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Argonne National Laboratory

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

August 1967

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REACTOR DEVELOPMENT PROGRAM

Highlights of Project Activities for August 1967

EBWR Plutonium Recycle Experiment

After unloading the natural and enriched uranium fuel assemblies, criticality could not be achieved with the remaining 36 irradiated plutonium fuel assemblies. Three additional plutonium fuel assemblies, of which one was irradiated, were required for this purpose.

The experimentally determined multiplication constant for the unrodded core containing 39 fuel assemblies was found to be 1.0008 ± 0.0001 . The computed multiplication constant for a loading of 40 assemblies of which 36 were irradiated was calculated by a four-group two-dimensional diffusion code, 20-Grand, to be 0.99.

The plutonium fuel assemblies were unloaded from the reactor, and the experimental portion of the program was terminated.

EBR-II

The spare fuel-handling gripper for the main core was installed, replacing equipment that was suspected to be damaged. The new unit has been tested successfully and is now available for operation. Preliminary inspection of the removed gripper indicated no apparent damage to the orientation blade or bellows.

A core and blanket randomization experiment to check the model that was developed to explain the decreasing power coefficient in EBR-II was carried out, with no significant recovery of reactivity being observed. Further study of the problem is continuing.

Neutron radiography of the capsules from experimental subassembly XO11 (the subassembly which had experienced a fission gas release) was started at the TREAT reactor.

Fabrication of EBR-II driver fuel continued routinely. Eighteen subassemblies were fabricated this month including one which was produced in the cold line.

ZPR-3

Work continued on Assembly 50, the third in a series of the simple plutonium-fueled assemblies being studied in ZPR-3. The corrected critical mass for this assembly is 225.6 kg of $\text{Pu}^{239} + \text{Pu}^{241}$. Central fission ratio,

perturbation, and reactivity measurements were made. Data resulting from measurements with Assembly 48A which had been made in previous months are given.

ZPPR

Construction of the ZPPR facility continued. Placing of the sand fill over the vault workroom was started, and 90% of the corrugated steel shell has been placed around the backup containment of the cell.

A test of the strength of the concrete in the cell ring beam was made and it was found to be satisfactory.

AARR

All AEC requests for further information on the Preliminary Safety Analysis Report have been answered. Review of the design by the Advisory Committee for Reactor Safeguards is believed to be imminent. Recommendations for the seismic-design criteria have been sent to the AEC.

The test section for the transient heat-transfer tests is ready for installation in the test loop.

Either the "dimple" or "spline" type of support for the beam-tube jackets is satisfactory from the standpoint of heat transfer and temperature.

A dynamic analysis indicates that the potential for significant pump-induced pressure resonances exists in the primary coolant system of AARR.

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Reactor Technology
(TID-4500)
AEC Research and
Development Report

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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

August 1967

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Issued September 28, 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:

February 1967	ANL-7308
March 1967	ANL-7317
April 1967	ANL-7329
May 1967	ANL-7342
June 1967	ANL-7349
July 1967	ANL-7357

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I. PLUTONIUM UTILIZATION--EBWR

A. Operations

The final power operations of EBWR were completed in June 1967, and the final critical runs were completed in July 1967.

Activities at EBWR have since been aimed at placing the plant in a lay-up state and gathering limited information for the postirradiation program of vessel surveillance.

1. Lay-up Activities

The lay-up of the reboiler facility has been completed, and activities during the month have centered around lay-up of the turbine plant. Following removal of all fuel and samples, the vessel was drained, flushed, and placed in dry lay-up. Some of the turbine plant's piping, pumping, and tankage were drained. Turbine lay-up will be continued through September.

Movement of recycle fuel from the Bldg. 331 storage pit to the Bldg. 310 storage pool was begun, and at the month's end about half the cask movements were completed.

2. Postirradiation Vessel Surveillance

In-vessel corrosion samples were removed and transferred to the hot labs for evaluation. One measurement of vessel corrosion taken from the steam area showed a corrosion of 0.020 in., some eight times the predicted rate. Measurements in other areas are not included in the present lay-up schedule because of subjection to excessive radiation levels and the additional work involved in removing the vessel shock shield.

B. Reactor Physics

After 51 natural uranium fuel assemblies and 60 enriched uranium fuel assemblies were unloaded, it was found that criticality could not be achieved with the remaining 36 plutonium fuel assemblies which had an average exposure of approximately 3000 MWd/tonne. Criticality was reached with a loading of 39 assemblies, eight control rods fully withdrawn to 48 in., a corner control rod withdrawn to 28.5 in., and with the reactor water unborated. Of the three added fuel assemblies one had received a nominal exposure of 1300 MWd/tonne during the first burnup step the other two assemblies were fresh.

The multiplication constant for the unrodded assembly was experimentally found to be 1.0008 ± 0.0001 from period measurements based on moving a partly inserted rod. However, a four-group computation using

the two-dimensional diffusion code, 20-Grand¹ was made for a 40-assembly loading (36 assemblies with 3000 MWd/tonne exposure and 4 fresh assemblies) which yielded a multiplication constant $k = 0.99$.

¹Tobias, M. L., and Fowler, T. B., The 20-Grand Program for the Numerical Solution of Few-group Neutron Diffusion Equations in Two Dimensions, ORNL-3200 (1962).

II. LIQUID-METAL FAST BREEDER REACTORS

A. EBR-II

1. Operations

The main core gripper with which difficulty had been experienced (see Progress Report for July 1967, ANL-7357, p. 3) was removed from the primary sodium tank. Preliminary inspection showed no apparent damage to the orientation blade or either bellows. The details of this work are reported in II.A.6.a.

A specially designed tool was used to check upper adapter alignment of selected subassemblies in the reactor. Misorientation of the subassembly U-1605 upper adapter was found. This subassembly in core position 13-F-6 was the one on which the gripper was manually rotated with jaws open to check for damage to the orientation blade.

After installation of the spare gripper lower assembly and re-assembly of the entire mechanism, normal fuel-handling sequences were satisfactory. Subassembly B-371 was then inserted in position 6-F-4 to fill the vacancy left by removal of B-341. Randomizing of subassemblies continued as programmed.

Reactor startup occurred on August 17. Control rods were calibrated, and zero-power rod drops were performed. Power-coefficient measurements were taken to 45 MW, with no significant change noted in the magnitude or characteristics of the power-coefficient curve over those exhibited previously. The reactor was shut down on August 18.

Further plans for operations and testing with the present core loading are being formalized.

a. Subassembly Transfers. Driver subassembly B-341 was transferred to the Fuel Cycle Facility (FCF) for examination. Experimental subassembly XO11 was sent to FCF for disassembly and examination of the capsules to try to determine which capsule was the cause of the previously reported fission gas releases. Experimental subassembly X900 was placed in the storage basket as part of a test for determining the effect of washing and reusing Type 304 stainless steel tubing in the form of stressed capsules.

b. Operational Support and Maintenance

(i) Spare Parts. The spare lower assembly was installed on the main fuel-handling gripper. The portion of the lower assembly that was removed has been sent to the ANL Illinois Central Shops for salvage of parts in fabricating a new spare lower gripper assembly.

Purchase requisitions have been issued for the procurement of the materials for fabrication of a spare subassembly holddown.

Delivery of the spare Fuel Element Rupture Detector (FERD) loop EM pump has been delayed from July to September 1967 because of fabrication difficulties.

Purchase orders for the following spare parts have been placed with the vendor:

- (a) primary M-G generator rotor and exciter;
- (b) secondary M-G generator rotor;
- (c) Variacs for the primary auxiliary EM pump and a disconnect switch;
- (d) gear motors for the large and small rotating plugs;
- (e) gear motor for the main core fuel-handling gripper-jaw drive;
- (f) starter for the thimble cooling gas boosters;
- (g) packing and seals for the dampers in the shield-cooling system.

(ii) Maintenance. Inspection of the Variacs removed from the primary auxiliary EM pump system indicated they were not suitable for reuse. Spare Variacs were installed and new spare Variacs were ordered.

The Fuel Unloading Machine (FUM) gripper was removed, cleaned, and the sensing rod was repaired. This rod had been damaged during removal of subassembly B-341.

c. Operator Training. Rewriting of the Power Plant, Electrical, and Coolant training packages continued. A new training writeup for the Steam system is being prepared.

Three requalification exams were administered and passed successfully. Training sessions were conducted for the operating crews to explain possible operating problems connected with the recent reduction in power coefficient.

d. Operating Manual. Review and revision of the facility Operating Manual continued. Operative Manual Revision Bulletin No. 54 "Argon Cooling System--(FUM)" containing revised procedures for operation of the system was issued. Revision Bulletin No. 55 "Miscellaneous Photo's" and Revision Bulletin No. 56 "Restricted Operation--Fuel Handling System" were also issued this month.

SOP No. 76 "Power Coefficient" which includes special procedures and limits for reactor operation during the power coefficient investigation for Runs 25 and 26 was issued. This document also includes instructions to the operators on action to be taken if any of a number of unusual operating characteristics should arise.

2. FCF Operations

a. Hot-line Operation. Table I summarizes FCF production activities for the month. Seven melt refining runs, involving approximately 70 kg, were made with unirradiated fuel. The Vertical Assembly Dis-assembly fuel machine was used to fabricate six of the subassemblies.

TABLE I. Production Summary

	8/1/67 through 8/31/67	Total This Year
1. Subassemblies received:		
Core, Control, Safety (for processing)	0	40
Other (XO11, B-341)	2	119
2. Subassemblies dismantled (for processing)	2	47
3. Subassemblies dismantled (for examination, etc.) (experimental subassemblies not included)	2	33
4. Subassemblies fabricated (includes those made with unirradiated fuel and in cold line)	18	124
5. Subassemblies transferred to reactor	0	71
a. Subassemblies stored in L&S vault	5	15
6. Elements decanned:		
From irradiated subassemblies	100	4,396
Rejects	78	772
Other	-	0
Total Decanned	178	5,168
<u>Melt Refining</u>		
	<u>Irradiated Fuel</u>	<u>Recycle Material</u> <u>New Fuel</u>
7. No. of runs	2	10 7
8. Average pour yield, %	94.5	93.8 97.0
1967 to date (Total--140)	34	59 47
<u>Processing</u>		
	8/1/67 through 8/31/67	Total This Year
9. Injection-casting runs (total number)	15	129
10. Elements processed:		
Accepted	1541	11,250
Rejected	184	1,464
11. Elements welded:	1278	11,125
Rewelded:	1	13
12. Elements leak tested:		
Accepted	1490	10,630
Rejected	8	174
13. Elements bonded (including recycle)	1530	12,941
14. Elements bond tested:		
Accepted	1287	10,957
Rejected	47	657
15. Elements to surveillance	0	407
<u>Waste Shipments</u>		
16. Cans to burial ground	12	80
17. Oxide and glass scrap to ICPP	3	48

Subassembly X011 was dismantled and the experimental capsules visually inspected. No apparent capsule failure was observed. Eight of the nineteen capsules were transferred to TREAT for neutron radiography; the remaining eleven capsules will be transferred to TREAT for radiography.

Examination of subassembly B-341 was completed. The subassembly is being stored in the Air Cell pending a decision as to final disposition. Contact with Idaho Nuclear Corporation was made in regard to disposition of iron-contaminated fuel. Three of the four subassemblies containing iron-contaminated fuel were dismantled, and plans are underway to ship all the iron contaminated fuel to ICPP.

b. Cold-line Operation. The following is the status of the various components that are being used in the cold line for production of unirradiated fuel:

Three alloy-preparation runs were completed with enriched uranium. The first two were made in zirconia-coated, graphite crucibles, and the third in a zirconia crucible. The second run was not completed since the molten metal would not pour into the ingot mold. Inspection of the crucible indicated a large amount of oxide or carbide on top of the uranium melt. It is suspected that this was due to an air leak into the furnace, probably through the freeze seal.

Two runs were made in the injection-casting furnace using depleted uranium in one and depleted uranium-5% fissium alloy in the other. Neither was completely successful, but after changing some of the conditions and operating procedures, two successful runs were made with the enriched ingots. The chemical analysis of a pin from the first run showed 1000 ppm carbon. It is assumed that the carbon was dissolved in the metal during alloy preparation in a coated graphite crucible. The second run was made from the ingot cast from a melt in a zirconia crucible, the carbon in a pin from this ingot was 150 ppm. Analysis of pins from both ingots showed that the total uranium and enrichment were within specifications.

Pins from the second casting batch have been loaded into jackets, welded, leak tested, bonded and bond tested, and assembled into a subassembly.

About half of the components for the cold-line assembly machine have been received. The balance of the parts should be received and assembled during the next month. The existing assembly machine, which was designed primarily for experimental and blanket subassemblies, is being used meanwhile.

A contract is being negotiated with an outside vendor to supply 600 kg of fissium (95% U-52% U^{235}) in ingot form to be used as the source material for the cold line. Preliminary indications are that 100 kg may be produced in September 1967 and the remaining 500 kg within calendar year 1967.

c. Maintenance and Repair. All electromechanical manipulators and the cranes are in operation, the second time since the start of hot processing in 1964 that this has occurred.

The program for computer control of maintenance activities is being written for all auxiliary equipment and will be put into effect in the next month.

d. Operator Training. On-the-job training continued for technicians who have been given new work assignments. Six technicians have been assigned to the "cold-line" operations.

e. Operating Manuals and Reports. In the past month, 5 new sections and 1 revised section have been completed and submitted for review and acceptance. Six sections, revisions, and parts of sections have been issued as accepted portions of the FCF Operating Manual.

The criticality rules and procedure for operation of the cold-line within the ITF building were submitted to the Laboratory Director's Office for approval. Tentative and limited approval to operate was received.

f. Product Analysis

(i) Chemical Analyses. The number of fuel-production analyses are tabulated below. All values fell within the normal production range.

Analyzed for	Production Fuel		Range
	Number	Average Value	
U (total and isotopic)	23	95.01 w/o	94.53-95.82
U^{235}		52.23 w/o	52.07-52.30
Pu (total and isotopic)	2		
Enriched Stock (U^{235})	6	93.17 w/o	93.14-93.20
Mo	31	2.46 w/o	2.11-2.66
Ru	26	1.90 w/o	1.78-2.03
Pd	26	0.175 w/o	0.12-0.22
Rh	34	0.295 w/o	0.274-0.363
Zr	30	0.092 w/o	0.054-0.136
Nb	13	0.020 w/o	0.016-0.025
Fe	32	221 ppm	127-570
Al	22	266 ppm	64-787
Si	29	207 ppm	80-380
C	9	202 ppm	146-233
Total	283		

The optical and electronic system of the atomic absorption spectrophotometer has been checked out by the manufacturer's service engineer. Calibration and study of conditions required for analyses of "cold" fuel constituents are underway.

