\Delta H_{298} = 3.8 \pm 0.7$. The value obtained, $\Delta H_{298}^{\circ}(\text{U,g}) = 115.1 \pm 4.8$ (σ) kcal/mole, supports the enthalpy of sublimation reported by Rauh and Thorn.²⁵ However, with a 2σ uncertainty of 9.6 kcal/mole, the higher values²⁶⁻²⁸ reported for $\Delta H_{298}^{\circ}(\text{U,g})$ can not be excluded.

4. Radiation Damage of Structural Materials

a. History of EBR-II Program. For any period during which a sample has been in EBR-II, this program calculates the decay corrections so that foils may be analyzed or burnups can be calculated from fission product activity. The program, EBR-II History, is written in Fortran IV for the CDC 3600.

To use counting data from foils one must correct for radioactive decay from the time the foil was removed from the reactor until it is counted. However, decay also takes place during irradiation. Activity induced at the beginning of an irradiation will be underestimated if it is assumed to have been generated instantaneously at the time of removal. Obviously, for long-lived isotopes and short irradiations, the corrections are small. If not made, however, some error is deliberately introduced, and results become difficult to rationalize with those from other foils. For long periods of operation at various power levels these corrections are absolutely essential, and can be very tedious.

³⁰Counsell, J. F., Dell, R. M., Junkison, A. R., and Martin, J. F., Trans. Faraday Soc. 63, 72 (1967).

³¹Brugger, J. E., Chemical Engineering Division Research Highlights, May 1965-April 1966, ANL-7175 (1966), p. 142.

³²Godfrey, T. G., Wooley, J. A., and Leitnaker, J. M., ORNL-TM-1596 (1966).

³³O'Hare P. A. G., and Hubbard, W. N., Trans. Faraday Soc. 62, 2709 (1966).

Since activities are additive, one can simply compute the contribution of each day's induced activity, correct for its decay, and add all of them up. This program performs this manipulation for 12 isotopes for any period of operation of the EBR-II. These isotopes include the residue from the following three (n, γ) reactions, three (n,p) reactions, and six fission products (although others could be substituted).

<u>Isotope</u>	<u>Reaction</u>	<u>Residue</u>	<u>Half-life (days)</u>
Co ⁵⁹	(n, γ)	Co ⁶⁰	1925.
Fe ⁵⁸	(n, γ)	Fe ⁵⁹	45.
Ti ⁴⁶	(n,p)	Sc ⁴⁶	83.9
Fe ⁵⁴	(n,p)	Mn ⁵⁴	313.5
Ni ⁵⁸	(n,p)	Co ⁵⁸	71.3
Cr ⁵⁰	(n, γ)	Cr ⁵¹	27.8
	(FP)	Cs ¹³⁷	10647.
	(FP)	Mo ⁹⁹	2.75
	(FP)	Ce ¹⁴⁴	285.
	(FP)	Zr ⁹⁵	65.
	(FP)	Ba ¹⁴⁰	12.8
	(FP)	Sr ⁸⁹	51.

The operating history of EBR-II is stored in the program in units of thermal megawatt-seconds for each day since the beginning of July 1965. Continual updating is accomplished by adding cards to the history as the data become available.

The input consists of three dates: the date on which the sub-assembly went into the reactor, the date it came out, and a reference date or counting date, to which all decays are corrected. A sample printout for one case is shown in Fig. 12 for an experiment inserted January 9, 1966, removed December 31, 1966, and the activity-counting data referenced to July 1, 1967. This period includes all operation in 1966 and is made up of EBR-II runs 14 through 24.

For each such case, a line of data is printed out giving the date in, date out, and reference date, NIN, NOUT, and NREF (the number of each of these three dates counting consecutively from January 1, 1965), NDAYS (the number of days in-reactor), TOTAL MWS (megawatt-seconds of operation while in-reactor), and AVE MW (the average operating power over the period). The plant factor for the period can be calculated by dividing AVE MW by the designated full power level.

For each isotope EPMWS is the effective megawatt-seconds of operation. To use this figure, the counting data are first converted from disintegrations per second per milligram to active atoms per 10²⁴ atoms (multiply by the decay constant and divide by atoms/mg). Then,

dividing by EPMWS gives the rate of activations per second per 10^{24} atoms per megawatt of reactor power. This is the number needed to determine fluxes from foil results. Fluence is obtained by multiplying the instantaneous flux by TOTAL MWS.

CASE....EBR-II RUNS...DATE IN...DATE OUT...DATE REF....NIN...NOUT...NREF...NDAYS...TOTAL MWS.....AVE MW
 9 RUNS14/24 1/ 9/66 12/31/66 7/ 1/67 374 730 912 356 6.08527+008 1.97841+001

ISOTOPE CO59 (N,G) CO60	HALFLIFE 1.92500+003	EPMWS 5.37509+008	EFF 8.83295-001
ISOTOPE FE58 (N,G) FE59	HALFLIFE 4.50000+001	EPMWS 7.57022+006	EFF 1.24402-002
ISOTOPE TI46 (N,P) SC46	HALFLIFE 8.39000+001	EPMWS 4.73450+007	EFF 7.78027-002
ISOTOPE FE54 (N,P) MN54	HALFLIFE 3.13500+002	EPMWS 2.89590+008	EFF 4.75886-001
ISOTOPE NI58 (N,P) CO58	HALFLIFE 7.13000+001	EPMWS 3.18623+007	EFF 5.23597-002
ISOTOPE CR50 (N,G) CR51	HALFLIFE 2.78000+001	EPMWS 8.40591+005	EFF 1.38135-003
ISOTOPE CS137 (FP)	HALFLIFE 1.06470+004	EPMWS 5.94972+008	EFF 9.77724-001
ISOTOPE MO99 (FP)	HALFLIFE 2.75400+000	EPMWS 2.90364-014	EFF 4.77158-023
ISOTOPE CE144 (FP)	HALFLIFE 2.85000+002	EPMWS 2.69546+008	EFF 4.42948-001
ISOTOPE ZR95 (FP)	HALFLIFE 6.50000+001	EPMWS 2.48426+007	EFF 4.08241-002
ISOTOPE BA140 (FP)	HALFLIFE 1.28000+001	EPMWS 1.66157+003	EFF 2.73048-006
ISOTOPE SR89 (FP)	HALFLIFE 5.10000+001	EPMWS 1.17556+007	EFF 1.93838-002

Fig. 12

EFF is an efficiency ratio: $EPMWS/TOTAL\ MWS$. Note that EPMWS takes into account decay until the reference date. If one is interested only in the decay during irradiation, the reference date can be made the same as Date OUT.

b. Calculated and Measured Activation Rates in EBR-II. Before Run 25 began, several short-term, low-power runs were made. Foils exposed in Runs 25A and 25D for about 30 kWhr each were analyzed to provide the activation ratio plots. The Ni/Fe activation ratio does not change much through the entire core. The U^{238}/U^{235} fission ratio drops off somewhat as the spectrum softens near the blanket, and near the core-blanket interface the local flux variations will not be shown by the calculations.

Fig. 13. Fe^{54} Activation Distribution (Radial)

cutting a wire into small pieces and counting every one, thus establishing the flux shape. The probable cause of the deviation between the experimental data and the calculational results is that the calculations were based on a homogenized model of the core. Of course, the core itself consists of a vertical array of tubes; hence, in reality streaming will affect the flux distribution.

5. Basic Fabricability

a. New Fabrication Concepts--
ANL Three-roll Tube Reducer. The design and operational evaluation of the three-roll tube reducer (see Progress Report for March 1967, ANL-7317, p. 77), has been essentially completed for the use of moving mandrels. A wide range of materials have been so processed. Some of these include cast brass (sensitive to tensile stresses); Zircaloy IV (sensitive to lubrication practice); tungsten alloys (sensitive to all modes of deformation, particularly near room temperature); and other materials such as the vanadium alloys, as well as other Group-V metals and alloys that are somewhat less recalcitrant in their response to plastic deformation or processing reduction. Roll contour has been established and the effects of tube feed-rate (advance) and angular rotation (index) have been tentatively determined.

Figure 13 shows the $\text{Fe}^{54}(n,p)\text{Mn}^{54}$ activation rate at 1-MW power as a function of core radius. The agreement with calculation (MACH-1 Code at ANL and a transport-theory calculation by R. Dahl, PNWL) of the shape of the distribution is satisfactory. The datum point at 2.6 cm (Pin 1A-1) was used for the normalization.

Figure 14 shows results of calculations at this radius in the axial direction. The experimental curve was obtained by

The results of recent studies have shown that the use of a fixed plug (~8 in. in length) could easily replace the considerably longer mandrels previously used. These studies have reconfirmed earlier results, indicating that advancing the tube over the plug presents no problem, even with materials possessing severe galling tendencies.

Shop construction of a plug-feed mechanism is nearing completion. The use of this device, when installed on the tube-reducer, should establish the utility of this technique as compared with the moving mandrel method of tube-reduction.

b. Determination of Elastic Moduli of High-temperature Materials by Ultrasonics. Calculations of the elastic moduli of cold-rolled steel and of Type 304 stainless steel on the basis of ultrasonic velocities measured at temperatures up to 1000°C were reported (see Progress Report for June 1967, ANL-7349, p. 101) to yield results within 5% of values in the literature.