(ii) Postirradiation Analysis of EBR-II Fuel. No spent driver-fuel subassemblies were returned from the EBR-II to the FCF during this report period; therefore, no product analysis effort was conducted.

The ICPP is continuing its corrosion test of five EBR-II-irradiated fuel elements (see ANL-7357, p. 57), which are currently stored under water in the ICPP basin.

A one-inch piece of irradiated EBR-II fuel pin was sent to the ICPP for their use to obtain data on fission product leaching. It is planned to carry out these measurements in September.

3. Outside Fuel Procurement

a. Procurement of Mark-IA-IB Fuel

(i) Fuel Elements. Two visits were made to facilities of the fuel-element fabricator, the Aerojet-General Corporation, San Ramon, California, to evaluate progress and discuss contractual matters. An ANL representative also accompanied Aerojet personnel on a visit to the facilities of the tubing manufacturer, the J. Bishop Company, Malvern, Pennsylvania.

(a) Facility Modifications. Aerojet has substantially completed modification of its facilities for element manufacture. The alloying and casting hood has been installed, and work tables are being set.

(b) Equipment. The inert-atmosphere glovebox and the alloy-preparation are expected in October and in early November. Fabrication of the injection-casting furnace and the centrifugal bonding machine will be initiated shortly. The sodium extruder is ready for testing. The atomic absorption spectrophotometer for alloy analysis is being set up. Fabrication of equipment for wire wrapping, leak testing, and welding is under way. Design of equipment for eddy-current inspection and pin processing is well along.

(c) Materials. Orders have been placed for all important process materials. The delivery of tubing material is such that the first samples should be run in November. The restrainer manufacturer has submitted samples. Wire for wrapping will be procured in slightly above finished length, and will be cut to length and annealed in bundles.

(ii) Assembly Hardware. Bids have been received from potential fabricators of the core, fueled inner blanket, control, and safety-rod preassemblies, and are being evaluated.

4. Reactor Analysis and Testing

a. Digital-data Collection. The intercom system interconnecting the EBR-II control room, data-acquisition room, and the computer room has been completed. The major components in the data-acquisition system have been completed. The amplifiers (6 channels) are now operating and have been completely checked out. A few wiring changes, made in interconnecting signal cables, have improved the signal-to-noise ratio.

The check of the 100-point data scanner revealed a design error in the tape format, and a field modification will be made.

b. Nuclear Analysis

→(i) Power Coefficient after Attempted Core and Blanket Randomization. A preliminary review of data from power-coefficient measurements made during Run 25 indicated an apparent decrease in power coefficient between the beginning and end of the run. One possible explanation for this decrease concerns movements of the fuel and blanket subassemblies, within the tolerances permitted by the grid plate and buttons, in such a way that the core at zero power tends each time to start from a more expanded position. Since the core apparently achieves a tightly compressed and reproducible configuration at full power, the expanded zero-power configuration could conceivably lead to a reduced reactivity decrement at power. A change in effective core radius of just a few mils could account for the observed variations.

To test this hypothesis and to determine the effect of positional uncertainties on reactivity, each core and blanket subassembly were to be disturbed mechanically in such a way that they would tend to return to their configuration when originally inserted into the reactor. In the mechanical procedure every seventh subassembly in the core and blanket was disturbed with the holddown-gripper systems. As a subassembly is contacted by the gripper, all six adjacent subassemblies are spread slightly outward by the holddown ring. Therefore, by gripping every other subassembly in every third row, it was possible to make contact with all core and blanket subassemblies in the reactor. This operation was performed after the power-coefficient measurements on July 22 and 23. A special experiment intended to evaluate the effects of core and blanket randomization was performed on August 17 and 18.

No significant recovery in reactivity decrement was observed. The results of the three power-coefficient measurements conducted on July 22 and 23 and August 18 are shown in Fig. 1. In this figure, power coefficients have not been corrected for expansion effects of the control-rod bank, since all runs were made at the same banked position. All data agree within the limits of experimental accuracy. The reproducibility is particularly good for the descending portion of the curve, for which previous measurements indicated a significant degree of drift and uncertainty. The results of the experiment neither prove or disprove the existence of a contribution to the power coefficient due to changes in the pattern of subassembly packing.

Fig. 1. Power-coefficient Curves before and after Subassembly Jiggling

It is still considered possible that the arrangement of the subassemblies within their fabrication tolerances can affect the character of the power-coefficient curve. The particular mechanical motions used to disturb any ordered pattern were very slight and may not have provided the desired effect.

c. Thermal and Hydraulic Studies

(i) Reorificing of Stainless Steel Subassemblies in Rows 7 and 8.

The results of calculations of the temperature distribution in the 7th-9th rows for the case with reduced flow in the 8th row stainless steel subassemblies showed that reducing the flow in the 8th row alone would not be effective in eliminating the reverse bowing in the stainless steel reflector. The HECTIC-II program has been modified further in order to permit the use of different ambient temperatures and ambient temperature gradients on opposite sides of a subassembly. Calculations have been started to determine the temperature distribution in the 7th-9th rows with reduced flows in both the 7th and 8th row subassemblies. For the calculations a flow of 3.3 gpm has been assumed for the 7th row subassemblies and of 1.35 gpm for the 8th row, with the expectation that outlet temperatures on the flats of the 6th-9th rows will be 870-880°F.

(ii) Calculational Techniques. The combined heat transfer program which takes as input the radial fission-rate distribution and the number of effective subassemblies in each typical core position (see Progress Report for July 1967, ANL-7357, p. 24) has been completed and checked. It provides a complete axial and radial temperature distribution for the minimum symmetry section (one-twelfth) of the EBR-II core.

(iii) HFEF Calculations. The problem of how long experimental high-burnup oxide and carbide fuels must "cool" prior to HFEF examination was investigated. The results of calculations were presented in the HFEF Project Prospectus.

d. Safety

(i) Power-coefficient Variation during Runs. A question has been raised relative to a change in the power coefficient which might have occurred during a run, specifically, during Runs 24 and 25. A very careful survey has been made of all available data on the power coefficient for Runs 14 through 25, in which every startup to 45 MW was analyzed. Because it is not required that the reactivity decrement be measured for any startups during a run, other than the initial startup, the only data which can be compared are those for which a good zero-power measurement was obtained using a control rod which had been specifically calibrated for that range of control rod needed to raise the power and under similar banked-rod conditions. Such measurements were the exception rather than the rule unless a power coefficient has been specifically called for, such as at the beginning of a run.

Table II presents the results of these analyses for Runs 25, 24, 22, and 16. Only the overall reactivity decrements to 45 MW expressed in inhours lost are tabulated. Table II contains data on all the startups

regardless of the startup condition, and those corrected data which have been obtained under calibrated, controlled conditions, and thus may be intercompared, are enclosed in boxes.

TABLE II Reactivity Decrement between 0 and 45 MW (lh)

	4/26	4/28	4/28	5/17	5/19	5/22	7/5	7/7	7/10	7/14	7/18	7/19	7/23	8/18
Run 25														
Raw Data ^a	42.3	38.4	38.0	37.4	39.1	33.3	39.6	32.5	34.8	35.9	25.9	21.5	32.7	32.0
Corrections														
Initial Power					+0.8	0.0	+1.0	0.0	0.0	+1.00	1.0	0.0	0.0	+1.00
Temperature	+2.0	0	0	+1.0	+0.0	0.0	2.0	-1.0	0.0	-2.00	0.0	+2.00	0.4	-0.8
Burnup	-6.0	0	0	-2.2	-1.1	0.0	-3.0	-2.0	0.0	0.0	0.0	0.0	-0.8	-0.6
Rod Bank ^b	0.0	0	0	0.0		+1.0	+1.0	+2.1	+2.1	+4.7	+4.7	+4.7	+4.7	+4.7
Corrected Data	38.3	<u>38.4</u> ^c	<u>38.0</u>	36.2	<u>38.8</u>	34.3	<u>40.6</u>	31.6	36.9	34.9	31.6	28.2	<u>37.0</u>	<u>36.3</u>
MWd	0	40	60	250	306	450	960	975	1096	1270	1425	1481	1543	1552
Rod Bank	11.0	11.0	11.0	11.0	11.0	11.4	11.4	12.0	12.0	13.00	13.00	13.00	13.00	13.00
Run 24														
Raw Data		64.3	64.0	56.5		69.7	72.2		77.0	77.8	73.0			
Corrections														
Initial Power		0	0			0.0	0.0		0.0		0.0			
Temperature		0	0			0.0	0.0		0.0		0.0			
Burnup		0	0			0.0	0.0		0.0		0.0			
Rod Bank		2.4	2.4	8.1		2.4	0.9		0.4	0.4	2.4			
Corrected Data		<u>66.7</u>	<u>66.4</u>	<u>64.6</u>		<u>72.1</u>	<u>73.1</u>		<u>77.4</u>	<u>78.2</u>	75.4			
MWd		0	500	555		0	220		0	350	695			
Rod Bank		12.0	12.0	14.0		12.0	11.4		11.2	11.2	12.0			

^aRaw data give the difference in control-rod position as determined from "book" value times the period calibration correction.

^bRod-bank correction is normalized to the 11.0-in. rod-bank position.

^cCorrected data enclosed in boxes are reactivity decrement obtained under controlled conditions in which a proper zero-power datum point was obtained using a calibrated control rod.

(a) Raw Data. During the initial startup program of EBR-II, a detailed curve for control-rod worth was established. The data were smoothed and the shape of the curve was least-square fitted to a calculated reference curve. Subsequently, whenever the total control-rod worth is established it is inserted into the computer program and a tabulated reactivity worth versus rod insertion table obtained. This value is called the "book" value. When precise reactivity data are required, the control rod is recalibrated over the range of rod insertion to be used, by measuring the asymptotic period corresponding to a definite increment for rod insertion and applying a correction factor to the "book" value. Such measurements usually are in the 5-9-in. rod-insertion range, for which the calibrated control-rod curve is nearly linear. The raw data give therefore the "book" reactivity difference associated with the calibrated rod increment required to go from zero power to 45 MW times the period correction factor obtained for that rod under the banked rod condition specified.

(b) Corrections. To these raw data, four corrections are to be applied as follows:

Initial Power--A correction is made for the initial critical condition for initial reactor power. Because of temperature-stability conditions due to heat losses, pump power, decay heat, etc., it is normal for the reactor to be leveled around 500 kW for final checkout prior to raising power. As a result, some initial critical conditions are at 400-500 kW instead of 50 kW or less. The power coefficient in this low range, 0-0.5 MW, is assumed to be 2 lh/MW, a number which appears to be reasonable from extrapolation of the initial slope of the reactivity decrement curve. All data for 400-500 kW are corrected by this power coefficient to a zero-power level.

Criticals taken at 100 kW or less are usually subject to wider fluctuations in bulk sodium temperature because of the need to use auxiliary heaters and the difficulties associated with secondary sodium convection heat losses.

Temperature--Bulk sodium temperature values are established by averaging the outputs of 5 thermocouples in the pump-inlet circuit and high-pressure plenum. The slow time response of the thermocouples and the inherent noise and temperature variations cause considerable difficulty in determining the bulk sodium temperature. Under conditions in which the reactor temperature has been very carefully stabilized at low power, the uncertainty is of the order of $\pm 0.5^\circ\text{F}$. Under conditions of power operation or in the stabilizing of the reactor at a high power level, the uncertainty is of the order of $\pm 1.0^\circ\text{F}$. The balancing of secondary sodium flow with power level to maintain a constant bulk sodium temperature requires considerable time to obtain the precision necessary for accurate reactivity determination. In Table II a correction has been made to normalize all bulk sodium temperatures to the reference 700°F .

Burnup--During the increase in power, the reactor is using fuel and hence losing reactivity at the rate of 0.15 lh/MWd. This value was measured during Run 25. Measurements in previous runs have indicated a value of 0.14 to 0.16 lh/MWd. When a routine power-coefficient measurement is made, the time involved varies from 4 to 8 hr at an average power level of about 22.5 MW. Hence, the burnup correction is of the order of 0.5 to 1.0 lh. During a rapid restart, the time to reach power is of the order of 2 hr and no correction is applied. However, as is the case of Run 25, because of the time required for detailed measurements, corrections as large as 6 lh were required. The failure to consider such corrections during the initial startup resulted in reporting erroneously high values for the reactivity decrement. The corrections made in Table II were obtained by carefully establishing the integrated power over the time of the measurement from hot critical to 45 MW, using the factor of 0.15 lh/MWd. A new instrument has been installed in the reactor panel which now electronically integrates the power and displays the integrated power in MWd.

Rod-bank Effects--It is usually necessary, depending on the length of the power run, to load the reactor with excess reactivity between 200 and 350 Ih. Normal operation requires that this reactivity be compensated by banking all control rods but one at some power level, usually between 11.5-12.5 in., and controlling one rod at a level usually between 6-9 in. As burnup proceeds, the controlling rod is raised until it approaches the 9-in. region, and then the bank position is increased by some definite increment that returns the controlling rod to about 6 in. Depending on the length of the run, the bank position is changed one to three times during a run. The initial power coefficient is measured with the rods banked in the initial running condition. As the power is raised, the temperatures of the banked control-rod shafts increase proportionately with power. This increase results from hotter sodium coolant flowing around the shafts. Such an increase causes the fuel section of the rods to be lowered relative to the core, thereby removing reactivity from the system. The reactor structure rests on the bottom of the reactor vessel, while the control rods are suspended from the vessel cover, and are free to expand and contract independently of the core. Since the position indicators are on the upper shafts, where the temperature is constant, this change is not recorded on the position indicator. The amount of reactivity lost is given by $\Delta l \left(\frac{\partial \rho}{\partial l} \right)$ where Δl is the increase in length of the rod and $\left(\frac{\partial \rho}{\partial l} \right)$ is the change in reactivity as a function of a change in position at the banked rod elevation. This loss of reactivity requires the controlling rod to be inserted farther, and is an integral part of the reactivity decrement required to go to power. The value of $\frac{\partial \rho}{\partial l}$ is a function of control-rod position, being a maximum at the center and small at 14 in. The value is obtained from the calibrated-control-rod curves.

As a run proceeds, the position of the control-rod bank increases from around 11.5 to 13.0 in., and the reactivity loss due to the effect of rod elongation decreases; hence, a smaller overall power coefficient is observed at the end of the run.

The expansion of the control-rod shaft is calculated from the temperatures of the sodium and, hence, the stainless steel shaft in the region of the core, upper axial reflector, the upper plenum, and the penetration through the reactor vessel cover. Because the flow through the control rod is varied with rod insertion, this effect should also be included. The total expansion of the control-rod shaft at 45 MW is calculated to be approximately 0.12 in.

Another small correction included in the calculation is based on the relative motion of core fuel with respect to control-rod fuel (without shaft expansion). This effect originates from a significant temperature difference between the fuel in normal and control-rod subassemblies. Hence, the core tends to grow upward with respect to control-rod fuel

as the power increases. Corrections for such effects, although smaller than those for control-rod expansion, should be considered. Efforts to evaluate such effects in greater detail are still in progress.

Calculations of reactivity effects associated with shaft expansion and relative growth between core fuel and control rod fuel have been carried out. The results indicate clearly that the combined

TABLE III. Reactivity Associated with
Control-rod-bank Corrections,
Run 25, at 45 MWt

Rod Position (in.)	Odd Rod (lh)	Even Rod (lh)
11.0	1.01	1.10
11.5	0.89	0.99
12.0	0.80	0.88
12.5	0.69	0.76
13.0	0.56	0.62
13.5	0.42	0.46
14.0	0.26	0.28

effects of shaft expansion and relative core growth constitute a significant portion of the power coefficient and depend very strongly on the actual banked rod position. The amount of change in terms of reactivity due to this relative growth depends on the differential rod worth at the position of the rod. The reactivity changes associated with each bank rod position in going from 0 to 45 MW are summarized in Table III for single rods, one even and one odd.

Hence, for the hypothetical situation in which the banked rod position is changed for 10 rods (5 odd and 5 even) from 11.0 to 14.0 in., the 45-MW reactivity defect measured at 11.0 in. would be larger than that at 14.0 in. by $5(1.01 - 0.26) + 5(1.10 - 0.28) = 7.85$ lh.

With this approach, the different rod bank positions for the various runs have been normalized to the reactivity decrement expected for an 11-in. rod bank, the correction has been incorporated in Table II. The expansion effect is directly proportional to power level at a given rod-bank position (for a given flow) and hence, reactivity corrections at intermediate power levels can be evaluated.

There are two major uncertainties in calculating this effect. First, the rod calibration in the region from 12 to 14 in. has never been measured with sufficient detail to obtain accurate differential measurements of the rod worth. Secondly, the temperature of the control-rod shafts must be inferred from calculation of the temperature of the sodium coolant surrounding the shafts. The accuracy of the expansion calculation is estimated to be within the limits of $\pm 20\%$. The differential reactivity data in the 12 to 14-in. region will be measured during the next startup.

Corrected Normalized Data--The values for reactivity decrement which are enclosed in boxes in Table II are those which have been

measured under calibrated conditions. The other values are included to show the range of value which might be expected under noncontrolled conditions. It should be emphasized that in all cases after restart the control rods returned to the same position at 45 MW as they had just prior to the shutdown or scram.

Conclusions--From the tabulated data there does not appear to be any significant weakening of the power coefficient with burnup. For example, in Run 25 from 0 to 960 MWd the decrement appears constant. The greatest difference occurs after the rods are rebanked at 13 in., likewise for Run 24. From 0 to 500 MWd the reactivity decrement is constant, but between 500 and 555 MWd, after the rods are rebanked to 14 in., there appears to be about a 2-lh loss. Neither Run 22 nor Run 16 shows any tendency for a weakening of the total reactivity decrement with burnup. Data have been accumulated for other runs, but calibrated data for a range of different integrated powers is very sparse and what has been presented in the tables are considered to be the most pertinent.

5. Development of Methods of Nuclear Analysis

a. Physics Analysis of the EBR-II Power Coefficient. In the study of the power coefficient for (88 subassembly) Run 25, a series of two-dimensional transport calculations were performed to investigate some separate components. The computation concerned only dimensional and density changes of the materials in the reactor with temperature. The reactivity feedbacks resulting from the mechanical movement of the core, "bowing effect," and the Doppler effect have not been included in this phase of the study. A brief description of the results of the computations appears in Table IV.

TABLE IV. Calculated Values of Reactivity Change for Various Conditions of EBR-II Run 25

Problem No.	Problem Description	Reactivity Change			Comment
		$-\Delta k/k_{eff}$	Inhour	-lh/MW	
1	Uniform Core Temperature: 700°F	-	-	-	Basic configuration at zero power
2	45-MW Operation	0.002276	97.6	2.18	-
	a. Fuel Temperature: 960°F	0.000573	24.6	0.54	Fuel expansion
	b. Steel	0.000780	33.5	0.70	Structural radial core expansion
	c. Sodium--Core	0.000426	18.3	0.41	Sodium-density change
	--Core and Upper Reflector	0.000866	37.2	0.83	-
	--Overall	0.000932	40.0	0.89	Core blanket and reflector
3	Uniform Core Temperature: 900°F	0.005356	230	-	Isothermal change at zero power

In order to estimate the effect of loading changes in EBR-II, temperature effects were computed for a reactor identical to that used in Run 25 except that the stainless steel assemblies in Rows 7 and 8 were replaced with depleted uranium blanket assemblies. This loading is considered as being representative of EBR-II Runs 20 through 24. The results indicate that the density and dimensional-dependent reactivity effects do not vary significantly between the two runs.

The computed axial-fuel-expansion coefficient has been found to be in good agreement with the preliminary values obtained from the two-power-level, variable-coolant-flow experiment. Essentially, with coolant flow proportioned directly to power, the thermal conditions of the bulk sodium, subassembly stainless steel, and all other components except for the fuel remain constant. The difference in the reactivity as measured by control rods is then that due to the change in the fuel temperature and thermal-axial expansion.

The experiment on Run 25 yielded a 8.0-lh decrease in reactivity in changing the operating conditions from 41.5 MWt and 100% flow to 22.5 MWt and 54% flow. Using the computed prompt fuel-temperature coefficient of $-0.095 \text{ lh}/^{\circ}\text{F}$, a fuel-temperature change of $\sim 60^{\circ}\text{F}$ gives a 6-lh decrease in reactivity. For Runs 20 through 24 the depleted uranium blanket in Rows 7 and 8 would contribute less than 2% to the above value.

The computed isothermal coefficient of $-1.15 \text{ lh}/^{\circ}\text{F}$ ($230 \text{ lh}/200^{\circ}\text{F}$, see Table IV) is in agreement with experimental value of $-1.04 \text{ lh}/^{\circ}\text{F}$.

This agreement between the experimental and the computed values has not allowed the further analysis of the components of the power coefficient to proceed with a higher degree of confidence.

A preliminary attempt was made to separate the computed linear component from the experimental power-coefficient curve. The linear component was taken to be comprised mainly of the fuel-expansion coefficient and the overall sodium-expansion coefficient, which adds to $-1.43 \text{ lh}/\text{MW}$. To include the structural density effect in the axial direction only would increase the coefficient to about $-1.5 \text{ lh}/\text{MW}$. The difference between this computed linear effect and the experimental curve then yields the component of the power coefficient resulting mainly from the radial motion of the core and blanket subassemblies--including the bowing effects. The results of separating the linear component from the total measured reactivity response with reactor power for Run 24, and Run 25 are presented in Fig. 2.

The general shape of the curves is to be compared with "bowing" models to describe the radial deflection of subassemblies at the various power levels. The model describing the "collective" deflection of core and blanket subassemblies for Run 24 was used for the initial comparison.

The given radial deflection over the length of each subassembly row was statistically weighted to obtain an effective radial displacement. Figure 3 is a plot of the reactivity change resulting from this combined radial movement as a function of the power level. The computed values of the "bowing effect" appear to follow the general shape of the Run 24 curve in Fig. 2, but the magnitudes differ appreciably.

Fig. 2. Reactivity due to Bowing vs. Reactor Power for Run 24 and 25 without Including -1.5 Ih/MW for the Linear Component. (Not corrected for control-rod-expansion effect.)

Fig. 3. Calculated Reactivity Effect due to Bowing vs. Reactor Power for Run 24

Since this is the initial attempt at developing a model which is consistent with temperature distributions in the subassemblies, the computations appear to indicate that Rows 7 and 8 undergo an inward movement of about 2 to 8 mils over the power range and that this motion is the major source of the positive reactivity feedback.

The model describing the collective motion of the subassemblies for Run 25 was found to be inconsistent with the expected reactivity feedback. The model is now in the process of being refined in the direction of minimizing the exaggerated outward motion of the Row-8 stainless steel subassemblies.

The effort to test and aid in establishing a consistent "bowing" model using reactivity diagnostics will necessarily include a continual refinement of the separate components of the power coefficient.

6. Reactor System Testing, Surveillance, and Evaluation

a. Fuel-handling System. As reported (see Progress Report for July 1967, ANL-7357, p. 3) subassembly B-341 could only be released with difficulty. From the results of tests, it was inferred that the core fuel-handling gripper was damaged and would require removal from the primary sodium tank.

Following the removal of the upper gripper-drive assembly and the shroud tube, a special pulling fixture was used to withdraw the gripper into a seal pipe fixture having an argon atmosphere. The fixture was then isolated from the primary tank and the gripper allowed to cool before transfer to a storage hole. The radiation field at the outside of the fixture at the gripper end of the drive was 400 R/hr. Because of the high radiation level of the assembly special procedures were used to inspect the gripper jaws and orientation blade. The gripper was suspended over a mirror mounted on the reactor building operating floor at 45° to the horizontal. By using an optical apparatus from a distance of approximately 45 ft, a closeup vertical view of the bottom of the gripper was obtained. Examination showed that no pieces had been lost from the gripper and that there is no major damage to the gripper orientation blade.

The upper gripper shaft was disassembled to permit inspection of the bellows assemblies. Both bellows were in good condition, with no indication of leakage or binding. The lower, highly radioactive 5-ft portion of the gripper shaft was cut off, and the radioactive gripper assembly was then cleaned; it will be disassembled and inspected at the TAN hot shops. The upper shaft assembly was returned to Argonne, Illinois for use in fabrication of another spare unit.

Preparations for installing the replacement core gripper were completed. The gripper gland standpipe on the small rotating plug was removed, cleaned, and replaced. The spare gripper unit was assembled, checked out, and installed in the removal pipe fixture. The assembly was put in place over the gripper standpipe with a special gate valve sealing the bottom of the removal fixture and isolating the primary tank. The entire gripper assembly was then purged by evacuation and backfilling several times with argon. When the purge was completed, the valve was opened and the unit was slowly lowered into the primary tank. The removal pipe was disconnected, and the packing gland reassembled and replaced. The upper drive was then installed along with the superstructure and shroud-tube assembly. All electrical cables were reconnected and the checkout of the unit was started. The push-and-pull force switches were checked, and the operational check of the gripper at the core and the transfer arm was completed. Maintenance work was also carried out on the force setter on the core holddown. All operations were satisfactory and normal fuel handling operations were resumed.

b. Cooling Tower. With the plant idle, no further attempts have been made to schedule the ASME performance test (see ANL-7357, p. 31).

Changes are being made to improve operability of the cooling tower. When cold weather and low plant loads result in frequent overflow cycles and bypassing to the basin, risers must be drained into the valve pits. As a result, drain pits are flooded and silt is deposited in valve parts. Additional drain valves are being installed on the risers above ground so that most of the contents can be drained to the ditch. The valve pits are being cleaned so that the remainder of the overflow water may be drained to and from the pits as quickly as possible.

c. Startup Feedwater Pump. Wiring was completed to the vibration switch mounted on the leakoff line from the latern ring chamber of the startup feedwater pump. This switch will automatically unload the pump in the event of packing failure which results in line vibration.

d. Repair of Mark-II Oscillator Rod Drive. Assembly of the Mark-II oscillator driveshaft is in progress. The modification to the sensing rod has been completed, and the gripper has been assembled on the upper tubing. Preparations for the seal welding and helium leak testing are in progress. Consideration is also being given to changing the flow around the existing new oscillator rod in its thimble to correct the bowing and rubbing encountered with the previous rod. The new rod contains one tube of B₄C instead of the two that were in the old rod.

7. High Power Operation

Essentially no progress was made on the proposal for 50-MWt operation (see Progress Report for July 1967, ANL-7357, p. 32). Personnel assigned to this item were employed full time on the analysis and investigation of the power-coefficient anomaly (see Section II.A.4.b).

The proposal for the reduced-flow experiment to test the effects of high-temperature operation on the performance of Mark-I-type fuel has been reviewed by both the Irradiations Review Committee and the Idaho Division Committee on Reactor Safety. The proposal will be revised to incorporate recommendations from these committees and from the Idaho Division Director.

A proposal for reorificing the driver-fuel subassemblies to provide increased cooling in Rows 4, 5 and 6 was prepared. The Irradiations Review Committee has been requested to review the proposal to install prototype subassemblies under thermocouple positions in Rows 4 and 5. Testing of these subassemblies is included to support the modification proposal.

8. Fuel Swelling and Driver Surveillance

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

(i) Irradiation of MC-S-type Fuel. Since Run No. 26 has not been started, there was no increment in burnup this month.

(ii) Encapsulated Irradiation of SL-type Fuel. Same comments as for Item (i) above.

(iii) Effect of Design Change. Same comments as for Item (i) above.

(iv) Effect of Enrichment. Same comments as for Item (i) above.

(v) Effect of Position and Orientation in the Casting Bundle. Same comments as for Item (i) above.

(vi) Effect of Trace-element Impurities. Subassembly 2077 (see Progress Report for July 1967, ANL-7357, p. 34) containing Illinois castings 409 (high-purity alloy) and 410 (trace elements C, Si, and Al added) has received no irradiation and is awaiting the start of Run No. 26.

The five melts to be produced at Idaho in the FCF have been cast, and are being analyzed for alloying elements and trace elements. The available data are given in Tables V and VI.

TABLE V. Trace-element Specifications for FCF-produced Melts

Melt No.	Trace Elements (wt ppm)				Ingot
	Carbon	Silicon	Aluminum	Iron	
I	^a	-	-	-	EUf-64S
II	250	300	-	-	EUf-58S
III	250	300	250	750	EUf-42S
IV	250	300	750	250	EUf-46S
V	-	300	-	-	EUf-59S

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

TABLE VI. Alloy and Trace-element Analyses for FCF-produced Melts

Melt No.	Ingot	Alloying Elements (w/o)					Trace Elements (wt ppm)	
		Mo	Ru	Rh	Pd	Zr	Al	Fe
I	EUJ-64S							
II	EUJ-58S	2.49	1.91		0.16		75	189
III	EUJ-42S	2.43	1.89		0.18	0.08	209	591
III	EUJ-42SR	2.43	1.92	0.281	0.18	0.08	126	750
IV	EUJ-46S	2.48	2.05	0.302	0.19	0.08	787	153
V	EUJ-59S		1.92	0.289	0.19		262	123

When ingot EUJ-42S was remelted to adjust the iron content, aluminum was lost. No carbon or silicon analyses are available, and no injection castings will be made until these analyses are completed.