Rods of Type 304 stainless steel fabricated from steel used in EBR-II were received for measurements of elastic moduli. The specimens were designed for an older furnace which had become unusable. Instead of making the old furnace operational, modification of a new furnace seemed the best way to expedite the operation.

The plans were drawn up and fabrication has begun on a more versatile specimen holder to enable the testing of rods over a range of diameters with the least amount of effort and cost. Fabrication of the new specimen holder will be completed in the near future, and testing of the four specimens will begin at that time.

Measurements of elastic moduli at temperatures above 1100°C with short specimens mechanically bonded to a buffer rod is the next step in the furnace work. Obviously the test material used up to this point (Type 304 stainless steel) cannot be used, so a tungsten buffer rod, specimen, and coupler are being fabricated. A tungsten buffer rod and coupler will extend the short sample-buffer rod technique up to temperatures close to the melting point of tungsten. The tungsten to be used in the buffer rod was ultrasonically tested and X-rayed to insure defect free material before fabrication began.

c. Development of a Neutron-image Intensification System. Further analysis has been made of the use of the neutron television system as a possible tool for the study of the casting process. As reported previously (see Progress Report for April 1967, ANL-7329, pp. 67-68), this inspection system does seem to lend itself well to the study of heavy-metal casting. Although X-ray techniques have shown promise for casting

studies involving metals as heavy as steel,³⁴ the neutron technique would be preferred for heavy metals because of the relative ease of neutron penetration.

Figure 15 shows the result of a casting-study trial. Every other frame of a neutron television motion-picture recording of a sequence in which Wood's metal is poured into an aluminum mold is shown. The diameter of the mold opening is about 0.63 cm. One interesting point concerning the motion-picture record is that the metal drops down one side of the vertical member (frames numbered 4-6) before backfilling to fill that member (frames 7-10). The expansion chamber (directly above the vertical member) starts to fill before the vertical member is filled (frames 6-9). One would, therefore, expect to find trapped gas at that corner of the casting. Since the metal apparently freezes before filling the lower horizontal arm, one would also expect to find defects in that area.

Fig. 15. Individual Frames of a Motion-picture Recording Showing a Televised Neutron Transmission Image of the Casting of Wood's Metal into an Aluminum Mold. The sequence begins in frame 1, with the molten metal entering from the top.

Defects in both these areas have been observed on radiographs of castings made during these trials.

³⁴Rose, H., Dyke, W. P., Grundhauser, F. J., and Barbour, J. P., *X-ray Movies of Metal Casting*, *Materials Evaluation* 24, 92 (1966); Halmshaw, R., *Cine-Fluorographic Studies in the Production of Castings*, presented at the Fifth International Conference on Nondestructive Testing, Montreal, May 1967, to be published.

d. Measurement of Thermal Conductivity of Irradiated Fuel as a Function of Burnup and Temperature. Measurements of thermal diffusivity and conductivity have been completed for EBR-II Mark-I and Mark-IA (Type SL) fuel specimens. A longitudinally sectioned specimen (for measurements in the radial direction) and a transversely sectioned specimen (for measurements in the axial direction) were fabricated from the top, middle, and bottom of a fuel rod of each type. By sectioning the fuel rod in this manner, the results will determine if there are any differences in thermal properties in the axial and radial directions, or along the length of the fuel pin.

Table XXX gives the measured values for the diffusivity, α , and conductivity, K , of a specimen of EBR-II, Mark I, E47-Melt 114--Pin 12. Table XXXI gives the measured values for α and K of a specimen of EBR-II, Mark IA, SL32--Pin 32.

TABLE XXX. Thermal Diffusivity and Thermal Conductivity of a Specimen of EBR-II Mark-I Fuel

T (°C)	α (cm ² /sec)	$K \left(\frac{\text{cal cm}}{\text{cm}^2 \text{sec}^\circ\text{C}} \right)$
28	0.0675	0.0472
98	0.0713	0.0510
193	0.0766	0.0559
293	0.0829	0.0615
393	0.0865	0.0655
498	0.0910	0.0725
605	0.0989	0.0828

TABLE XXXI. Thermal Diffusivity and Thermal Conductivity of a Specimen of EBR-II, Mark-IA, Type SL Fuel

T (°C)	α (cm ² /sec)	$K \left(\frac{\text{cal cm}}{\text{cm}^2 \text{sec}^\circ\text{C}} \right)$
28	0.0716	0.0525
107	0.0733	0.0559
200	0.0784	0.0600
297	0.0825	0.0613
404	0.0885	0.0720
509	0.0911	0.0740
592	0.100	0.0812

The measured values of α and K for the six specimens of any one pin agreed within $\pm 3\%$, indicating that no variation in thermal properties exists. The conductivity of the Mark-I specimens is in excellent agreement with previously measured³⁵ conductivity of the U-2.5 w/o Fs alloy. The Mark IA (Type SL), however, has initially 10% higher α and K , but at about 300°C its values are the same as the Mark I, and they remain the same to the final test temperature 600°C.

³⁵Nevitt, M. V., and Zegler, S. T., Structures and Properties of Uranium-Fissium Alloys, ANL-6116 (1961).

C. Engineering Development

1. Development of Master-Slave Manipulator Systems

a. Electric Master-Slave Manipulator, Mark E4A. The slave arm has been tested in air to determine the heating and force output as a function of time. A single fan is used for cooling six of the seven motions; a small fan will be installed later for the other motion. For the test, four drives were energized with 150 V on the control and the fixed fields.

At the start of a run the output force was about 61 lb. As the motors warmed up, the output dropped fairly rapidly during the first few minutes and then levelled out at about 43 lb. The force was down to 50 lb at 2 min after start.

The maximum copper temperature after 45 or more minutes was about 170°C. This is within the safe rating of the class-H insulation used.

A box, 3 x 5 x 5 ft high, has been fabricated for testing the slave arm in an argon atmosphere. The box is equipped with a heat exchanger to cool the argon and has removable side panels which have glove ports.

A "motion exerciser" is being designed to apply six simultaneous motions to the master arm. This will be used to give typical, but accelerated, tests to the entire arm.

b. Low-inertia Servo Motors for Manipulators. The special 2-pole, 2-phase low inertia servo motor is being wound. The size of the wire was changed because of difficulty in obtaining the wire originally specified.

Studies are continuing on dc shell-type motors. So far, this type of design looks quite good.

2. General Heat Transfer

To improve the fundamental techniques by which liquid-metal-cooled reactors can be designed for better performance, the boiling and two-phase behavior of liquid metals in forced-convection flow, improved methods for predicting temperature in reactor coolant channels and for designing liquid-metal heat exchangers and steam generators, and heat-transfer and flow characteristics of highly dispersed mixtures of liquid and vapor are being investigated.

a. Boiling-sodium Experiment. To provide the understanding of liquid-vapor behavior that is essential for analyzing the safety of sodium-cooled fast reactors for large central-station power plants, as well as for the design and operation of Rankine-cycle power systems for space applications, the heat transfer and two-phase flow characteristics of boiling sodium are being investigated at temperatures up to 2100°F and pressures of approximately 8 atm. Among the variables to be investigated are boiling heat flux and temperature difference up to the occurrence of critical heat flux, boiling and adiabatic two-phase pressure losses, vapor volume fraction, boiling-stability parameters, and ultimately the transient behavior of some of the same quantities.

(i) Niobium-1% Zirconium Loop. The principal research tool consists of a Nb-1% Zr loop that is housed in a 10^{-7} - 10^{-8} Torr vacuum chamber for protection against oxidation at high temperature. The loop, which is constructed of 0.5-in. tubing, has a sodium flow of 25 ft/sec and a heat-flux capability from a thermal-radiation heater of 10^6 Btu/hr/ft².

Pressure-transducer and flowmeter calibrations have been completed at temperatures up to 1300°F. The thermal-radiation boiler has operated satisfactorily at 20% of rated power (heat flux $\sim 2 \times 10^5$ Btu/hr/ft²). Control of condenser heat loss appears to be adequate with use of the tantalum shutters. Some preliminary heat-loss calibrations will be run prior to low-temperature (1600°F) boiling runs.

b. Heat Transfer in Liquid-metal Heat Exchangers. To determine the accuracy of traditional engineering methods for predicting heat-transfer rates and temperature distributions in liquid-metal heat exchangers, and to find improved prediction methods, analyses based on the use of fundamentals are performed and are compared with results obtained using empirical heat-transfer coefficients and with experimental data. The analyses from fundamentals are also used to infer improved methods of prediction.

(i) Counterflow Double-pipe Heat Exchangers. Experimentation with the third heat-exchanger test section has been completed. A draft of a topical report describing both analytical and experimental portions of the research has been nearly completed. Further experimentation is not planned for the immediate future since the major objectives, i.e., verification of the analytical predictions, have been accomplished.

Further investigations based on applications of the analysis continue with emphasis on interpreting overall heat-transfer rates as predicted by the familiar effectiveness versus NTU type of engineering design procedures and on predicting detailed wall-temperature distributions. Figure 16 shows an example of heat-exchanger effectiveness versus NTU_{∞} for liquid-metal to liquid-metal heat exchangers. The subscript " ∞ " refers to the use of overall heat-transfer coefficients computed with fully developed tube- and annulus-side coefficients corresponding to the boundary condition of uniform heat flux. The parameter Ω_w represents the thermal resistance

Fig. 16. Effectiveness vs. NTU_{∞} for Liquid-metal Counter-flow Double-pipe Heat Exchangers

of the heat-exchanger wall relative to the total thermal resistance according to the usual additive thermal-resistance concept used with heat-exchanger design. Thus $\Omega_w = 0$ identifies no thermal resistance through the wall, whereas $\Omega_w = 1$ identifies heat transfer controlled completely by the wall resistance. The parameter H is the heat-capacity--flow rate ratio (annulus side divided by tube side); K represents the thermal resistance of the fluid in the annulus relative to the thermal resistance of the fluid in the tube. The curve labeled $\Omega_w = 1$ corresponds to the traditional ϵ versus NTU relationship as given, for example, in Fig. 2 of

the book by Kays and London.³⁶ The curves shown for $\Omega_w < 1$ illustrate differences between traditional design methods and predictions from a more exact analysis.

(ii) Nonsymmetrical-duct Heat Exchangers. Construction of the small heat-transfer loop for the purpose of performing experiments with sodium in model heat exchangers has been delayed because of priorities. The first test section, however, a double-pipe heat exchanger with adjustable eccentric placement of the inner tube, has been completed.

(iii) Liquid-metal-heated Steam Generators. Preparation of an interim report, summarizing (1) design procedures representative of current engineering practice for predicting the rate of steam production in liquid-metal-heated steam generators and (2) an initial exploratory analysis of "more exact" prediction methods, is nearly complete.

c. Heat Transfer in Liquid-metal-cooled Reactor Channels. To determine the accuracy of traditional engineering methods of predicting temperatures in liquid-metal-cooled reactor channels, and to find improved prediction methods, heat-transfer fundamentals are employed for analysis whose results are compared with those from using traditional methods. Although such studies of reactor channels and studies of heat exchangers involve different classes of boundary problems, both studies have the same motivation: the accuracy and convenience of traditional engineering methods

³⁶Kays, W. M., and London, A. L., Compact Heat Exchangers, The National Press, Palo Alto, California (1955) p. 32.

depend on heat-transfer coefficients being relatively insensitive to heat-flux distributions, duct shape, and distance from duct inlet, but these coefficients for liquid metals are much more sensitive to these variables than are nonmetallic fluids.

Previous analyses for predicting temperatures in liquid-metal-cooled ducts with nonuniform axial-heat-flux distributions have been studied for extension to transients. The results of this initial study show considerable promise with respect to being able to suggest a sufficiently simple "engineering-type" relationship to account for forced-convection heat transfer with liquid metals in ducts experiencing heat-flux transients.

Within the framework of "engineering-type" relationships, the sense of the results of the initial study can be briefly described as follows. Current engineering practice uses the familiar relation

$$q = h(t_{\text{surf}} - t_{\text{bulk}}) \quad (1)$$

for steady conditions, and also when the heat flux q is not uniform with respect to axial distance. Previous studies³⁷ under this research category have shown that Eq. (1) should be replaced by

$$q + R_1 \frac{dq}{dz} + \dots = h(t_{\text{surf}} - t_{\text{bulk}}) \quad (2)$$

when q is not uniform. In Eq. (2), z is a dimensionless axial position, h is the fully developed heat-transfer coefficient for uniform flux, and R_1 is a quantity which depends only on Reynolds and Prandtl numbers and geometrical shape, but not on the flux distribution. The "+ ..." represents higher-ordered terms which usually can be neglected. The initial study related to heat-flux transients has shown that Eq. (2) also applies when written as

$$q + R_1 \left(\frac{\partial q}{\partial z} + \frac{\partial q}{\partial \theta} \right) + \dots = h(t_{\text{surf}} - t_{\text{bulk}}), \quad (3)$$

where R_1 and h are the same as in Eq. (2) and θ is a dimensionless time.

Computer codes for the purpose of predicting the effects of power excursions usually (if not always) use Eq. (1) to account for forced-convection heat transfer during transients. Equation (3) offers a potentially more accurate and nearly as simple alternative. Investigations will continue with emphasis on determining the regions of applicability of Eq. (3).

³⁷Stein, R. P., "Liquid Metal Heat Transfer," in *Advances in Heat Transfer*, Vol. 3 (T. F. Irvine, Jr., and J. P. Hartnett, Eds) Academic Press, New York (1966) p. 133, Eq. (49).

3. Condensing Injector Studies

To determine the actual efficiencies of the condensing injector, a key component of a liquid-metal magnetohydrodynamic power-conversion cycle, so that the attractiveness of the cycle can be determined, loss mechanisms are being evaluated in experiments on test loops having various test fluids, such as sodium-potassium alloy, water, freon, and mercury.

A condensing injector having a large area-contraction ratio has been developed and is being tested with supersonic inlet vapor velocity. Entrance, profile, discharge vapor and liquid velocities, pressures densities, and mass ratios are being measured to evaluate performance.

The analysis of the condensing-injector data and analytical models of the flow processes in the condensing region are essentially complete. Final efforts in combining the various flow models and drafting a final report are underway.

A summary analysis of the data indicates two distinct flow patterns exist in the condensing-injector region. Initially, as the liquid and vapor jets are nozzled into the condensing section, the flow is well-defined and stratified, with the liquid jet in the central region. Near the end of the condensing region (4 in. long), the flow pattern is dispersed or homogeneous with apparent uniform density and no evidence of a central liquid core. A transition region between the two distinct patterns exists in which the flow proceeds from one in which a central core is still discernible as more dense than the outer vapor flow, but less dense than the initial liquid core, to a pattern of uniform density. This region is characterized by the complete disintegration of the liquid jet into the outer vapor flow, referred to as jet breakup. High shear rates at the liquid-vapor interface and some escape of the dissolved gas content of the liquid jet appear to cause jet breakup. No analytical or correlative means are available which predict the axial length of jet breakup. However, it is apparent from the data that a relation between the inlet vapor-liquid momentum ratio and jet-breakup length exists for this test injector, and can be used to determine the transition between analytical flow models.

Two analytical formulations are employed for the distinct flow patterns. These are joined at the point of jet breakup. The stratified condensing-flow model adequately predicts the axial pressure profile up to the point of jet breakup. This model shows that the rate of the vapor condensation determines the axial pressure profile to a far greater extent than either wall or interfacial shear. These condensing rates are described by a heat-transfer coefficient of the order of 10^5 Btu/hr/ft²/°F, based on the plane liquid-vapor interfacial surface area and bulk temperature difference.

A Carpenter-Colburn³⁸ type of relation accounting for relative velocity and friction effects indicates a coefficient less by an order of magnitude, with the remaining discrepancy attributed to a much greater effective condensing-surface area due to entrainment and interface wave structure. Also, effective heat-transfer coefficients for a similar steam-water direct condensing situation appear to be greater by an order of magnitude.³⁹

At the point of jet breakup, a homogeneous-flow model is employed that, to some degree of satisfaction, shows trends in the axial pressure profiles and maximum discharge pressures with various inlet conditions. In this region, the flow exhibits varying degrees of similarity with single-phase compressible flow and possible occurrence of shock phenomena. In cases of large liquid-vapor flow ratios at the inlet, the maximum discharge pressure is approached through a sharp pressure rise just upstream of the constant-area section. The assumption of negligible phase change in this region is satisfactory and enables calculation of pressure profiles and maximum discharge pressures in close agreement with experiment. In cases of low liquid-to-vapor flow ratios at the inlet, the transition from stratified to dispersed flow is spread out and complete breakup occurs farther upstream. The pressure profile approaching the maximum discharge pressure is much less steep, and it is evident from the data and analysis that the assumption of no phase change is inaccurate. Strong reliance on the experimental data is necessary to obtain agreement with axial pressure profiles and maximum discharge pressures for such cases. Correlation of the required phase change with inlet flow ratios may be of some use in these instances.

Finally, it is noted from the data that no discharge pressure in excess of the possible momentum pressure rise available at the point of jet breakup is obtained, so that the additional area convergence from the point of jet breakup to the minimum-area section is to no advantage in bringing the dispersed mixture to rest or increasing the stagnation pressure over what could be obtained by continuing a constant-area section at the point of jet breakup. Thus the jet-breakup length seems to be of some importance in determining the maximum practical area convergence.

4. Engineering Mechanics

a. Anisotropic Thermoelasticity. To develop refined analytical and experimental methods for determining thermally induced displacements

³⁸Carpenter, E. F., and Colburn, A. P., The Effect of Vapor Velocity on Condensation Inside Tubes, Proceedings of the General Discussion on Heat Transfer, Inst. Mech. Eng. (London), and ASME, 1951, pp. 20-26.