(vii) Effect of Pressure: Three sets of six specimens each of U-5 w/o Fs, 52% enriched, jacketed in Inconel 625 with a 15-mil wall, have been placed in CP-5 to assess the role of pressure in the swelling behavior of this EBR-II fuel. These specimens are characterized by having relatively long (2.047 in.) plenums which have been sealed under predetermined pressures. This procedure was designed to maintain constancy of gas pressure on the fuel in spite of volume changes represented by swelling.

The status of the three groups of specimens as of August 21, 1967, is given in Table VII.

TABLE VII. Status of Pressurized Specimens of Melt No. 409

Capsule No.	Specimen No.	Cast Pin and Section No.	Date In CP-5	Planned Out of CP-5	Temp ^a (°C)	Pressure (psig)	Approx Burnup on 8-21-67 (a/o)
P-1	1	53-5	5-22-67	9-67	570	40	1.3
	2	-11	5-22-67	9-67	570	80	1.3
	3	-1	5-22-67	9-67	570	200	1.3
	4	-7	5-22-67	9-67	570	400	1.3
	5	-3	5-22-67	9-67	570	800	1.3
	6	-9	5-22-67	9-67	570	1600	1.3
P-2	1	53-4	7-21-67	12-67	580	2050	0.4
	2	-10	7-21-67	12-67	560	944	0.4
	3	-6	7-21-67	12-67	580	512	0.4
	4	-12 ^b	7-21-67	12-67	560	162	0.4
	5	-2	7-21-67	12-67	580	103	0.4
	6	-8	7-21-67	12-67	560	47	0.4
P-3	1	54-1	8-17-67	1-68	585	1950	0.05
	2	-2	8-17-67	1-68	585	975	0.05
	3	-7	8-17-67	1-68	585	487	0.05
	4	-4	8-17-67	1-68	585	243	0.05
	5	-5	8-17-67	1-68	585	98	0.05
	6	-6	8-17-67	1-68	585	49	0.05

^aThis is an average maximum temperature to date.

^bShort specimen.

The radiation capsules were provided with thermocouples at strategic internal locations, but no auxiliary heat to regulate temperature was provided. The fuel temperature and plenum pressure are therefore solely dependent upon heat generation in the fuel, subject to capsule geometry. The heat generation is a function of operating power level and position of the capsule in the reactor grid.

(viii) Hot-lab Examination of Irradiated Fuel

(a) Chemical Analysis. Another shipment of 30 irradiated fuel elements have been received from Idaho and is being sampled for chemical analysis. Most of this material is of the low-swelling type.

No new results on aluminum or iron analyses have been reported by Oak Ridge National Laboratory. Additional carbon and silicon results reported are included in Table VIII.

TABLE VIII. Chemical Analysis (Argonne, III., and Oak Ridge) of Irradiated Fuel Pins

Casting Batch and Pin No.		Melt Refining No.	Subassembly and Element No.	Chemical Analysis (wt ppm)			
				Carbon	Silicon	Iron	Aluminum
4001	1	22	C-184 87	107	88	470	295
4002	9	23	C-180 60	132	94		
4002	14	23	C-180 57	121	88		
4002	24	23	C-184 32	139	57	190	170
4002	82	23	C-184 60	146	89	155	80
4003	3	24 ^a	C-186 53	200	162	425	110
4003	55	24 ^a	C-186 52	199	165		
4003	19	24 ^a	C-186 46	198	290		
4003	48	24 ^a	C-184 57	192	217	520	195
4003	71	24 ^a	C-180 32	193	203		
4003	86	24 ^a	C-179 82	214	181	420	135
4003	90	24 ^a	C-179 81	211	216	455	140
4003	92	24 ^a	C-179 79	189	226	440	190
4004	9	25	C-184 7	108	140	435	125
4008	19	30	C-196 31	137	71	160	205
4011	8	C-29 ^b	C-196 88	250	128		
4020	92	39	C-198 9	283	138		
4022	33	C-30	C-217 3	187	180	340	160
4023	56	33	C-199 26	162	132		
4024	15	28	C-218 33	207	105		
4025	24	C-33	C-218 1	257	207	240	105
4025	44	C-33	C-217 78	262	213	280	215
4026	21	C-32	C-217 57	224	219	430	175
4027	16	46	C-218 64	206	170		
4027	24	46	C-217 60	229			
SL-5	-	-	Cast Pin	146	104		
SL-6	-	-	Cast Pin	141	126		
SL-12	-	-	Cast Pin		102		
SL-17	-	-	Cast Pin		54		
SL-20	40	-	C-185 2	149	85	150	175
SL-20	52	-	C-184 36	143	90	145	275
SL-21	20	-	C-186 60	153	87	155	200
SL-21	-	-	Cast Pin		70		
SL-29	51	-	C-184 43	157	102	200	225
SL-29	68	-	C-179 52	138	116	105	95
SL-30	8	-	C-184 37	217	108	255	140

^aMR-24 was a C-type melt.

^bMR-C-29 was a melt of casting heels only.

(b) Microstructural Examination. Detailed metallographic examination by Pacific Northwest Laboratory of three different Mark-IA fuel types (SL, MR, and MRC) has been completed. Typical chemical analyses of the three U-5 w/o Fs fuel types and irradiation conditions for the fuel pins examined are given in Table IX. The SL-type fuel material cast at Argonne, Illinois, typically had the lowest concentration of impurity atoms and also showed the highest swelling rate in EBR-II, swelling at an average rate of 14 to 16% per a/o burnup. The MRC-type fuel material represented by casting batch 4003 typically had the highest concentration of impurity atoms and showed the lowest swelling rate in EBR-II. A chemical analysis² of several pins from each casting batch was presented in Table XIII (p. 28) of the Progress Report for June 1967, ANL-7349.

TABLE IX. Typical Chemical Analyses of EBR-II Fuel Pins Examined by Pacific Northwest Laboratory

Type Material	Melt No.	Subassembly and Pin No.	Maximum Subassembly Burnup (a/o)	Typical Chem Analysis (ppm)			No. of Sections Examined
				C	Si	Fe	
SL	SL-20	C-185-40	1.13	145	90	150	3
MR	4002	C-180-14	0.82	130	80	170	3
MRC	4003	C-180-71	0.82	190	200	450	13

Transverse cross sections were prepared for metallographic examination from the top, middle, and bottom of the SL- and MR-type fuel pins. Pin C-180-71 (MRC type) prepared from consolidation melts in FCF was examined in detail at 13 equally spaced transverse cross sections.

Prior examination of unirradiated injection cast U-5 w/o Fs fuel pins showed the structure to consist of equiaxed gamma grains approximately 20μ in diameter with U_2Ru needles precipitated at the gamma-grain boundaries. Below $552^\circ C$, alpha-uranium, U_2Ru , and U_2Mo are the phases in equilibrium. Isothermally annealing above $552^\circ C$ transforms some of the alpha-uranium to bcc gamma and causes the U_2Ru needles to spheroidize.

Hot X-ray diffraction of four samples from pin C-180-71 and two locations from pin C-180-14 showed the structure to consist essentially of alpha uranium with a slight amount of retained gamma. Each of the 19 specimens was examined in the as-polished and in the etched condition with the light microscope at magnifications from 75X to 750X. Electron microscope replicas were prepared from the outer edge and the fuel-centerline

²Ppm--on a weight basis.

areas of each of the 19 specimens and examined in the electron microscope at magnifications from 3600X to 18,000X. General conclusions drawn from this detailed examination are the following:

(1) No indication of either sharp gamma-grain boundaries or the precipitation of U_2Ru needles was observed in any of the irradiated samples examined by electron microscopy. Unirradiated U-5 w/o Fs heat treated at the same temperature and for the same time as the EBR-II fuel pins show heavy precipitation of both the U_2Ru and the U_2Mo intermediate phases. Some indication of prior gamma-grain size, probably as a result of coring in the original as-cast structure, could be seen in the irradiated fuel in both the as-polished and the etched condition at low magnifications. However, no sharp gamma-grain boundaries were detected in the irradiated samples nor was there any indication of enhanced precipitation of intermediate phases at the prior gamma grain boundaries.

(2) No apparent structural differences were observed between the three different U-5 w/o Fs fuel types (SL, MR, MRC) examined at comparable burnups and temperatures. All three fuel types showed a regular distribution of carbide particles in a uniform matrix.

(3) The three types of material showed a uniform distribution of very fine microtears, approximately 0.1μ in length, that were visible only when examined with the electron microscope at high magnification. The number of microtears per unit area increased with both temperature and burnup, i.e., an area in the center of the fuel pin, irradiated at a higher temperature contained a larger number of microtears than an area near the circumference of the fuel pin. The density of microtears increased with an increase in burnup at constant temperature. Based on the appearance of the irradiated structure and the observed dependence of swelling on temperature, it is concluded that the controlling swelling mechanism in uranium-5 w/o fission irradiated in EBR-II is mechanical tearing. Density measurements made on three sections from pin 71 and on the middle and top sections from pins C-185-40 and C-180-14 in general corroborated the volume changes as calculated from diametral measurements of the bare fuel pins.

(ix) Analysis of Fuel-pin Swelling. The figure of merit $Y/(Bu)_{\max}^2$ (see ANL-7357, pp. 35-36) has been used to separate all of the available swelling-burnup data into four groups. Analysis of the groups by plotting the ratio $Y/(Bu)^2 = R$ against burnup, treating each group separately, is not quite complete as yet, but preliminary results indicate the following:

(1) For the low-swelling group, R has a value of approximately 4 and is nearly independent of burnup up to 1.2 a/o maximum burnup. We have no swelling data beyond this burnup for this group.

(2) For the moderate-swelling group, R has a value of approximately 6 and shows a slight dependence upon burnup up to a value of about 1.15 a/o maximum burnup. Beyond this burnup, the R value breaks upward.

(3) The high-swelling and the very-high-swelling groups are not clearly distinguishable.

(4) Most of the SL-type fuel falls in the high- and very-high-swelling groups, but a few of the melts seem to belong in the low- and moderate-swelling groups. However, these data are for relatively low burnups of about 0.75%. We will obtain additional data on these melts at burnups of approximately 1.15 a/o through surveillance examination of fuel elements from subassembly B-343, which reached its target burnup at the end of Run No. 25.

(5) There is still considerable scatter in the data. At least part of this scatter can be ascribed to variations in fuel temperature. During the next month operating temperature will be factored pin by pin into the analysis.

The correlation of swelling behavior with silicon content now includes additional analytical results (see Table X). Some of the analytical data may be of questionable validity, since some of the values are from a single determination on a single pin, but all available values have been included. In general, one can say that if the silicon content exceeds about 200 wt ppm, the swelling is low or moderate; if the silicon content is less than about 150 wt ppm, the swelling is high or very high. If the silicon content is between 150 and 200 wt ppm, the swelling is erratic. As mentioned under Item (viii) above, a new lot of fuel pins, many of which are of low-swelling material, are now being sampled.

TABLE X. Swelling Behavior and Trace-element Content

Casting Batch	$\gamma/\text{Bu}^2_{\text{max}}$	Carbon (ppm)	Silicon (ppm)	Aluminum (ppm)	Iron (ppm)	Casting Batch	$\gamma/\text{Bu}^2_{\text{max}}$	Carbon (ppm)	Silicon (ppm)	Aluminum (ppm)	Iron (ppm)
4001	8.46	107	88	295	470	4027	10.18	218	188	77	285
4002	9.89	135	76	139	148	4031	5.23	210	340	234	440
4003	6.17	200	208	170	423	4032	6.16	238	150	161	375
4004	9.60	108	140	125	435	4033	4.34	204	243	153	365
4008	9.89	137	71	205	160	4034	10.79	106	102	136	294
4010	3.95	243	200	217	330	4035	4.36	178	250	121	665
4011	11.25	250	140	198	292	4036	4.33	188	270	20	330
4020	11.33	283	138	-	-	4037	2.31	215	160	187	350
4022	9.15	187	163	191	292	4062	-	-	220	-	230
4023	11.15	162	132	-	-	4068	5.05	-	170	-	265
4024	10.12	207	88	229	240	4073	3.85	-	160	-	290
4025	5.14	260	198	118	267	4103	6.23	-	170	-	330
4026	5.95	224	189	224	465	4124	-	-	-	150	500

As a result of these findings, a recommendation has been prepared that the minimum silicon content in the EBR-II driver fuel be specified as 250 wt ppm. It has also been recommended that this revised

specification be put into effect without delay because of the imminent production of driver fuel with virgin starting materials. Melts of this type would be expected to have silicon contents significantly less than 250 wt ppm unless deliberate additions are made.

No definite specification on maximum silicon content can be made at this time because of inadequate information, but low swelling has been exhibited by driver fuel with the highest silicon content thus far determined (340 wt ppm). Also, experience at other sites with a similar alloy, U-4 w/o Mo, shows that low swelling is obtained with silicon contents near 1000 wt ppm. Recommendations are that, until additional data are available, efforts be made to keep the silicon content in the driver fuel from exceeding 500 wt ppm. Melts containing 500 to 1000 wt ppm silicon should not necessarily be rejected, but should be usable with adequate surveillance procedures.

b. Mark-II Driver-fuel Development

(i) Fuel for Higher Temperatures and Burnups. The EBR-II Mark-I series of driver-fuel elements offer limited possibility for improvement of achievable burnup levels. The presently allowable peak burnup is 1.2 a/o at a reactor power level of 45 MW. The allowable burnup level is expected to decrease to a peak of approximately 1.0 and 0.8 a/o, respectively, when the reactor power is increased to 50 and to 62.5 MW. The anticipated decreased allowable burnup results from the expected higher rates of fuel swelling that are associated with the higher fuel-element temperatures at the increased power levels. The lower fuel burnups will in turn result in correspondingly increased consumption rates of fuel subassemblies. Subassembly consumption rates will be further increased when the reactor plant factor is increased to 70%, as projected.

Clearly, a more advanced design of driver fuel element is required with performance goals that will permit burnups of at least 2 a/o to be achieved at a reactor power level of 62.5 MW. Developments in the LMFBF fuels program indicate that development of an advanced (Mark-II) design of an element containing U-5 w/o Fs alloy is feasible and should be undertaken without delay.