³⁹Levy, J., The Mixing of Vapor and Liquid Jets, Aerojet-General Corp. Report No. 1344 (Final), October 1957.

with the precision needed in the design of fast-reactor core components, a study is being made of the significance of thermal distortions caused by direction-dependent effects introduced by the intrinsic anisotropic properties of fuel-element materials and by the effective anisotropic properties that describe the way regular substructure patterns influence larger structural elements. Models are being developed to describe the overall deflection of fuel-element assemblies and entire core structures under both normal and abnormal operating conditions. Initial efforts on steady-state problems for rotationally symmetric, transversely isotropic materials will be extended later to include other forms of anisotropy, dynamic effects, stress concentration, and the effect of anisotropy on thermally induced failure.

The last of a series of sample steady-state anisotropic thermoelasticity problems intended to assess the numerical difficulties associated with the analytical study of anisotropic material behavior has been completed. Figure 17 illustrates the radial and axial variations in a typical dimensionless stress component for a specific anisotropic material and

Fig. 17. Radial and Axial Variation in a Typical Dimensionless Stress Component for a Specific Anisotropic Material and its Relation to the Corresponding Isotropic Solution

its relation to the corresponding isotropic solution. The stress shown is the circumferential component of the stress field caused by a discontinuous axial variation in internal heat-generation rate in a right circular cylinder being cooled at its outer surface by a temperature-independent boundary

Fig. 18. Axial Variation of One Component of the Transient Solution

conductance. For convenience of illustration, anisotropic-material constants corresponding to a very nearly isotropic material were used. Although this problem is in most respects similar to those which will arise in applying anisotropic models to describe the behavior of composite structures such as fuel-element assemblies, there remain several unstudied questions concerning methods required to determine equivalent elastic constants for these structures. Investigations relating to desirable methods of material description are currently being initiated.

As mentioned previously (see Progress Report for May 1967, ANL-7342, pp. 97-98), the study of certain selected transient (quasi-static) isotropic problems has been initiated to determine relationships between the time variation of the temperature field and the time required to reach peak thermal stress magnitudes. Figure 18 illustrates the axial variation of one component of the transient solution corresponding to the problem previously discussed--the transient behavior having been introduced by assum-

ing that at the initial time ($\tau = 0$) the heat-generation rate jumps from zero to some constant value. Such preliminary calculations indicate that the peak stress magnitudes are not significantly higher than the steady-state values. This differs markedly from the behavior of the stress field caused by step changes in surface temperature wherein the transient stresses are many times greater than the steady-state values and occur much sooner than the time of peak temperature.

b. Core Structural Dynamics. To provide understandings needed for precise design of reliable fast-reactor internals, under predictable operating conditions and fuel-structure life or limitations, the flow-induced displacements and motions of reactor core components and their effects are being studied with an integrated theoretical-experimental approach that will provide empirical information, theoretical correlation, and experimental verification. Among the features being investigated are prediction of mean-square response, energy-transfer mechanisms, pressure-drop relationships, system influence on stability of flow and fuel, and a flow-velocity-dependent relationship.

In the study of the parallel-flow-induced vibration of cylindrical rods, sufficient data have been obtained to allow interpretation of an observed phenomenon and preliminary verification of this interpretation. As discussed previously (see Progress Reports for May 1967, ANL-7342, pp. 96-97, and June 1967, ANL-7349, p. 105), the root-mean-square displacement of a rod, as obtained from experimental power spectral density curves, has been shown to increase monotonically with mean axial flow velocity according to the relation.

$$r_{\text{rms}} \approx A(x)u^{1.9} \quad (1)$$

over the range of flow velocities tested ($0 \leq u < 11$ ft/sec). However, from the experimentally obtained spectral density curves we observe that, while the mean-square displacement is increasing, the contribution to this mean-square value in the narrow band of frequencies centered at the first natural frequency appears to be approaching a limit. We might also point out that the experimentally determined spectral densities of the response become increasingly broad with increasing flow velocity (thus accounting for the monotonic increase of mean-square response).

To interpret these observations, consider the series representation of the displacement spectral density function as given by Eq. (2), p. 81, of Progress Report for March 1967, ANL-7317. Neglecting modal-frequency cross-coupling and the convection of the random pressure field, and further assuming the rod to respond only in its first mode, the spectral density of the response can be written as

$$\Phi_y(x, \omega) = |H(x, \omega)|^2 \Phi_p(\omega), \quad (2)$$

where

$\Phi_p(\omega) = C_{11}(\omega)$, spectral density of pressure field;

$|H(x, \omega)| = \frac{\alpha_1(x)}{|Z_1(\omega)|}$, magnitude of the modal harmonic response functions;

$$Z_1(\omega) = M_1[(\omega_1^2 - \omega^2) + jR_1\omega], \text{ modal impedance.}$$

Plots of $|H(x, \omega)|$ versus ω for a given value of x result in a series of curves corresponding to different values of damping (R_1). For $R_1 < 0.5$, these curves peak at the approximate value of ω equal to ω_1 (the first natural frequency), with the magnitude of the peak proportional to $1/R_1$.

Now consider the spectral density of the pressure field and assume the shape of the spectral density curves to remain the same, but the magnitude or energy to increase, with increasing mean flow velocity. With a pressure field fitting this description and considering the curves of $|H(x, \omega)|$ for varying R_1 , both in conjunction with Eq. (2), the earlier observations regarding the increasing mean-square displacement response and varying shape of the response spectral density curves with mean flow velocity can be explained if the amount of damping, R_1 , were to increase with mean flow velocity.

To obtain preliminary verification of this interpretation, a relatively rigid metal rod, of diameter and length equal to that used in vibration tests, was instrumented to measure dynamic pressure on the surface of the rod at the center and at the quarter-point. The rod was installed in the flow loop. Pressure-time histories were recorded and subsequently processed to give spectral density curves corresponding to the range of mean flow velocities tested. By means of the experimentally determined spectral density curves for the displacement response and

Fig. 19. First-mode Harmonic Response Function for Various Mean Axial Flow Velocities

pressure, the first-mode harmonic response function was constructed according to Eq. (2). Normalized to a value of 1.0 at a frequency of 2 cycles/sec, the results are given in Fig. 19. The reduction in peak magnitude with increasing flow velocity does indeed indicate an increase in damping with increasing flow velocity.

Conclusions to be derived from the work done to date include (1) the need to consider the entire system (that is, pump, piping, flow obstructions, etc.) in regard to its effect on the energy spectrum of the pressure field and (2) the futility of attempting to describe the

displacement response by a relationship that does not adequately account for the random pressure field as a forcing function. It is hypothesized, at this point in the program, that the future design of reactor cores to avoid detrimental core-component vibrations will involve design of the coolant system to include acoustic dampers, flow straighteners, and the like, to control better the energy spectrum of the pressure. In other words, system design will be emphasized rather than core-component design, which merely shifts resonant frequencies and varies damping, an effect which is also predominately felt only in the resonant-frequency bands.

D. Chemistry and Chemical Separations

1. Fluoride Volatility Processes

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

(i) Fission Product Fluoride Chemistry: Chemistry of Ruthenium Fluorides. Two experiments ($\text{BrF}_5\text{-F}_2\text{-1}$ and $\text{BrF}_5\text{-F}_2\text{-2}$) have been performed as part of a study of the distribution of ruthenium fluorides formed in the various steps of the fluid-bed fluoride volatility process. In each experiment, 2 to 5 g of ruthenium metal (tagged with Ru^{106}) in a nickel boat was contacted with fluorinating gas in a horizontal $2\frac{1}{2}$ -in.-dia, 16-in.-long reactor. Downstream from the reactor, a NaF trap at 400°C and a soda lime trap at 100°C were monitored for Ru^{106} content.

First, the ruthenium was contacted with 10 v/o BrF_5 in nitrogen with the reactor and gas-phase temperature at 250°C^{40} and with a flow rate of about 300 ml/min. Next, to simulate the reaction conditions used in current laboratory-scale process-development studies, the residue from the first step was fluorinated for 2 hr with 100% fluorine at a flow rate of about 150 ml/min and temperatures ranging from 300°C at the start to 550°C at the end. In each experiment, new charges of NaF and soda lime were placed in the traps after the BrF_5 step.

During the first 30 min of the BrF_5 step, the reaction rate was rapid, the rate constant⁴¹ $>0.012 \text{ min}^{-1}$, and about 70% of the ruthenium was reacted and volatilized from the reactor; the reaction rate then decreased sharply (rate constant of 0.001 min^{-1} , see Fig. 20). During the subsequent F_2 step, the reaction rate was also low, and less than 50% of the remaining ruthenium was fluorinated. These results suggest the formation of an unidentified product in the residue that is resistant to fluorination by both BrF_5 and F_2 .

⁴⁰It was reported in the Progress Report for June 1967, ANL-7349, p. 107, that ruthenium metal did not react with BrF_5 at 250°C . Possibly, the gas-phase temperature was not as high as 250°C in those earlier experiments.

⁴¹The reaction rate constant k' was determined by use of the equation $(1 - F)^{1/3} = 1 - k't$, where F is the fraction of solid reacted and t is the time in minutes.

Fig. 20. Rate Data for Two-step Reaction of BrF_5 and Fluorine with Ruthenium Metal. (Derived from Ru^{106} counts.)

After each step of experiment $\text{BrF}_5\text{-F}_2\text{-1}$, the NaF and soda lime charges were removed from the traps and surveyed. The ratios of the activity in the NaF trap to the activity in the soda lime trap were $>2000:1$ and $150:1$ for the BrF_5 and F_2 steps, respectively. These results indicate that the proportion of volatile ruthenium fluoride transferred through the NaF trap was at least ten times greater in the F_2 step than in the BrF_5 step, implying that the volatile ruthenium compounds are not the same in the two steps.

To investigate further the behavior of these apparently different ruthenium compounds, another experiment was performed to determine the effect of F_2 fluorination on the ruthenium fluoride adsorbed on or reacted with the NaF. After the BrF_5 step of experiment $\text{BrF}_5\text{-F}_2\text{-2}$, the NaF trap at 400°C and the soda lime trap at 100°C were subjected to a fluorine flow of about 150 ml/min for $1\frac{1}{4}$ hr. The ratio of Ru^{106} activities in the NaF and soda lime after this treatment was $200:1$, in good agreement with the results obtained in the F_2 step of experiment $\text{BrF}_5\text{-F}_2\text{-1}$.

b. Fluorination Chemistry and Procedures: Fluorination of $\text{UO}_2\text{-PuO}_2\text{-F.P. Pellets--2-in.-dia Fluid-bed Reactor}$. A 2-in.-dia fluid-bed reactor is being used for developmental studies to establish reaction conditions that minimize residual plutonium in the final alumina bed. Uranium dioxide-plutonium dioxide pellets containing nonradioactive fission product oxides are the feed material. In the process, the pellets are first oxidized to $\text{U}_3\text{O}_8\text{-PuO}_2$ 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 the subsequent recycle-fluorination step, plutonium is recovered as volatile PuF_6 by the reaction of PuF_4 with fluorine.

The plutonium concentration in the final bed from an alumina-reuse experiment (runs Purse-20, -21, and -22; see Progress Report for June 1967, ANL-7349, pp. 107-108) has been determined. This experiment simulated the processing of a fuel with a burnup of 30,000 MWdt, or three times the fission product content of fuel used in earlier experiments in this facility. The results, 0.008 w/o plutonium in samples from the final bed corresponding to 0.6% of the plutonium charged to the reactor, indicate that reuse of alumina to reduce the fraction of plutonium retained by the alumina is feasible for fuels of higher burnup.

Useful information about the rate of plutonium volatilization during the recycle-fluorination step has been obtained from the Purse series of exploratory runs. Results from runs Purse-4, -5, -9, -13, -17, -18, and -19 (see Progress Reports from April 1966, ANL-7204, p. 63, to May 1967, ANL-7342, pp. 99-100) were selected for study because these runs all met the following criteria: (1) the recycle-fluorination step began at 300°C, (2) the plutonium charge was 2.8 g, and (3) the residual plutonium in the final alumina bed was less than 3% of the plutonium charged. In these runs, fluorination was performed at 300°C for 3 hr, and then the temperature was increased in increments of 10°C/12 min until it reached 550°C.

Analyses of bed samples taken at hourly intervals during the recycle-fluorination step were averaged. These average plutonium levels were compared with the plutonium levels obtained after the BrF_5 step, and it was observed that:

(1) During the first hour of recycle-fluorination at 300°C (when the fluorine concentration was increased from 10 to 90 v/o), one-third of the plutonium reacted.

(2) During the second hour of recycle-fluorination at 300°C, an additional 12% of the plutonium reacted; during the third hour, 5% reacted.

Figure 21 shows the progress of fluorination from 3 to 8 hr, when the temperature was increased from 300 to 550°C, and during the last hour at 550°C. During the time period when the reactor temperature was increased from 350 to 400°C, an accelerated rate of plutonium fluorination accounted for reaction of 26% of the plutonium.

Fig. 21. Plutonium Reacted during Recycle-fluorination at 300 to 550°C Using Recycled Gas Containing 90 v/o F₂. 95% confidence intervals are shown for each time period. In run Purse-17, 50 v/o F₂ was used.

Data from thirteen Purse-series runs in which the plutonium charge ranged from 2.8 to 28 g were fitted to the diminishing-sphere model. Rate constants, which were calculated for each hour in the recycle-fluorination sequence, agreed within a factor of two or three for each hour of the various runs.

c. Neptunium Fluoride Chemistry.

Neptunium hexafluoride will possibly be removed from a gaseous stream in the fluoride volatility process by reaction with solid NaF. Recent work (see Progress Report for June 1967, ANL-7349, p. 108) showed that pure NpF₆ vapor reacts with powdered NaF at 150°C to form a violet solid that has been tentatively identified as 3NaF·NpF₅. Experimental work is in progress to determine whether it is possible to remove neptunium from the solid complex by fluorination.

Samples of the complex have been treated with fluorine in static systems at temperatures of 100 to 400°C and pressures of 100 to 1400 mm. In one series of experiments, the pressure at the end of a test was compared with the initial pressure, and in a second series the initial and final vapor densities of the gas phase were compared. An increase in either would indicate that NpF₆ had been formed according to the reaction

No evidence that the complex had been fluorinated to form NpF₆ was obtained in these experiments.

d. Recovery of Uranium and Plutonium from Low-enrichment Fuels: Engineering Development for UO₂-PuO₂ Fuel; Engineering-scale Alpha Facility. The major objectives of experiments under way in the engineering-scale alpha facility are to demonstrate handling of PuF₆ in engineering-scale equipment and to determine the feasibility volatility flowsheets.

Almost 600 g of plutonium has been processed in fluorination studies in the Engineering-scale Alpha Facility to date. Information derived in the course of some 15 months of operations can now be re-evaluated in terms of one of the primary objectives, namely, the ability to transport PuF₆ in engineering-scale equipment and to show satisfactory material balances.

Of the 574 g of plutonium charged, 557 g of plutonium has been accounted for, giving an overall material balance of 97%. This balance is considered to be very satisfactory in view of possible analytical and sampling errors for the large number of samples taken.

Plutonium was introduced into the fluorinator system in the following materials in the course of 15 experiments involving charges of 20 to 100 g of plutonium:

103 g as unirradiated 0.5 w/o $\text{PuO}_2\text{-UO}_2\text{-F.P.}$ pellets

410 g as PuF_4 powder

40 g as PuF_4 in the residue resulting from oxidation and BrF_5 fluorination of unirradiated 5 w/o $\text{PuO}_2\text{-UO}_2\text{-F.P.}$ pellets

21 g as a PuF_6 "spike" in a fluid-bed thermal decomposition experiment.

574 g Total

Of the 557 g finally accounted for, about 58 g of plutonium as PuF_4 was in alumina-bed samples removed from the fluorinator and in the final alumina bed. The remaining 499 g of plutonium was converted to PuF_6 during fluorination operations and was transported through the pilot-plant system of filters, valves, and associated piping to cold traps and subsequently to NaF sorption traps, or directly to NaF sorption traps. Included in this 499 g is 78 g of plutonium recovered in cleanup-fluorination operations from the upper regions of the fluorinator (including sintered metal filters), the cold traps, and product receivers. Little plutonium remained in the equipment after the cleanup step as determined by neutron surveys and by analysis of leachings of a sectioned hexafluoride receiver.

In summary, results have demonstrated that PuF_6 can be safely and quantitatively transported in engineering-scale equipment after the equipment has been thoroughly conditioned by fluorination. In future work, a more complete demonstration of PuF_6 handling is planned in a campaign of runs with larger quantities of plutonium, introduced into the fluorinator system in the form of PuF_4 .

e. Engineering Development of Fluid-bed Fluorination Procedures: Process Development Studies for Uranium Dioxide Fuels. Engineering-scale studies are being performed in a 3-in.-dia fluid-bed reactor facility to determine the effects of important process variables on the fluorination of UO_2 fuels with BrF_5 . The current program involves a series of eight statistically designed experiments to measure the effects of six independent variables on UF_6 production rate, BrF_5 utilization, off-gas composition, and uranium removal from the fluid bed; one of the runs was repeated to obtain an estimate of experimental error, making a total of nine runs that have been performed.

The final two runs, BrF₅-10 and -4, involved the oxidation of 2.2 and 4.4 kg of UO₂ to U₃O₈, using 19 v/o oxygen at 450°C. The uranium hexafluoride product was recovered using 11 v/o BrF₅ at 225°C and 20 v/o BrF₅ at 300°C in the two runs. Oxidation and fluorination proceeded smoothly in both runs.

Quantitative effects of the independent variables will be reported after all analytical results are obtained.

f. Small-scale Studies with Irradiated Fuels. Equipment used for earlier experiments with irradiated fuels (see Progress Report for April 1967, ANL-7329, p. 74), which was originally designed for the study of alloy fuels, has been disassembled and removed from the hot cell. A new improved reactor and its trapping systems, designed for the processing of oxide fuels, have been assembled outside the hot cell for shakedown testing with nonirradiated UO₂. It is planned to determine the distribution of short half-life fission products by reirradiating 2-yr-cooled Yankee UO₂ fuel and processing it in the hot cell after five days of cooling.