The design of the Mark-II element is predicted on the experimentally observed fact that, if metallic fuel is permitted to swell sufficiently within its jacket, interconnection of voids within the fuel permits escape of enough fission gas to enable the jacket to restrain further swelling. Experimental irradiations under the LMFBF program have shown that experimental metallic fuel elements designed on this basis can be irradiated to at least 4.6 a/o burnup without jacket failure.

In the light of these observations, a reference design for the Mark-II element has been established. The fuel alloy is U-5 w/o Fs alloy, sodium-bonded to a Type 304 stainless steel jacket. The dimensions, along with those of the Mark-I and -IA designs, are as follows:

	Mark I	Mark IA	Mark II
Fuel-pin diameter, in.	0.144	0.144	0.130
Fuel-pin length, in.	14.2	13.5	14.2
Sodium bond thickness, in.	0.006	0.006	0.010
Jacket thickness, in.	0.009	0.009	0.012
Jacket ID, in.	0.156	0.156	0.150
Jacket OD, in.	0.174	0.174	0.174
Fuel-pin restrainer gap, in.	0.40	0.40	0.70
Plenum length, in.	2.2	2.9	8.3
Fuel-element length, in.	18	18	24

(ii) Element-irradiation Tests. To determine the effect of variations in design parameters on fuel-element performance, a total of 62 elements, individually encapsulated for irradiation on a substitution basis in a Type B-37 special irradiation subassembly in EBR-II, will be used. Postirradiation examinations will be made at each projected level of 0.5, 1, 2, 3, 4, 5, 5.5, and 6 a/o burnup. Both 80 and 93% enriched fuel will be used so that simulation can be obtained of reactor operation at 50 and 62.5 MW. The irradiations will be used to evaluate the relative performance of the reference 24-in.-long element compared to one 26 in. long. Two different gas-plenum volumes will thereby be tested. The irradiations will also be used to determine the relative performance of three candidate axial-fuel-restrainer designs.

Calculations showed that, in addition to the variation due to the two plenum lengths, the bond-sodium overfill had a significant effect on the final plenum pressures. The variables affecting the plenum pressures were represented by two families of curves (see Figs. 4 and 5), one

Fig. 4. Jacket Stress vs. Sodium Overlay for 24 in. Element

Fig. 5. Jacket Stress vs. Sodium Overlay for 26 in. Element

for each fuel-element length, calculated to show the effect of sodium-overfill height on the hoop stress. The jacket stress includes one-fourth of the thermal stress as well as the hoop stress, in accordance with the maximum level of combined stresses allowed by the "Miller Criterion" (D. R. Miller, Trans. ASME, pp. 190-196, June 1959).

A total of 67 injection-cast fuel pins jacketed, sodium bonded, and sealed in Type 304L stainless steel tubes of 0.174-in. OD and 0.150-in. ID were received. Of these 35 were jacketed in tubes 26 in. in length; the remainder were in tubes 24 in. in length. These variations allow testing (i) the effect of pressure over a relatively wide range and (ii) different burnup levels. An important factor influencing the generation of pressure in the jacket is the sodium level. Levels of 0.21 and 0.71 in. above the top end of the fuel pin were aimed at in the 24- and 26-in.-long elements, respectively.

The elements were radiographed and the sodium-bond integrity was examined by means of the pulsed electromagnetic technique. The elements were then pickled to remove the black oxide that results from the impact bonding procedure, rinsed thoroughly in water, weighed, dimensionally inspected, and density measurements were made by immersion in carbon tetrachloride.

The fuel elements were then placed into secondary containment capsules to which had previously been added a predetermined amount of sodium for conducting the heat generated in the fuel element to the EBR-II coolant sodium. The capsules were then sealed by welding in an atmosphere of helium, leak tested, and finally bonded by vibrating in a furnace at 500°C for 3 hr. This step was considered necessary to promote wetting of all the surfaces by the sodium and thus eliminating voids or gas bubbles which could cause hot spots. This type of defect was closely examined for by radiography and pulsed electromagnetic inspection. After completion of these tests, extension elements with capsule identification were welded to the ends and the capsules were shipped to Idaho for assembly into a B-37-type experimental subassembly. Thirty-seven capsules constituted a subassembly with 25 additional capsules to substitute at burnup intervals of 0.5, 1, 2, 3, 4, 5, and 5.5 a/o.

A safety analysis based on nominal conditions was prepared and sent to EBR-II for review in May 1967. In July 1967, an addendum was issued wherein specific conditions, i.e., sizes, weights, etc., of the as-built elements were used to make the necessary calculations for a more accurate safety analysis.

(iii) Subassembly Proof Tests. It is desired to characterize the performance of the Mark-II fuel with selected design specifications under conditions amenable to statistical treatment, particularly in the establishment of confidence levels relating to fuel performance. The

second phase of the evaluation of the in-reactor performance of the Mark-II element consists of irradiations of unencapsulated elements in complete Mark-II 91-element subassemblies. These subassemblies will therefore function in EBR-II as driver-fuel subassemblies. They will be exposed directly to sodium flow, so that any effects due to the primary coolant can be determined.

Five such subassemblies are scheduled for irradiation in the range of 2 to 5 a/o burnup to begin in FY 1968. Their insertion will be timed so that they do not accumulate more than 0.5 a/o burnup before the encapsulated elements with 2 a/o burnup (see Sect. II.A.8.b.ii) have been removed and examined.

Selection of the preferred axial-fuel-restrainer design and the gas-plenum volume for these irradiations will be based on results of examinations of the irradiations at 1 and 2 a/o burnup.

The necessary hardware for the assembly of the elements is being purchased or manufactured. A total of 2478 ft of Type 304L stainless steel tubing for the fuel jackets has been received and nondestructively examined. Of this amount, 7% was rejected for defects greater than 10% of the wall thickness. A total of 3316 ft of Type 316 stainless steel tubing having the same dimensions was subsequently received (from a different vendor), and 33% rejected for the reason given above. Despite the high rejection rates, sufficient quality tubing of both types is available to fabricate 1000 Mark-II elements.

An order has been placed for the manufacture of 1000 top and bottom end plugs. Arrangements are being made for the casting of the pins and assembly of the elements in the Idaho FCF cold line. One-thousand Mark-II-size injection-casting Vycor molds have been transferred from Illinois to the Idaho site for this purpose.

9. Fuel-related Equipment

Equipment related to fuel handling and reactor control rods is being improved to raise the capability for fast-flux irradiation, reliability, and performance of EBR-II.

a. Improved Gripper and Holddown Force-limit Device. To increase the reliability, accuracy, and sensitivity of the core gripper and holddown push-and-pull force-limiting devices used in fuel-handling operations, improved force-limit devices have been designed to control remotely and monitor the forces exerted by the fuel-handling mechanisms. A prototype is being fabricated and will be tested.

Components for the test device have been fabricated and are being assembled. Components for modifying the gripper mechanism are being fabricated on schedule (see Progress Reports for May 1967, ANL-7342, p. 6, and for February 1967, ANL-7308, pp. 3-4).

b. 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.

(i) Control Rod of Greater Worth. 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.

The preliminary computations on the all-absorber B_4C control rod indicate that the life expectancy of the new control rod would be a minimum of six months; the internal pressure then would be 1310 psi, and the pressure stress developed in the stainless-steel containment tube would be 11,700 psi (see Progress Report for July 1967, ANL-7357, pp. 6-7). Bases of the computations were that (a) the B_4C absorber is in a hollow cylindrical shell of 1.750-in. OD, 0.632-in. ID, and 16-in. length, (b) B^{10} burnup would be 2 a/o, (c) B_4C density is 86%, (d) B^{10} enrichment is 94%, (e) heat generation is 290 W/cm³, (f) 100% gas release to a 63-in.³ space, (g) the control rod is located 8.036 in. from the core center, and (h) the reactor operates at 62.5 MW.

With the reactor sodium inlet at 700°F and outlet at 900°F, the sodium flow through the control rod is 20.6 gpm. The boron carbide would have a maximum temperature of 3580°F and a radial thermal expansion of approximately 0.008 in. The radial clearance between the boron carbide and the cladding would be 0.012 in. According to the ASME unfired-pressure-vessel code, the design fiber stress at 100°F is 12,500 psi. Therefore, due to pressure buildup, the mechanical limitation of the cladding would be reached in approximately 6.4 months. The estimations were based on the very conservative assumption of 100% gas release.

A research and development program has been tentatively planned to furnish basic information in the areas for which experimental data are not available. The program would entail irradiation, calibration, and helium "venting" experiments. The irradiation experiment would involve (a) irradiating a prototype B_4C section in a modified core-type subassembly in which the upper reflector section was replaced, (b) simulating the control-rod-withdrawn condition (20% inserted) during the irradiation in a Row-5 location, (c) measuring dimensional changes, composition, and pressure of released gas, (d) measuring boron carbide burnup as a function of length, and (e) determining the gas-release fraction.

The calibration experiment would involve (a) constructing a prototype control rod with correct boron enrichment and B_4C density (even though questions of compatibility, burnup, etc., are not resolved), (b) operating the prototype in the reactor (possibly only at low power) to determine worth versus position in at least two control-rod positions while determining flux distributions in adjacent subassembly positions; and (c) using the information obtained to determine whether drive speeds, stroke, etc., are acceptable.

For the helium "venting" experiment: (a) since the gas released is helium, consider the possibility of venting the gas to the sodium, (b) test the compatibility of the B_4C , cladding, and sodium combination, and (c) test the helium release in sodium (design of release valve). The tests would involve a control-rod sodium-environment test facility for all control-rod modifications; static sodium would be used but there would be provisions for flowing sodium and for all control-drive mechanical functions.

c. 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.

The slight binding in the travel of the instrument support plate was corrected (see Progress Report for July 1967, ANL-7357, p. 7). The instrumentation is operating satisfactorily. All equipment has been operated and functions in accordance with design requirements. Crates for shipping the equipment to Idaho are being fabricated.

d. Mark-II Oscillator Rod. 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 detailed drawings for the Mark-IIA reduced-diameter oscillator rod, the special thimble, the storage-basket adapter tube, and the special gripper for handling the adapter tube have been completed and are being checked (see Progress Report for July 1967, ANL-7357, p. 7).

e. Engineering Consultation. This work in support of EBR-II operation and improvement involves responses to various unanticipated conditions requiring corrective action, as well as planning and evaluation of new programs.

(i) Control-rod Bushing. Examination of control rods removed from the reactor has indicated corrosion or erosion of the beryllium-copper-bearing surfaces. An engineering study is being conducted to determine the

cause of this condition and to select a bushing material that can provide more satisfactory service, as well as to re-evaluate the design configuration and material composition of the top and bottom guide bushings of the lower preassembly of the control rods.

Aluminum bronze is the primary choice for replacement. This material has been used satisfactorily for applications of similar nature in the reactor sodium. Special bushing assemblies, consisting of alternate aluminum-bronze and beryllium-copper rings, have been fabricated and assembled into a special test control rod. After this test control rod has been installed and used in the reactor for a specified time, it will be removed and inspected to obtain an accurate comparison between the two materials.

Two potential alternate designs for bushing assemblies have been selected. One design would utilize the general configuration used for four separate rings, but would be a single-piece bushing. The second design would maintain the same overall dimensions, but the labyrinths would be simplified to provide flat surfaces on the lands and grooves instead of the multiple steps of the original design. This change should not appreciably affect the leakage rate past the bushings and should be less expensive to fabricate. Drawings of the alternate-design bushings have been completed and are being reviewed.

10. Reactor and Primary Coolant System Equipment

a. Ventilation in Purification and Sampling Cells. After radioactivity had sufficiently decayed, the Purification Cell was entered to determine the reason for low flow through the cell-exhaust duct. The duct inlet was partially plugged with dust. The removal of this restriction resulted in an increase of exhaust flow of about 300 percent. The ventilation system for the Purification Cells is now performing satisfactorily.

The proposal for modifying the ventilation in the Sampling Cell (see Progress Report for July 1967, ANL-7357, p. 39) was approved. An engineering submittal is being prepared to remove temporary and establish permanent ventilation ducting.

b. FERD Loop Plugging Meter and Sampler. The installation drawings for the FERD loop plugging meter and sampler are nearing completion. Prefabrication of components for this system is progressing.

A proof test of a prototype FERD loop plugging meter has been successfully completed. The test facility included a portable sodium-charging tank with a capacity of approximately 100 gal. Because of the capacity of this tank and the low capacity of the EM pump used in the

plugging loop, sufficient mixing to provide sodium homogeneity was not possible, and control of the plugging temperature was not feasible. The plugging temperature was determined by noting the temperature at which a break in the flow curve occurred. During the test, plugging temperatures in the range 250 to 360°F were detected. The breaks in the plugging meter flow curves were quite easily detected.

c. New Plugging Meter for the Primary and Secondary Purification Systems. The plugging valve for the new plugging meter has been successfully proof tested. Some of the components for the system are being prefabricated.

d. Blanket-gas Monitor. It is proposed to monitor the reactor gas blanket for the radioactive isotopes xenon-133, xenon-135, and argon-41 on a continuous basis. This is currently being done on a routine basis by samples which are analyzed in a 256-channel analyzer. The new monitor would readout and alarm in the control room.

Also as part of the improved instrumentation, it is proposed to indicate in the control room the nitrogen, hydrogen, and oxygen contents of the argon cover gas. These gaseous contaminants are currently indicated on a bar-graph recorder in the depressed area of the reactor building. These signals will be retransmitted to the control room.

The new instrumentation is intended to provide better reportable information from the reactor system and to improve surveillance of the gas system. In particular, it will aid in detection and evaluation of gaseous fission product release from the reactor fuel and serve as a backup to the fission-gas monitor.

The specifications for this equipment have been written and submitted to a reviewing committee. The equipment is expected to be purchased on bid.

e. Surge Tank for the Primary-sodium Purification System. Detail design is complete and drafting of drawings is underway to add 8 in. of height plus internal baffling to the surge tank. The 8-in. increase in height was chosen because (1) operating experience indicates the present height is marginal to prevent the sodium level from reaching and plugging the vacuum line, and (2) an 8-in. increase minimizes the changes necessary in shield blocks, and allows all valves and operators to remain below the present reactor building main floor plates. It will be necessary to provide a raised area in the reactor building floor plates equal to the height added to the surge tank.

f. Primary Tank Pressure Relief. The proposal for installation of this additional primary tank pressure-relief system (see ANL-7357, p. 37) has been reviewed and accepted. Final design layout and location of all system components is underway.