IV. NUCLEAR SAFETY

A. Fuel Meltdown Studies with TREAT

1. Modifications of High-level Facility for Plutonium-bearing Sample

Fast reactor safety experiments are being performed in the Transient Reactor Test Facility (TREAT) in order to study the behavior of fast reactor fuel and coolant under transient accident conditions. The goal is development of models to describe prefailure fuel movements, fuel failure, postfailure fuel movements and failure-propagation phenomena, coolant pressure pulses, coolant expulsion, secondary movements, and the degree of coherence in all these phases. Prior irradiation of fuel can play a large role in influencing the course of such accident conditions. Irradiation-induced factors include changes in physical properties of the cladding, buildup of fission gases both inside the cladding and within the fuel structure, changes in fuel structure, and changes in chemical composition of the fuel due to depletion and buildup of fission products.

Experimentation with plutonium-bearing fuel requires facilities having special capabilities for containment. A glovebox facility (see Progress Report for April 1967, ANL-7329, p. 93) is to be built in the Hot Laboratory, Building D-301, for experiments on low-level plutonium-bearing samples. A companion facility, to be realized by modifying caves 1 and 2 in D-301, is to be provided for operations with high-activity plutonium-bearing pins. Figure 22 shows the conceptual design for modification of the caves for plutonium service. Back of the caves is a controlled-access service and loading area. Each cave will have a steel liner, to provide alpha containment, and booted manipulators. The area between the modified caves is to become a shielded transfer area, with cave windows, booted manipulators, and an alpha steel containment shell as shown in the figure. Short-term storage and a variety of cave operations can be accommodated in the central area in addition to transfer operations, so that the "effective" cave work stations will actually be increased by the modifications.

Irradiation specimens can be introduced into the modified cave complex, given examinations, loaded into TREAT loops or capsules, and later returned, unloaded, and given post-transient inspections.

2. Mark-II Transparent TREAT Capsule

A Mark-II transparent TREAT capsule has been designed (see Fig. 23) to permit performing fast reactor safety transient experiments with fast reactor fuel samples too long (greater than about 45 cm) to put within the standard transparent capsules which fit in the reactor core viewing slots. The new capsule occupies the space of two standard TREAT

Fig. 22. Preliminary Design of D-301 Conversion for Fast Reactor Safety Experiments with Plutonium-bearing Fuel

Fig. 23. Mark-II TREAT Transparent Capsule

fuel elements and can be loaded into the core vertically. Fittings for the instrumentation leads and for the cover gas system are at the top of the capsule. The slot liner capsule has only one gas system sweeping through the unsealed inner subassembly into the outer, sealed slot liner subassembly. The Mark-II capsule consists of two sealed cans, one inside the other, with separate gas-cover-system circuits, so that independent double containment of samples can be provided to meet special experimental requirements. The configuration and dimensions are selected so that the Mark-II transparent capsule can use the same TREAT handling coffin and the same shipping casks as the Mark-II integral TREAT loop (see Progress Report for March 1967, ANL-7317, pp. 106-107).

Each subassembly of the Mark-II capsule has a 0.635-cm-thick window of high-purity fused silica, mounted in a massive stainless steel frame to provide secure sealing, protection of window against flexing of the capsule walls, and a heat sink to afford thermal protection to the silicone window gaskets. The steel-can walls provide the pressure containment for the capsule, and these walls are protected against meltdown products by an interior liner. The sample pin is shown in Fig. 23 to the left of the center plane of symmetry; a mirror is located to the right of the center plane to provide a partial rear view of the sample.

B. Materials Behavior and Energy Transfer

1. Pressure Generation due to Violent Meltdown: Transient Heat Transfer Studies in Liquid Sodium

Energy transfer and pressure generation, which may result from the contacting of fuel materials at high temperatures with the sodium coolant, are important factors in safety analyses of fuel failure for sodium-cooled fast reactors. The rates of heat transfer from bodies moving through molten sodium are currently under study. The experimental swinging-arm apparatus (see Progress Report for November 1966, ANL-7279, p. 88) has been used to obtain additional data on heat transfer from a 1/2-in.-dia tantalum sphere to liquid sodium. In this portion of the study, the initial sphere temperature was kept below the boiling point of sodium. These data were used to obtain a generalized correlation of heat transfer from a sphere to liquid sodium during forced convection.

It has been previously shown that the heat transfer expressions derived from an idealized frictionless-flow model for heat transfer from a sphere could not be used directly to correlate the experimental heat transfer rates from extremely hot spheres (see ANL-7279, p. 90). The forced-convection measurements described herein will provide a more realistic representation of the heat transfer rates than the highly idealized frictionless-flow model.

The experimental data are summarized in Table XXXII. For each experiment, the data were put into a generalized dimensionless form for correlation purposes. The heat transfer parameter of interest is the Nusselt number, defined by

$$\text{Nu} = hD/k, \quad (1)$$

where

h = heat transfer coefficient;

D = diameter of sphere;

k = thermal conductivity of sodium.

In these experiments Nu is primarily proportional to h , since D is 1/2 in. and k does not vary greatly. The properties of sodium were evaluated at

TABLE XXXII. Heat Transfer Data from 1/2-in.-dia Tantalum Sphere to Liquid Sodium

Run No.	Sphere Velocity (ft/sec)	Average Temperature of Sphere Wall (°C)	Sodium Temperature (°C)	Average Heat Flux [cal/(cm ²)(sec)]	Nusselt Number	Reynolds Number
1	11.30	478	300	250	10.50	124,000
2	10.30	452	300	221	10.75	110,500
3	10.30	434	300	202	11.05	108,800
4	9.45	430	300	208	11.75	99,500
5	5.12	381	300	77	9.98	51,400
6	6.50	386	300	118	9.92	65,500
7	11.30	350	200	251	11.55	101,400
8	9.10	376	200	246	9.73	85,200
9	7.10	357	200	190	8.40	65,200
10	5.55	336	200	178	8.95	49,200
11	5.30	352	200	200	9.10	47,500
12	7.80	356	200	197	8.98	71,500
13	9.80	384	200	218	8.33	88,000
14	10.88	401	200	270	9.49	104,200
15	10.88	403	200	263	9.12	104,600
16	10.88	390	200	256	9.72	103,300
17	11.16	404	200	259	8.94	101,400
18	3.84	372	200	191	7.70	35,600
19	4.98	416	200	239	8.18	49,600
20	5.98	397	200	238	8.59	58,000
21	4.98	392	200	230	8.40	48,000
22	6.40	446	300	183	9.28	68,500
23	6.23	446	300	161	8.15	66,600
24	6.40	444	300	170	8.72	68,500
25	6.45	446	300	178	9.20	69,800
26	9.90	800	450	554	13.70	142,100
27	10.90	750	450	531	14.58	144,000
28	11.35	700	450	373	12.50	152,500
29	6.38	800	300	723	11.90	84,500
30	5.45	800	300	561	9.22	72,600

the film temperature, $1/2(T_w + T_{Na})$, where T_w is the temperature of the sphere wall. The Reynolds number, defined by

$$Re = \rho U_\infty D / \mu, \quad (2)$$

where

ρ = density of sodium;

U_∞ = velocity of the sphere;

μ = viscosity of sodium,

was calculated for each experiment. Another parameter of importance in liquid-metal heat transfer phenomena is the Prandtl number, defined by

$$Pr = C_p \mu / k, \quad (3)$$

where C_p is the specific heat.

The heat transfer data obtained in this study are compared in Fig. 24 with an empirical expression used by Vliet and Leppert⁴² to correlate heat transfer from spheres to water:

$$Nu(Pr)^{-0.5} = 2.7 + 0.12 (Re)^{0.66}, \quad (4)$$

with a viscosity correction recommended by Vliet and Leppert omitted. The data are also compared to the theoretical expressions for heat transfer from a sphere developed by Hsu⁴³ and Sideman.⁴⁴ Both Hsu and Sideman

developed an equation for heat transfer from a sphere to a flowing fluid by assuming frictionless flow past the sphere. Both expressions are identical in form, differing only by a constant factor. Hsu's expression is

$$Nu(Pr)^{-0.5} = 0.921 (Re)^{0.5}, \quad (5)$$

and Sideman's expression is

$$Nu(Pr)^{-0.5} = 1.13 (Re)^{0.5}. \quad (6)$$

Fig. 24. Comparison of Sodium Heat-Transfer Data with Equations of Other Workers

⁴²Vliet, G. C., and Leppert, G., Forced Convection Heat Transfer from an Isothermal Sphere to Water, Trans. of the ASME (Series C), Journal of Heat Transfer **83**, 163 (May 1961).

⁴³Hsu, C. J., Heat Transfer to Liquid Metals Flowing Past Spheres and Elliptical-rod Bundles, BNL-7664-R (1964).

⁴⁴Sideman, S., Ind. Eng. Chem. **58**(2), 54 (1966).

Figure 24 shows that neither the empirical expression due to Vliet and Leppert nor the frictionless flow expressions of Hsu and Sideman correlate the data well. The experimental error of the data is considered to be at most $\pm 20\%$. Even when this is taken into account, all of the comparison expressions fall above the experimental values.