The pressure-vacuum relief device, consisting of an oil-filled "U" tube with 2 in. of length added to the short leg, will be fixed to a platform attached to the outer shield wall of the primary tank. A large tank, about 4 ft in diameter by 15 ft high, will be located just alongside of and below the relief device. The tank will be located in a vertical position, resting on the basement floor, and will contain a longitudinal internal baffle, except for the bottom portion. Four-inch-diameter piping with a shutoff valve will connect the reactor "N2" nozzle with the relief device. This valve will be permanently locked open after installation. A 4-in. line will connect the discharge from the relief device to one side of the baffle in the tank. A tank-discharge line will connect to the other side of the baffled tank and to two parallel lines with check valves. One line will open to reactor building atmosphere, while the other line will tie into the suspect exhaust system.

g. Engineering Consultation.

(i) A cost estimate for the installation of the main secondary EM pump has been completed. The estimate will be used in obtaining a Davis-Bacon determination and will also be used for planning the installation work.

(ii) A meeting was held to discuss the high-temperature nuclear-instrumentation test facility planned for EBR-II. Discussion included scheduling scope of the project, test program, FFTF test requirements, etc.

(iii) A series of discussions were held with the special ANL group investigating the troubles encountered with operation of the fuel-handling gripper. Possible causes of the malfunction and future operating plans to avoid a recurrence of this situation were covered.

(iv) Discussions were held with representatives of the Liquid Metal Engineering Center (LMEC) relative to operating experience of EBR-II equipment in sodium. A representative from LMEC is spending approximately one month at EBR-II gathering information on all operating equipment in the EBR-II primary tank. Several discussions have been held with the LMEC representative concerning component design, operating problems, and material selection.

h. Instrumentation

(i) Detector for Bulk Sodium Level. During repair of the core gripper the primary tank temperature was lowered to 510°F and then returned to 700°F. The raising and lowering of the tank temperature provided an additional check on the level-detection system. For the temperature limits stated, the primary tank level will vary 6 in. The level detector followed this level change, indicating the calculated level within 0.4 in. on decreasing temperature (level) and within 0.3 in. on increasing temperature (level).

No spurious alarm trips occurred during this reporting period.

(ii) Rod-position Indication. To improve the accuracy and reduce the number of components in the servo system for rod position, a shaft encoder has been purchased for conversion of the shaft position into digital data proportional to rod position. The shaft-encoder output will then be fed to counting circuitry for conversion to engineering units and to a light bank for readout. Only the fine servo (324° rotation per inch of rod travel) will be used in this system.

Design and fabrication of the integrated circuit counter is in progress. Testing of the system will start following completion of the counter.

(iii) FERD Loop Instrumentation. The FERD Loop Channels have been connected in a 2-of-3 coincidence circuit and added to the reactor shutdown circuit. The signal for reactor shutdown was obtained by adding a millivolt to milliamp (MV/I) converter to the output of the log count-rate amplifiers and using the output of the (MV/I) converters to drive a monitor switch and the recorder. To protect against a loss of detector high voltage, a monitoring amplifier was designed to monitor the output of the high-voltage power supply and to initiate a trip on a reduction of voltage of approximately 20% of normal.

This study to upgrade the equipment in the system has been limited to a continual review of detectors for high sensitivity to neutrons in a strong gamma field. A decision on detectors will be required before design work can start on the moderator and shield.

11. Secondary Sodium and Power System

a. Inspection of Steam Generator. Studies of the several approaches to provide for inspection of the steam generator are progressing. Cost data for original cost, installation, and disassembly and reassembly for the different design concepts are being gathered.

12. New Subassemblies Design and Experimental Support

a. 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 requires changes to the upper and lower axial-reflector sections. Out-of-pile and in-pile tests will be used to verify the design.

(i) Fuel Subassembly. Mark-II and Mark-IIA fuel assemblies are being designed and will be flowtested to ensure that their pressure drops match those of the existing fuel subassemblies.

Development of the Mark-II and Mark-IIA subassemblies continues. These subassemblies are being developed to provide the 24-in. element length (Mark-II elements) and an alternate 26-in. element (Mark-IIA).

Drawings for the reference design for the Mark-II (24-in.-long) fuel-element subassembly have been completed and checked, and are being reviewed. A technical memorandum describing the Mark-II core subassembly design and test is being prepared.

Preliminary drawings for the test prototype design of the Mark-IIA (26-in.-long) fuel element have been completed. A model is being fabricated for testing in the water test loop to evaluate flow and pressure-drop characteristics. Because of the 2-in. extension of the fuel-element gas space, the subassembly upper shield section had to be shortened an equal amount. The effect of the shorter shield on pressure drop will be determined and adjusted to match the standard subassemblies for each row position in the core.

b. Radial Blanket. To increase the reactivity available in the core so that more irradiation experiments can be accommodated, replacement of the stainless steel reflector elements in Rows 7 and 8 by nickel elements is being studied. Reactivity, corrosion, transport, activation, and deposition processes involved in the use of large amounts of nickel in the reactor are being evaluated.

The replacement of seventh and eighth row uranium blanket subassemblies with stainless steel reflector subassemblies was accompanied by a change in reactivity coefficient in Run 25. As an experiment to provide information which will lead to better understanding of this change, it has been proposed that the stainless steel reflector subassemblies of rows seven and eight be replaced temporarily with uranium blanket subassemblies.

The reflector subassemblies of row eight would be exchanged with blanket subassemblies presently located in an outer row of the blanket. New uranium blanket subassemblies must be fabricated to replace the stainless steel reflector subassemblies in row seven. To complete the fabrication of the new blanket subassemblies at an early date, it will be necessary to use modified type "B" lower adapters which are available and are normally used for sixth row fuel subassemblies. The flow requirement for a blanket subassembly is substantially less than that for a fuel subassembly. To provide for this reduced flow, the Type "B" lower adapter will be modified to include a flow control orifice. The orifice design has been completed and flow test on a prototype of the blanket subassembly is in progress.

Detail drawings of the prototype Mark-II blanket subassembly have been completed. Fabrication of a model, to be used for water flow-tests, has started.

The design is based on the results of the heat-transfer calculations which indicate that a solid stainless steel blanket cannot be used because of high heat-generation rates and the resulting thermal stresses in the large cross section. The decision to use a 2-in. solid section of chrome-plated nickel is based on the acceptable thermal properties of nickel and its corrosion resistance, which is equivalent to that of stainless steel. The proposed subassembly has a solid hexagonal reflector with mechanically fastened spiral flow-divider strips that tend to equalize temperature distribution and, thus, eliminate bowing tendencies.

The compatibility of sodium with chrome-plated and with unplated nickel is being studied.

c. 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.

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

Detail drawings for flowtest prototypes of Model-E19 and Model-E37 are complete. A special test-facility arrangement is being developed for the Mark-E subassembly; to simulate a partial meltdown of the fuel-element bundle, it will release measured quantities of various sizes of heavy-metal particles in the fuel section during full-flow conditions. The particles held by the upper and lower collectors will be weighed at the completion of each test to determine the effectiveness of the design.

d. 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) Instrumented Subassembly. The concepts for the instrumented assembly and associated guide thimble that will locate the subassembly in the reactor have been devised, including the arrangement of the hexagonal tube that contains the fuel elements, instrument sensors, a bottom adapter, and a standard EBR-II cone-tipped upper adapter that couples to the extension tube (see Progress Report for March 1967, ANL-7317, p. 15).

Development of the fuel-element capsules continued. The 19-element array in an instrumented subassembly will accommodate 18 capsules, with the nineteenth space occupied by an instrument-lead conduit. These capsules will contain experimental fuel elements or other experimental materials; sodium between the experimental fuel element and the capsule wall will transfer heat. The capsule will contain fuel and fission products if the fuel-element cladding fails.

To provide for experimenter requirements, certain dimensions and features within the capsules are variable: the length of the fuel and fuel container, the height of the gas spaces and sodium levels in the fuel element and in the capsule, the location of thermocouple junctions, and the use of a vanadium barrier cup surrounding the metallic fuel element. An overall length of 40 in. has been set tentatively to allow sufficient space within the subassembly for arrangement of the instrument leads during assembly and to accommodate a flowmeter.

Coolant thermocouples can be wound around the outside of the capsule at a 6-in. pitch, in which case the thermocouple also serves as the spacer wire. In addition, each capsule can be appropriately instrumented to measure fuel-centerline temperatures and/or fission-gas pressures. For the ceramic-type fuel, a splice chamber in the fuel-element cap may be required to form a transition between the high-temperature thermocouples and the more conventional leads. For measuring fuel-centerline temperature in the metallic-fuel capsules, Chromel-Alumel thermocouples can be used; because they are more ductile; they will not require splicing.

(ii) Subassembly Location in Core. Plans are being made to install a prototype instrumented-subassembly system and to have the potential capability for installing instrumented subassemblies in four of the 12 control-rod penetrations at any one time.

The four positions are being selected on the basis of maintaining reasonably uniform control-rod spacing; access considerations and the installation of an oscillating rod and a drop-test rod are other factors influencing the selection of positions. It is desirable to have one or more control rods between adjacent instrumented subassemblies or between an instrumented subassembly and the oscillating rod. Because there is less congestion on the southeast side of the small rotating plug, this area is being considered for initial installation of the prototype instrumented subassembly. The No. 6 control-rod penetration is preferred on the assumption that the No. 8 position will be occupied by the oscillator rod. Subsequently, the No. 8, 10, and 2 locations may be occupied by instrumented subassemblies, which would maintain a reasonably uniform control-rod pattern. No. 11 could be considered in lieu of No. 10 for improved control-rod distribution.

13. Process Chemistry

a. Quality Monitoring and Control of Sodium Coolant

(i) Sampling of Primary Sodium. Samples of surface sodium were taken through the Fuel Transfer Port in order to determine the amount of cesium-137. Two attempts were made to obtain samples of any solid surface material by contacting the sodium surface with $1\frac{1}{2}$ -in. squares of stainless steel screen.

Two samples of sodium, one from the surface in the Fuel Transfer Port and the other from a depth of about 3 in. below the surface, were taken in extrusion-type sample vessels, using a suction technique.

(ii) Analysis for Carbon in Primary Sodium. The analyses performed on a sample from the primary sodium system taken in July gave such a wide scatter that the data are considered to be unreliable. Apparently the scatter was due to the presence of sources of contamination, a condition now corrected.

(iii) Segregation of Radiocesium in Samples of Primary Sodium. During further investigation of impurity segregation in sodium (see Progress Report for June 1967, ANL-7349, pp. 37-39), some fair autoradiographs (beta) of Cs^{137} in sodium samples were obtained. Eight autoradiographs on various sections through slow-cooled primary sodium samples were made.

The conclusions which may be tentatively drawn from examination of these radiographs are:

- (1) Cs^{137} migrates in sodium samples at some time during the cooling process.

- (2) Cs^{137} tends to collect at interfaces, whether they are sodium-solid or sodium-gas.
- (3) The material at the interface with sodium apparently makes no difference.
- (4) There is no axial segregation of Cs^{137} in right circular cylindrical samples.

(iv) Analysis for Oxygen in Primary Sodium. The glovebox for the amalgamation method has been installed. Present effort is being directed toward reducing the water and oxygen levels in the glovebox atmosphere prior to making analytical runs.

The mass-spectrometric results for the reaction products from a test of a cyanide reduction method for determining oxygen in sodium (see ANL-7357, p. 41) are being evaluated to determine whether further work on this method can be justified.

(v) Tritium in Primary Argon. A tritium concentration of 1.2×10^{12} tritium atoms per liter of primary argon was reported in July. A sample taken on July 24, 1967, after additional reactor operation, contained 1.6×10^{12} tritium atoms per liter of argon.

(vi) Primary Sodium Plugging Temperature. The plugging temperature of sodium in the primary system remained below approximately 225°F during the entire month.

(vii) Primary Argon. During the month, the pressure of oxygen in the cover gas was twice indicated. The first occurrence was due to a malfunction of a recording instrument. The second occasion was when the Charged Wire Monitor (CWM) was secured. The CWM gas-handling system supplies sample gas to the chromatograph. As much as 1600 ppm O_2 was recorded. There are two loops (A and B) and the CWM circuit. When the standby loop was put in service, the oxygen and nitrogen contents returned to normal. A general tightening of fittings and leak testing of the system was performed. Further investigation of possible sources of air contamination is proceeding.

Four periods of abnormally high hydrogen content were also recorded. Two were in conjunction with fuel handling and two with leaks in the CWM system.

(viii) Secondary Sodium Sampling and Analysis. The equipment for sampling secondary sodium operated satisfactorily. The plugging loop and sampling station were frozen on August 20 to permit installation of plugging loop test equipment.

TABLE XI. Analysis of Secondary Argon by Continuous Chromatograph

	Hydrogen (ppm)	Nitrogen (ppm)
High	25	2600
Low	4	1200
Average	10	1600

(ix) Secondary Argon. The data in Table XI, following, summarize readings from the continuous chromatograph recorder chart.

(x) Expanded Sampling Capabilities for Primary and Secondary Sodium

(a) Secondary Oxygen Meter Loop. All hardware is on hand

for installation of an economizer in the oxygen meter loop to permit operation of the oxygen meters at the recommended temperature of $600 \pm 5^\circ\text{F}$. The loop has been shut down for special tests of the plugging meter to be installed in the FERD loop. The economizer will be installed at the completion of these tests. It is expected that the electrolyte will be placed in the detector housings and testing of the oxygen meters started in early September.

(b) Sodium Sampling Devices and Hardware. The double-wall extrusion vessel for cooling sodium samples with liquid nitrogen was successfully tested.

Samples of primary sodium have been taken in the double-wall extrusion vessel and in a large cylindrical vessel with a helical cooling coil bonded to the outside surface. Both samples were cooled with liquid nitrogen. The cooling curves are shown in Fig. 6. The

Fig. 6. Rapid Cooling of Sodium Samples

slower cooling rate shown for the smaller vessel is attributed to the remote location of the thermocouple and the masking effect of the relatively large mass of stainless steel in the vicinity of the thermocouple. A sample in an extrusion vessel was also cooled with nitrogen gas at ambient temperature. The cooling time was approximately three times longer than for the sample cooled with liquid nitrogen. These samples will be used to study segregation of cesium in sodium samples as a function of cooling rate.

Modifications to the sample cup and heater for the vacuum distillation samples are complete. Pirani-type vacuum gauges were received. Initial installation of the distillation sampler will be in the secondary system, to provide experience and to identify the most important operating variables under nonradioactive conditions.