Fig. 25. Correlation of Data for Sodium Heat Transfer

Theoretically, the value of Nu due to pure conduction is 2 when $Re = 0$. The data were plotted (see Fig. 25) as $(Nu - 2)(Pr)^{-0.5}$ versus Re to obtain the convective effect. The data were correlated by the expression

$$Nu = 2 + 0.39 (Re Pr)^{0.5} \quad (7)$$

The exponential power, 0.5, of the product of Re and Pr corresponds to that given by the results of potential (frictionless-flow) theory.

The experimental data which were presented previously (see Progress Report for November 1966, ANL-7279, p. 88) have been compared

to heat transfer rates computed from Eq. (7). It should be noted that Eq. (7) is based on the experimental data of Table XXXII, obtained with sphere-wall temperatures below the boiling point of sodium, whereas the experimental data from ANL-7279 were obtained with sphere-wall temperatures above the boiling point of sodium. Therefore, a comparison of Eq. (7) with these experimental data constitutes an extrapolation to higher temperatures than conditions under which the correlation was obtained. In ANL-7279, it was found that the experimental data could be correlated well by Sideman's relation, Eq. (6), multiplied by an arbitrary constant and by assuming that the liquid sodium adjacent to the sphere wall assumes the temperature of the sphere wall. In an effort to obtain a more meaningful correlation, the data obtained at temperatures above the boiling point of sodium were compared to Eq. (7), as may be seen in Fig. 26. In addition, heat transfer rates computed from the correlation of Sideman, Eq. (6), are also plotted (solid line). For both Eqs. (6) and (7), the temperature of the sodium at the wall was assumed to be the wall temperature (superheated). The comparison shows that the data are bracketed by the two curves. Apparently the correlation for forced-convection heat transfer for sphere-wall temperatures lower than the boiling point of sodium is not valid at higher temperatures. The use of a fractional multiplier applied to Sideman's relation as reported previously [see ANL-7279, p. 91, Eq. (3)] results in the best correlation of the data.

Fig. 26

Heat Flux vs Surface Temperature for 1/2-in.-dia Sphere Moving through Liquid Sodium. (Sphere Velocity = 10 ft/sec; Sodium Temperature = 300°C.)

The results of a high-speed movie of a 1/2-in.-dia tantalum sphere leaving a pool of sodium indicate that the front of the sphere cools faster than the back of the sphere during forced-convection heat transfer to sodium. Temperature-sensitive infrared film was used for this experiment. The sphere was filmed shortly after its exit from the sodium. The sphere temperature at entry into the sodium was approximately 1500°C, well above the boiling point of sodium. The infrared film showed that the front portion of the sphere was noticeably cooler than the rear portion after its travel through the sodium pool. The actual temperature difference between the front and the rear of the sphere is not known; no attempt was made to obtain quantitative measurements in this experiment. Visually, the difference did not appear to be large. This experimental observation supports the previously described theory (ANL-7279, p. 90), since that theory predicts that the major portion of the heat transfer from the sphere occurs on the front side.

The swinging-arm heat transfer apparatus is being modified so that experiments can be performed in sodium at temperatures near its saturation point. Essentially, this requires enclosing the sodium so that the large amounts of vapor evolved from sodium above ~300°C are contained.

Errata

In the Progress Report for August 1966, ANL-7249, p. 86, the first full paragraph should begin:

The diffusion coefficient may be expressed by

$$D = 4.125 \times 10^{-5} T_f^{1.75}/P.$$

On p. 87 of the same report, second paragraph, the fourth sentence should read: The calculated average rates at 1 and 10 atm for $r = 0.018$ cm are, respectively, 3.5×10^{-5} and 3.9×10^{-5} moles H_2O /sec.

C. Coolant Dynamics

To determine the extent to which rapid changes of coolant density or flow patterns adversely influence heat transfer and reactivity, and thus the safety of sodium-cooled fast reactors, analyses and experiments are being used to develop analytical models that realistically describe both the coolant behavior and its mechanical interaction with the surrounding fuel structure during abnormal reactor conditions.

1. Critical Flow

Because two-phase critical flow could occur during an accidental power surge in a sodium-cooled reactor and could lead progressively to detrimental voidage of coolant channels, shock phenomena, and pressure buildup, measurements are being made of critical flow rates, pressure profiles, qualities, void distributions, and sonic velocities in simulated fuel channels so that methods can be developed for predicting critical flow velocities, metastability, superheat, velocity ratios, and flow structure in high-velocity two-phase flow.

To investigate the basic processes controlling the two-phase critical flow of sodium, studies are being carried out using steam-water as the working media.

Previous theoretical developments⁴⁵⁻⁴⁷ for two-phase critical flow have assumed a one-dimensional steady-state approach; however, the validity of this assumption has not been demonstrated experimentally. Andeen and Griffith⁴⁸ investigated these approximations for subcritical two-phase air-water flows. They measured the momentum flux by an impulse method and also measured the average void fraction in the test section by means of quick-closing valves. Their results show conclusively that, for low-velocity flows (small axial pressure gradients), the one-dimensional steady-state flux which is calculated from the measured void fraction and quality is considerably less than the measured flux.

The authors present three reasons for this deviation: (1) unsteady fluctuations such as slugging, (2) two-dimensional velocity profiles, and (3) turbulent fluctuations. All three reasons cause the real momentum to be greater than the one-dimensional case.

Although the data and conclusions of Andeen and Griffith are quite valid at low velocities, their experimental apparatus contains inherent

⁴⁵Fauske, H. K., Contribution to the Theory of Two-phase, One-component Critical Flow, ANL-6633 (1963).

⁴⁶Levy, S., Prediction of Two-phase Critical Flow Rate, Trans. ASME, J. Heat Transfer 87-C, 53 (1965).

⁴⁷Moody, F. J., Maximum Flow Rate of a Single-component, Two-phase Mixture, Trans. ASME, J. Heat Transfer 87-C, 134 (1965).

⁴⁸Andeen, G. B., and Griffith, P., The Momentum Flux in Two-phase Flow, MIT Dept. Mech. Eng. Report 4547-38 (1965).

errors whose magnitude increases with increasing velocity. Since the measured momentum flux is that of the exit plane, the exit void fraction is required to calculate the momentum efflux predicted under the one-dimensional, steady-state approximations. The quick-closing-valve method of measuring void fractions gives an average void fraction over a given axial length. This average void fraction is representative of the exit void fraction only when the momentum pressure drop is small compared to frictional pressure loss. (The method is exact when the momentum pressure drop is zero, but this is impossible to obtain in an adiabatic compressible system.) When the momentum pressure drop is the predominant term in the momentum equation, as it is in critical flow, the quick-closing-valve technique is no longer a reasonably accurate method of obtaining the exit void fraction. Hence, in flows characterized by large axial pressure gradients, one must determine the exit void fraction before the one-dimensional approximation is advocated or condemned.

The total pressure drop ΔP_T is the summation of the momentum and frictional pressure drops when the head drop is negligible:

$$\Delta P_T = \Delta P_M + \Delta P_f, \quad (1)$$

where ΔP_M is momentum pressure drop and ΔP_f is frictional pressure drop, all in lb_f/ft^2 . If one-dimensional, steady-state conditions are assumed, the momentum pressure drop for small-quality ($x_E < 0.02$) flow in a constant-area duct is

$$\Delta P_M = \frac{G^2}{\rho_\ell g_c} \left(\frac{1}{1 - \alpha_2} - \frac{1}{1 - \alpha_1} \right), \quad (2)$$

where G is flow rate in $\text{lb}_m/\text{sec}/\text{ft}^2$, g_c is the gravitational constant, ρ_ℓ is liquid density in lb_m/ft^3 , and α is the void fraction. Rose and Griffith⁴⁹ have shown that frictional pressure losses may be estimated by

$$\Delta P_f = f \frac{L}{D} \frac{\rho V_m^2}{2g_c}, \quad (3)$$

where L is length and D is diameter (both in ft), V_m is the homogeneous velocity:

$$V_m = \frac{G}{\rho_\ell (1 - \alpha) + \alpha \rho_g}, \quad (4)$$

in ft/sec , ρ_g is vapor density in lb_m/ft^3 , and ρ is the two-phase density

$$\rho = \alpha \rho_g + (1 - \alpha) \rho_\ell \quad (5)$$

in lb_m/ft^3 .

⁴⁹Rose, S. C., and Griffith, P., Flow Properties of Bubbly Mixtures, ASME Preprint 65-HT-58 (1965).

The total pressure drop, based on the one-dimensional, steady-state assumption, may be computed if the void fractions are known. This calculated pressure drop can then be compared with the actual pressure decrease to determine the validity of the assumed conditions.

To investigate this experimentally, Test Section R7 (see Progress Report for June 1967, ANL-7349, p. 121) was used. The void fractions were measured by the technique of gamma-ray attenuation at the exit plane and five positions upstream. The one-dimensional, steady-state pressure drop was then calculated by starting at the farthest point upstream and summing the momentum and frictional losses from point to point. The frictional losses were based on an average void fraction between the two points and an average friction factor of $f = 0.020$. The friction factor is actually a function of the Reynolds number, but since the friction losses are a very small part of the pressure decrease, the average value given above was used as a representative figure.