(xi) Chemical Analyses of Alloy Samples from the Rotating Plug Seal Troughs. Chemical analyses for tin, bismuth, and sodium in the alloy samples described in the June 1967 report (ANL-7349, p. 45) are tabulated in Table XII. The nominal eutectic alloy composition is 58 w/o Bi, 42 w/o Sn.

TABLE XII. Analyses of Rotating Plug Seal Alloy

Plug Orientation	Distance from Trough Bottom (in.)	Analyses		
		Sn (w/o)	Bi (w/o)	Na (ppm)
<u>Alloy Samples from LP Seal Trough</u>				
Operate (0°)	2	42.9	58.4	2750
Operate	7	42.8	58.5	2480
120°	2	40.9	58.9	3460
120°	7	47.8	50.7	1020
240°	2	40.9	60.0	2690
240°	7	49.1	51.7	1130
<u>Alloy Samples from SP Seal Trough</u>				
Operate (0°)	2	43.4	58.6	67.9
Operate	7	43.5	58.2	60.7
120°	2	41.7	58.8	57.9
120°	7	42.0	57.3	75.2
240°	2	41.3	57.2	122
240°	7	41.9	56.2	56

These results are to be compared with the analyses of samples taken from the seal troughs on December 6, 1966, shown in Table XIII. Extensive cleaning operations had been performed on the seal

troughs in September and October 1966 and new makeup alloy had been added in October. It will be noted that the sodium content of the most recent large trough samples is of about the same magnitude as the content in the December 1966 samples, whereas the sodium contents of the small trough samples are much lower than previously. There is no ready explanation for this difference.

TABLE XIII. December 6, 1966 Analyses of Rotating Plug Seal Alloy

Plug Orientation	Distance from Trough Bottom (in.)	Analyses	
		Bi (w/o)	Na (ppm)
<u>Alloy Samples from LP Seal Trough</u>			
Operate (0°)	2	59.0	2500
Operate	7	59.2	2400
<u>Alloy Samples from SP Seal Trough</u>			
Operate (0°)	2	56.5	820
Operate	7	58.1	860

(xii) FERD Loop Sodium Monitoring Station. To provide an improved method for monitoring the quality of primary sodium, a new sodium plugging indicator and sodium-bypass freeze sampler have been proposed for installation in the EBR-II Fuel Element Rupture Detection (FERD) Loop (see Progress Report for April 1967, ANL-7329, pp. 9-10).

A small electromagnetic sodium pump has been selected for use in the in-line sodium sampler. One pump has been shipped to the reactor site and two more are being fabricated as spares.

(xiii) Improved Vacuum-distillation Equipment. To provide improved sodium sampling at EBR-II, a modified sample cup and holder for sampling primary sodium has been designed and fabricated to replace the old nickel-foil cup design. The new removable cup is rigid and made of pure nickel. It attaches to a holder by the bayonet-lock method. An improved heater and thermocouple arrangement also was incorporated in the cup holder. Test of the new device indicate that the thermal contact between the cup and holder requires improvement before field tests begin.

b. Water Treatment Monitoring and Control

(i) Treatment Data (Condensate, Feedwater, and Blowdown).
Flow in the power cycle systems occurred for only a limited time during

the month. The steam generator was in hot standby most of the time, with intermittent operation of a feedwater pump to maintain the water level in the steam drum. The feedwater system was secured the remainder of the time.

Both blowdown demineralizers were regenerated during the month.

(ii) Condenser Cooling Water. Manual control of acid addition at the cooling tower for pH control was used during the month.

(iii) Chromate Reduction in Cooling Water Blowdown. A 35-gpm stream of condenser cooling water was treated with SO_2 to reduce the CrO_4 and the effluent was sent to waste. Essentially total reduction of the chromate was achieved with a SO_2 usage of 4.5 to 5 lb/lb of CrO_4 reduced. The system was secured late in the month when the cooling-water-circulation pumps were secured.

c. Carburization Potential Measurements. The proposed location and space requirement for the carburization potential glovebox and appurtenances in the FCF have been fixed and procurement of equipment has been initiated.

Sodium for carburization potential measurements will be sampled initially by the "cup-and-spigot" method. The sample will be drawn directly into an alumina exposure vessel which has been precleaned as necessary and in which the stainless steel "tabs" have been placed. This vessel will be sent to FCF for further sample preparation and performance of the test.

It is believed that the sampling method can be refined and improved to permit taking samples by a "flow-through" method. Consideration is being given to a flow-through sample vessel which would contain the tabs during sampling and be used subsequently as the exposure vessel.

14. Experimental Irradiation and Testing

a. Status of Experiments in EBR-II. The status of all experimental irradiations in EBR-II as of the end of August is the same as shown in Table IX of the July Progress Report, ANL-7357, p. 46.

b. Examination of Irradiated Capsules from Subassembly XO11. Experimental subassembly XO11 containing the failed experimental capsule(s) was transferred from the storage basket to the FCF and disassembled. Visual inspection of the 19 capsules in the FCF Air Cell did not disclose which of the capsules had failed.

To preclude an unfavorable or destructive reaction between water and the potentially exposed sodium and fuel of the failed capsule(s), the normal washing procedure in the Interbuilding Coffin (IBC) was not carried out. The normal procedure of oxidizing the adhering sodium in the subassembly by exposure to moist argon in the IBC was carried out. Therefore, white reaction products which covered an estimated 10 percent of the capsule surfaces remained after the oxidation step. The remaining 90 percent of the surfaces appeared clean. The white deposits were largely removed by swabbing each capsule with individual water-moistened sponges. At the time, after each capsule had been swabbed, the surface appeared to be free of white deposits, but where the white deposits had been dark stains were visible. Three days after swabbing, visual inspection gave no indication of capsule failure; however, about half the capsules showed small, white, smear-like areas at random positions on the surfaces. These areas did not appear to be sodium reaction products issuing from a crack or pin hole in the capsule wall, but appeared to remain from incomplete swabbing.

The capsules are being subjected to neutron radiography in the TREAT facility. The activities on the pieces of sponge used for wiping the capsules were examined by gamma spectrometry to identify the radionuclides present, on the premise that the failed capsule(s) might exhibit significant fission product contamination. The data are inconclusive. The wipings from capsules HOV-4 (ANL) and F4K (GE) showed traces of cesium-137. Other nuclides identified on all capsules are manganese-54, cobalt-58, cobalt-60, chromium-51, and tantalum-182. The first four are stainless steel activation products. The tantalum activity is known to be present in the Air Cell as a consequence of prior handling of irradiated tantalum-clad sources.

c. Procurement and Inventory Control of Experimental Hardware. The ANL Mark-II experimental irradiation capsules received during July were physically inspected along with the GE F3B series. All were in good condition.

d. Other Work. The prototype Mark B-37A irradiation subassembly, designed to accommodate 0.250-in.-OD capsules, was flow-calibrated for Rows 1 through 5 in the flow test loop.

Temperature distributions in Rows 1 through 6 for Run No. 17 were calculated by the HECTIC II code. These temperature data were tabulated for the driver-fuel elements used in the surveillance program and will be used in an attempt to correlate fuel swelling with operating temperature.

e. Nondestructive Testing

(i) Antimony-Beryllium Neutron Source. Because of delays in reactor operation, the antimony oxide gamma source in EBR-II will not be sufficiently activated for testing of the neutron source in September. As an alternative, one gamma source will be scheduled for activation in MTR.

Four additional antimony oxide source containers have been fabricated and await filling. Activation will be in EBR-II or in MTR.

Drawings of the foil-positioning fixture and other hardware for installation of the neutron source in FCF have been completed and are being reviewed.

A criticality review for installing the Sb-Be source in the FCF was completed by the ID Criticality Representative. The source presents no additional criticality hazard so long as the beryllium remains in the cubical source geometry.

(ii) TREAT Neutron Radiography Facility. The gamma shielding for the TREAT neutron radiography facility was completed and installed. A complete radiation survey was made utilizing a 500-R/hr antimony gamma source. The shield was determined to be satisfactory for irradiation fuel. Neutron measurements external to the shielding indicated a high background, necessitating the design of a new insert plug for TREAT. This high background limits the time during which personnel may be in the vicinity of the shield, but does not significantly affect the neutron radiography of specimens. Dummy capsule runs, utilizing a capsule holder which holds two irradiation 40-in. capsules and permits rotation of the capsules for different rotational views, were completely satisfactory.

Two irradiated PNL cermet capsules from XO11, designated 5P-9 and 5P-12, were transferred from the FCF Air Cell to the TREAT facility for radiography. The images obtained were of excellent quality with good resolution and clearly indicate the sodium bond level in the capsule (see Fig. 7). Radiography of other capsules from XO11 is in progress for determining the conditions of the contained fuel elements.

Fig. 7. TREAT Neutron Radiographs of Capsules 5P-9 and 5P-12 from Experimental Subassembly XO11

(iii) Other Nondestructive Tests.

Fabrication of the tubing conveyor to be used for eddy-current testing of the capsule tubing stock has been completed, and delivered for installation in the ITF building.

X radiography of the ANL M-3 series of experimental capsules has started. Bond voids were evident in several of the capsules. In order to determine whether these were bubbles or shrinkage voids, a selected capsule was heated to 450°F and slowly cooled from the bottom. Some improvement was noted when reradiographed cold. X rays taken with the capsule hot provided little information on the existence of voids. A study is being conducted to demonstrate the best combination of energy, film, and exposure for the identification of voids in the bond sodium.

f. Handling and Examination

(i) Materials Surveillance

Subassembly. Work is continuing on the metallography of the samples from SURV-I, the designation of the first EBR-II materials surveillance assembly. The measurement of the dimensions of the three stainless steel-clad graphite cans is expected to be completed in early September. Several parts of SURV-I were sent to ANL-Illinois in a recent shipment. The final shipment of material from SURV-I, including the graphite cans, is scheduled for early September.

(a) Measurements of the Inconel-X Springs from SURV-I.

The 18 Inconel-X coil springs from the first EBR-II materials surveillance subassembly were cleaned, and the loads exerted by the springs at several degrees of compression were measured. From these data the preload on the spring and the spring rate of each spring were calculated by performing a least-square analysis on the load data. The slope of the line that resulted from the least-squares analysis of load versus the compression distance for each spring represents the spring rate. The load at zero compression, also taken from the line, represents the preload of the spring.

The measured and calculated data for the springs at the time they were loaded are given in Table XIV. The measured and calculated data for the springs after irradiation are given in Table XV.

(b) Weight and Hardness of Selected Samples from

SURV-I. Hardness samples from SURV-I were removed from the sample

TABLE XIV. Measurements with Inconel-X Springs from SURV-I
(Set No. 1 before Irradiation)

Sample No.	Load (lb) at Compression Shown (in.)							Calculated Preload (lb)	Calculated Spring Rate (lb/in.)
	0.050	0.100	0.150	0.200	0.300	0.400	0.500		
1Y1		3.35		4.30	5.15	6.00	6.85	2.52	8.70
1Y2		3.45		4.35	5.20	6.05	6.90	2.61	8.60
1Y3		3.45		4.35	5.15	6.05	6.95	2.58	8.70
1Y11		3.35		4.30	5.10	6.00	6.85	2.51	8.70
1Y12		3.35		4.30	5.10	6.00	6.80	2.53	8.60
1Y13		3.35		4.30	5.10	6.00	6.85	2.51	8.70
2Y4		5.85		6.75	7.60	8.45		5.00	8.65
2Y5		5.85		6.75	7.65	8.45		5.00	8.70
2Y6		5.70		6.55	7.45	8.25		4.85	8.55
2Y14		5.50		6.40	7.35	8.10		4.65	8.75
2Y15		5.40		6.30	7.15	8.00		4.55	8.65
2Y16		5.40		6.30	7.20	8.05		4.52	8.85
3Y7	8.70	9.10	9.70					8.17	10.00
3Y8	8.75	9.15	9.60					8.32	8.50
3Y9	8.45	8.90	9.35					7.99	9.00
3Y17	8.60	9.05	9.35					8.25	7.50
3Y18	8.65	9.05	9.40					8.28	7.50
3Y19	8.65	9.10						8.20	9.00

TABLE XV. Measurements with Inconel-X Springs from SURV-I
(Set No. 1 after Irradiation)

Sample No.	Load (lb) at Compression Shown (in.)							Calculated Preload (lb)	Calculated Spring Rate (lb/in.)
	0.050	0.100	0.150	0.200	0.300	0.400	0.500		
1Y1		2.60		3.70	4.40	5.60	6.20	1.77	9.10
1Y2		2.80		3.70	4.65	5.45	6.35	1.94	8.85
1Y3		2.85		3.70	4.70	5.55	6.50	1.92	9.15
1Y11		2.90		3.65	4.75	5.55	6.40	1.98	8.90
1Y12		2.50		3.45	4.40	5.40	6.05	1.64	9.05
1Y13		2.75		3.60	4.60	5.40	6.30	1.86	8.90
2Y4		5.00		6.15	7.00	7.65	-	4.25	8.80
2Y5		4.95		5.50	6.70	7.45	8.60	3.86	9.25
2Y6		4.70		5.45	6.35	7.35	-	3.75	8.85
2Y14		4.35		5.20	6.15	7.00	7.65	3.55	8.40
2Y15		4.55		5.35	6.30	7.10	8.10	3.62	8.85
2Y16		4.25		5.15	6.00	6.80	7.70	3.42	8.55
3Y7	6.90	7.45	8.45	9.75				5.60	10.40
3Y8	6.15	6.65	7.05	7.75				5.75	9.10
3Y9	6.50	7.10	7.35	7.80				6.15	8.30
3Y17	6.95	7.25	7.75	8.25				6.45	8.80
3Y18	5.75	6.35	6.90	7.30				5.28	10.40
3Y19	6.70	7.40	8.00	8.30				6.25	10.80

tubes. Most of the samples that were exposed to the primary sodium during the irradiation in EBR-II were cleaned and weighed. Although the subassembly was soaked in alcohol for several days, not all of the hardness samples were clean when they were removed from the sample tubes. Therefore, the samples were additionally cleaned in an ultrasonic tank employing a 10 percent solution of "Radiac" in water.

The hardness readings were made on a hardness tester calibrated to ± 1 number on the Rockwell A scale using standard blocks.

The preirradiation weight, postirradiation weight, and weight change for the samples are listed in Table XVI.