Figure 27 shows the agreement between the calculated and measured profiles for two runs of Test Section R7. Table XXXIII lists the measured

Fig. 27

Comparison between the Experimental Pressure Profile and that Predicted by the Void-fraction Data

and calculated total pressure drops between position U5 and the exit plane for all the runs of Test Section R7 where this type of void-fraction data

TABLE XXXIII. Comparison between the Measured and Calculated Total Pressure Drop

Run	ΔP_T		% Deviation
	Measured (psi)	Calculated (psi)	
1	13.4	11.0	18
2	9.9	9.5	4
3	8.3	7.3	12
4	8.1	8.3	2
5	8.6	8.4	2
6	7.2	7.5	4
7	8.4	7.0	17
8	18.0	12.5	31
9	27.0	29.0	7

was obtained. As is demonstrated by the graphs and the table, the agreement is quite good, indicating that a one-dimensional, steady-state approach reasonably approximates the physical situation for high-velocity flows.

2. Coolant Dynamics

Sodium superheat and expulsion are being investigated so that analytical models can be developed for predicting coolant behavior during various stages of reactor accidents.

a. Superheat. To determine the mechanism and amount of liquid superheat that can build up before nucleate boiling begins in sodium under various simulated reactor conditions, measurements are being made of the independent and combined influence of pressure, heat flux, dissolved gas, oxide content, surface characteristics, system pressure-temperature history, and ionizing radiation.

The installation of the thermal-radiation-heater is complete. However, although the radiation heater is producing heat fluxes in excess of 40 W/cm^2 , considerable end losses were observed. Additional radiation shields are being installed to reduce these losses. The leaks in the bell-jar housing have been corrected so that sufficiently high vacuums have been obtained. This thermal-radiation heater is expected to extend significantly the previously attained heat-flux range.

Also, the data obtained from the test runs show significant effects of system pressure, heat flux, and the pressure-temperature history upon the incipient boiling superheats in sodium.

b. Expulsion. To enable formulation of an analytical model that describes how reactor coolant behaves during transient heating in a power surge or loss of coolant flow, measurements are being made of liquid superheats, vapor formation and distribution, and pressure propagation during coolant expulsion from a simulated reactor channel. Nonflowing tests to determine the effects of heating rates, subcooling, inertial head, dissolved gas, channel geometry, and surface finish are providing the foundation for experiments with rapidly heated flowing sodium.

Work is continuing on the installation of the programmed electron-bombardment heater. Upon completion of the installation, the

vendor plans to participate in final system check and analysis. Following this, final assembly of the test section and associated equipment will commence. Assembly of the sodium-expulsion apparatus is also in progress and much of the pertinent instrumentation is being assembled, checked out, and mounted in racks.

3. Convective Instability

Experiments will yield fundamental data on transient free-convection heat transfer in narrow-channel liquid-metal systems so that heat transfer effects in fast-reactor startup or loss-of-cooling accidents can be analyzed. Effects of Prandtl and Rayleigh parameters on system heat transfer are being determined analytically so that a readily usable nondimensional correlation can be derived.

Initial installation of the test section resulted in shattering one quartz plate which makes up a face of the section. It was discovered that a 7-mil anomaly existed in the flange which attaches to the containing tank. The whole apparatus was dismantled and the flange milled flat to 1 mil. This necessitated grinding the quartz end pieces to meet the flange width and machining the riser to meet the same requirements.

4. Electron-bombardment-heater Tests

Electron-bombardment-type heaters, which simulate reactor-heated fuel-pin clusters of various configurations, are being developed, fabricated, and tested in a large evacuated vessel. Work with single-pin heaters at transient heat fluxes up to 5×10^6 Btu/ft²/hr will lead to multipin heaters to simulate parallel channels.

The fabrication of the first EBH test section is essentially complete. The three banks of powerstats have been connected into the high-voltage power supply. The filament transformer to be used for heating the cathode has been received, checked out, and installed in the system. Also, much of the instrumentation for the experiment has been checked out and mounted in racks.

5. Thermophysical Properties of Sodium

The final draft of ANL-7323, which recommends "best values" of several properties of the liquid and vapor phases of sodium, was completed. Liquid-phase compressibility and sonic-velocity data have been incorporated. A contemplated section dealing with property values at the critical point has not been added because of time limitations.

6. Core Component Dynamics

To determine the manner in which coolant expulsion and core structural deformation interact so that the variation of pressure with time during a power transient can be interpreted, analytical models are being developed to explain and predict the results of out-of-pile experiments. The investigations of coolant-structure coupling will proceed from simple configurations to the more complex situations inside a reactor and to studies of wave propagation, impulse loading of shells, and the energy-absorbing capabilities of core components.

In this program, coolant-structure systems are assumed to be subjected to disturbances (energy insertions) which result in the generation of pressure transients. An objective of the program is to determine how the flexibility of the structure influences the characteristics and propagation of pressure transients.

An initial phase of the study of core component dynamics has been completed; its primary objective, to demonstrate the phenomenon of coolant structure coupling, has been met (see Progress Report for February 1967, ANL-7308, pp. 84-85). The coolant-structure system considered consisted of a coolant-filled rectangular channel formed by two parallel flat plates. The system was subjected to a rectangular power burst and the influence of structural flexibility on system response was investigated. This preliminary study is now being extended to include geometrical shapes more typical of fast reactors. In particular, cylindrical shells are being emphasized.

The instantaneous volume change, as afforded by the deformation of the bounding structure, is of primary importance, influencing the characteristics and propagation of the pressure transient. For the simple coolant-structure system chosen for preliminary study, the structural dynamic relationships relating this change in volume could be written immediately. However, in considering shell geometries, the structural-dynamic equations become sufficiently complicated that this aspect of the study warrants consideration a priori to the inclusion of the coupled coolant effects.

A phenomenon, related to the structural deformation of a shell under an impulsive pressure loading, was noted in a report by Proctor.⁵⁰ Proctor reports that experiments have shown that the near-rupture deformation pattern on a right-circular cylinder, for a pentolite explosion in water and air, never exceeds a length of approximately four radii. Specifically, illustrations showed the deformation length to be approximately four radii for water (incompressible fluid) and less than two radii for air (compressible fluid). This observation suggests the possible existence of a

⁵⁰Proctor, J. F., Explosion Containment for Nuclear Reactor Vessels, Nuclear Safety 7(4), 459-468.

"characteristic deformation length" which is dependent on the compressibility of the fluid and the shape or distribution of the pressure loading, but independent of the magnitude of the loading. The existence of such a characteristic length, if it can be proven, would be of great help in the future study of coolant-structure coupling.

This aspect of the structural dynamic modeling is being studied by approximating the cylindrical shell by an infinite beam on an elastic foundation. Various assumed pressure distributions are being studied as to the deformation pattern that results. A deformation pattern with nodes approximately four radii apart, for a pressure distribution representing a point charge in an incompressible fluid, would lend support to the hypothesis of a characteristic deformation length.

D. TREAT Operations

The program of 100 transient tests of prototype fuel rods for the Power Burst Facility Reactor which was started during May was completed on July 17, 1967.

The final operational test scheduled for the out-of-pile portion of the large TREAT sodium loop was a simulation of a reactor transient test. All control valves and other equipment functioned satisfactorily during this test. Further system tests will be made with the test section in place in the reactor prior to running the first transient experiment with fuel in the test section.

A calibration transient was performed on a vibratory-compacted, mixed uranium-plutonium oxide sample for the General Electric Company.

E. Plutonium Volatility Safety

1. Environmental Contamination Control for Volatility Processing

a. Effluent Gas Treatment: Chemistry of Tellurium Fluorides.
Studies have been continued on the removal of tellurium hexafluoride from off-gases generated in the fluoride volatility process. Earlier work (see Progress Report for April 1967, ANL-7329, p. 99) showed that adequate removal of TeF_6 from TeF_6 -air mixtures can be achieved by sorption on a bed of activated alumina.

Recent tests are related to the possibility that fluorine, in addition to TeF_6 , will be present in the off-gas stream. Although fluorine is to be removed by caustic-iodide scrubbing or trapping in activated alumina upstream from the TeF_6 -removal bed, it is possible that a small fraction of the fluorine will not be trapped. The presence of fluorine in small concentrations might lower the efficiency for removal of TeF_6 , or fluorine might displace sorbed TeF_6 on the bed.

A test was performed at 25°C with an air mixture containing 500 ppm TeF_6 and 500 ppm fluorine, flowing at 20 ft/min through a 2-in.-high activated-alumina bed. Comparison of the results of this test with those of another experiment in which conditions were comparable except that fluorine was absent revealed that fluorine slightly affects the TeF_6 -removal efficiency, but that adequate TeF_6 removal is still easily obtained in the presence of fluorine.

In a second study, a 2-in.-high bed of activated alumina was loaded at 13.8% of hypothetical capacity with radiotagged TeF_6 . A mixture of air and 1000 ppm fluorine was first passed through that bed at a flow rate of 20 ft/min and then through a second bed of activated alumina. Buildup of activity on the second bed indicated that fluorine displaces TeF_6 from an activated alumina surface. Therefore, if activated alumina is to be used for removing both fluorine and tellurium hexafluoride from off-gas streams, studies of the effects of gas velocity, fluorine gas concentration, and bed temperature on TeF_6 retention should be performed to determine what bed depths will be adequate in pilot-scale operations.

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ANL-7290

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