TABLE XVI. Weight and Hardness Measurements of Samples Exposed to Sodium in SURV-I

Material	Sample No.	Preirradiation Weight (g)	Postirradiation Weight (g)	Weight Loss (g)	Postirradiation Hardness (Rockwell A)
Aluminum Bronze	A-1	16.0706	16.0682	0.0024	59.0
	A-2	16.0466	16.0452	0.0014	
	A-3	16.0524	16.0498	0.0026	
	A-4	16.0663	16.0658	0.0005	
Stellite	B-1	18.0920	18.0904	0.0016	
	B-2	18.0977	18.0965	0.0012	
	B-3	18.0898	18.0881	0.0017	
	B-4	18.0911	18.0900	0.0011	
Inconel-X	C-1	17.7732	17.7720	0.0012	70.5
	C-2	17.7677	17.7675	0.0002	
	C-3	17.7705	17.7693	0.0012	
	C-4	17.6373	17.6340	0.0033	
420 Stainless Steel	D-1	16.5114	16.5088	0.0026	71.0 70.7 70.5
	D-2	16.4992	16.4980	0.0012	
	D-3	16.4831	16.4814	0.0017	
	D-4	16.5058	16.5042	0.0016	
Tool Steel T-1	E-1	18.6850	18.6761	0.0089	80.0 78.8 78.2
	E-2	18.7076	18.7070	0.0006	
	E-3	18.6791	18.6744	0.0047	
	E-4	18.6900	18.6876	0.0024	
347 Stainless Steel	F-1				49.2 52.0 52.0
	F-2	16.9514	16.9514	0.0000	
	F-3	16.9165	16.9150	0.0015	
	F-4	16.9323	16.9306	0.0017	
416 Stainless Steel	G-1	16.3335	16.3307	0.0028	67.1 67.4
	G-2	16.3135	16.3115	0.0020	
	G-3	16.2729	16.2700	0.0029	
	G-4	16.2481	16.2465	0.0016	
Berylco 25	H-1	17.9802	17.9768	0.0034	55.1 55.0
	H-2	17.9804	17.8542	0.1262	
	H-3	17.9834	17.8073	0.1770	
	H-4	17.9538	17.8033	0.1505	
304 Stainless Steel with Boron	I-1	16.8304	16.8263	0.0041	60.6 59.7
	I-2	16.7883	16.7841	0.0042	
	I-3	16.7070	16.7050	0.0020	
	I-4	16.8113	16.8091	0.0022	

TABLE XVI (Contd.)

Material	Sample No.	Preirradiation Weight (g)	Postirradiation Weight (g)	Weight Loss (g)	Postirradiation Hardness (Rockwell A)
17-4 PH Stainless Steel	J-1	16.6924	16.6917	0.0007	73.0
	J-2	16.6975	16.6960	0.0015	72.0
	J-3	16.7120	16.7108	0.0012	72.0
	J-4	16.6692	16.6683	0.0009	72.0
304 Stainless Steel	K-1	16.8403	16.8403	0.0000	49.0
	K-2				
	K-3	16.8593	16.8570	0.0020	51.4
	K-4	16.8571	16.8550	0.0021	51.2
Tantalum	M-1	36.2064	36.2026	0.0038	
	M-2				
	M-3	36.1974	36.1750	0.0224	
	M-4				

(ii) Testing of Procedures for Removal of Sodium from a Subassembly

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

The metallography program to be performed by GE on the capsule tubes from XG06 has been started. Initial microphotos show the capsule tubing (F2Q) to be in good condition and comparable to microphotos of the same type of tubing in the as-received condition with one exception: the grain structure from XG06 tubing after irradiation showed effects similar to those of heat treatment. A shipping cask has been sent to GE for transport of some of the XG06 capsule specimens to EBR-II for metallographic examination.

(b) Effect of Recycling Types 304 and 304L Stainless Steel Tubes. To evaluate the effect of storing tubing of Types 304 and 304L stainless steel in air after exposure to sodium, which is important for determining whether irradiated test fuel elements can be returned to the reactor after being nondestructively examined, a three-phase test program has been conducted.

In the Phase-III tests just completed, Type 304L specimens (that had been exposed to 1100°F sodium twice for 7-day periods with an intermediate 10-day storage in air) were stored in normal or in dry air for 10 days and then were exposed for a third time to 1100°F sodium for 7 days, removed, cleaned, and tested as before. Figures 8 and 9 show the Phase-III data in comparison with those from the previous phases; the specimens reported in Fig. 8 were stored in normal Illinois air, whereas those in Fig. 9 were stored in CaSO₄-desiccated-air containers to simulate the dry Idaho atmosphere at the EBR-II site. Photomicrographs confirm the resistivity data.

Fig. 8. Corrosive Attack of Type 304L Stainless Steel Tubing That Had Been Stored in NORMAL Air for 10 days between Exposures to 1100°F Sodium in a Loop for 7-day Periods, as Indicated by Changes in Electrical Resistivity during Subsequent Exposure to Strauss Reagent. Data for as-received tubing and for tubing that was not stored after a single exposure to sodium are shown for comparison.

Fig. 9. Corrosive Attack of Type 304L Stainless Steel Tubing That Had Been Stored in DRY Air for 10 days between Exposures to 1100°F Sodium in a Loop for 7-day Periods, as Indicated by Changes in Electrical Resistivity during Subsequent Exposure to Strauss Reagent. Data for as-received tubing and for tubing that was not stored after a single exposure to sodium are shown for comparison.

Noteworthy are the observations that both repeated cycling and storage in dry air lead to considerably faster corrosion. In general, the work completed confirms previous conclusions (see Progress Report for June 1967, ANL-7349, p. 22). Sodium-exposed specimens stored in dry air probably contain, initially during the period of storage in air, the same quantity of subsurface sodium as specimens stored in moist air. Since only a limited amount of moisture is available within the desiccator and the dry-air inventory is not replenished, more unreacted sodium within the subsurface grain boundaries is potentially available to react aggressively with any aqueous solution to which the specimen is exposed after storage. On the other hand, the sodium within the subsurface grain boundaries of a moist-air-stored specimen is available to react with moisture in the air over the entire period of air storage; less total sodium is then available to react with an aqueous solution, and there results less change in microstructure and electrical resistivity.

Although it originally had been planned to expose specimens to a fourth cycle of sodium and air storage, the three cycles have deteriorated the specimens so much that further cycling is impractical. Photomicrographs of the three-cycled specimen (in Fig. 9) that measured 150 microhm-cm after only 1 hr in Strauss solution confirm that such specimens have deteriorated grossly. Currently, the mechanical properties of the three-cycled specimens are being measured.

(c) Cyclical Testing of Stressed Capsules (Type 304L Stainless Steel) at EBR-II. This work involves metallographic comparisons of as-received Type 304L stainless tubing with samples from stressed (pressurized) capsules cycled to the EBR-II primary coolant, standard cleaning procedures, and storage conditions. The cyclic test conditions have been described previously (see Progress Report for June 1967, ANL-7349, p. 52; for July 1967, ANL-7357, p. 52).

The subassembly containing the initial 37-specimen capsules was loaded into the primary coolant and at month's end was residing in the storage basket for the initial 7- to 10-day soaking period. The subassembly has been identified as X900.

(d) EBR-II Subassembly Recycle Test. To provide further information about the behavior of irradiated stainless steel cladding which has undergone several cycles into sodium and through the standard EBR-II washing procedure, arrangements are being made to recycle a subassembly of 1.2 a/o burnup through five cycles of sodium exposure in the storage basket and washing in the IBC, using standard procedures. Inspection of cladding specimens is planned after the fifth cycles.

(iii) Demonstration of Integrity of Sodium Bond. Reirradiation of experimental capsules in EBR-II after interim examination will require

the maintenance of an element-to-capsule sodium bond of integrity equal to that for the original irradiation. Provisions have been made to assure that a vertical position of each capsule is maintained during all ex-reactor handling procedures. However, since preferential cooling of the capsules from the bottom cannot necessarily be assured, shrinkage voids could form in the bond sodium. These voids could under certain conditions become filled with the capsule cover gas, usually helium or argon. These voids then might become gas bubbles after melting of the bond sodium.

To ascertain whether such a shrinkage-void mechanism of bond deterioration might be significant, a test program is under way with a typical capsule. The cooling, or freezing, cycle is controlled to impose the greatest number and size of shrinkage voids. Radiography is used both in the frozen and the molten conditions for the detection of bond voids or bubbles. To date, there has been no indication of bubble formation after preferential freezing from the top and remelting. Following completion of the program a summary report will be issued.

(iv) Inspection of Subassembly B-341. Sixth-row subassembly B-341, with which difficulty was experienced in fuel-handling operations on July 27 (see II.A.6.a), was removed from the transfer arm and transported to the FCF Air Cell via FUM and IBC. It was received in the Air Cell on July 31. On August 1, the subassembly was placed in the Vertical Assembler-Disassembler machine (VAD) for inspection of the upper pole piece.

Visual inspection showed that the upper pole piece was bent and twisted from the normal position. The slot in the pole piece was burred and dented. The marks indicated that a clockwise torque had been applied to the upper pole piece. The complete pole piece (round and rectangular sections) had been twisted with respect to the hex tubing of the subassembly. Measurements showed that the slot in the upper pole piece had been twisted 35° clockwise with respect to the rectangular section.

The following additional measurements and observations were made:

(a) The tightness of the threaded joint between the upper pole piece and the lower part of the top adapter was tested. Very little resistance was encountered as the pole piece was unscrewed, indicating that the pin through this joint had been sheared.

(b) A straightness measurement was made in VAD. A maximum total indicated runout of 45 mils was measured over the length of the hex tube with the lower subassembly pole piece and the top of the hex tube supported. The top of the upper pole piece showed a total indicated runout of 275 mils.

(c) The diameters at the "buttons" on the hex can were shown by micrometer measurements to be within specifications (2.317-2.319 in.).

(d) A maximum twist of 1° is permitted on the hex can. The torsional deflection of three flats of the tube were measured and found to be within this specification.

15. FCF Equipment Improvements

a. Processing Equipment. The vertical assembly and dismantling machine (VAD) is in operation in the Air Cell of the FCF.

Design of an improved straightness and tensile-testing machine has been completed. This machine would replace the two now employed in the Air Cell for final inspection of refabricated subassemblies.

Testing of the second skull-oxidation furnace was completed, and the unit is ready for installation in the Argon Cell.

The final drawings for the pneumatic transfer system, including comments by vendors of pneumatic tube systems, have been prepared. Purchase orders are to be let during the week of August 28, 1967. The project is now about one month behind schedule, having been delayed by projects of higher priority. Completion is now forecast for the end of FY 1968.

Checking of drawings for the neutron radiograph components has been completed. Final comments and revisions are now being incorporated. Fabrication is expected to commence by September 15, 1967. Based on present evaluation, this project should be completed during FY 1968.

b. Auxiliary Equipment. Cost estimates for the interim storage facility, external to the existing cells and providing for a 3-month holdup capacity (see Progress Report for July 1967, ANL-7357, p. 53), are being analyzed.

A shielded cask employing bottom loading and discharging of a subassembly will be used (1) to transport subassemblies to and from the new storage facility, (2) to charge and remove a subassembly from a storage pit, and (3) as an alternative 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.

A feasibility study is being conducted to determine if the Argon Cell gas can be decontaminated by filtration prior to passing it through the duct work of the cell-cooling system. Such decontamination is necessary if the radiation and contamination problem in the subcells is to be relieved. Based on earlier tests, the study is continuing with the use of American Air Filter Company's³ Dri-Pak Filter Cartridge with number 100 filter media to (1) determine the minimum permissible gas flow for each cooling box; (2) ascertain if the existing circulating fans can be used in conjunction with a filtering system and provide adequate flow; (3) find the minimum filter area (if filters are to be installed in the cell, a minimum area is necessary to keep from taking up excessive usable cell space); (4) determine the lifetime expectancy of the filter system (if a filter system is to be effective, it should have a long life between changes as well as a high efficiency).

One of the rebuilt, surplus, argon-circulating fans was mocked up in a simulated duct and operated with various blade settings. The pressure drop across the fan was adjusted with variable orifices to simulate in-cell conditions. A filtering unit was installed at the inlet of the fan and operated with different filtering areas. Fan capacity and power requirements were recorded for each run. All data were normalized to in-cell operating conditions.

The lifetime expectancy of the proposed system was checked by mounting a representative filter sample on a special test pump which was operated in the Argon Cell. Rate of increase in differential pressure across the filter was recorded to determine the rate of loading the filter. Since horsepower requirements are proportional to pressure drop for a given flow rate, filter loading becomes a limiting condition.

The minimum permissible gas flow rate through each cooling box (based on a minimum permissible heat-removal rate of 350,000 Btu/hr with inlet argon at 90°F and a coil temperature of 28°F) was calculated to be approximately 11,800 cfm at cell conditions.

The results indicate that the existing fan units can be used in conjunction with an in-cell filtering system and not exceed the 40-hp rating of the fan motors, provided the proper adjustments are made. A filtering area of 20 sq ft can be used without adversely affecting the horsepower requirements of the fan. (Such an area would enable all of the filters to be placed under a work table in the Argon Cell.) With 20 sq ft of filter area, at least 1,000 hr of operation could be expected before a filter change would be necessary.

Combination of results of fan flow rate with filter tests indicate that, after 2,000 hr of operation with a filter, there would be a 10% increase in pressure drop and horsepower requirements, and a 10% decrease in flow as compared to existing conditions without a filter.

³American Air Filter Co., Inc., Louisville, Kentucky.

The waste material from the EBR-II Fuel Pin Process station consists of broken Vycor glass molds as well as small bits and pieces of enriched uranium. The material cannot be disposed of as a normal waste because it contains considerable amounts of enriched uranium. In the past, the scrap material has been consolidated and shipped to ICPP for processing. The shipping process is time-consuming and requires a rather large inventory because of the special process schedule at ICPP. A program has been started to develop equipment to separate the uranium from the glass scrap in the Argon Cell.

c. Repair and Decontamination. Major Equipment drawings for the Process Equipment Decontamination and Repair Facilities are completed. Work is proceeding on the small-component drawings. The engineering is 50% complete, and the entire project is 25% complete.

Part of the detailed drawings of the Manipulator Decontamination and Repair Facility have been submitted to the FCF staff for review. The drawings are about 75% complete. Total engineering is approximately 75% complete.

16. Hot Fuel Examination Facility (HFEF)

Comments on the draft Prospectus were received from the AEC and incorporated. A revised, approved Prospectus was released on schedule to eleven interested and qualified architect-engineer firms on August 15, 1967. Proposals are due September 12, 1967. The major change in the revised Prospectus was the addition of the requirement for preparation of system-design descriptions by the A-E and the Laboratory.

The F. C. Torkelson Company, Salt Lake City, Utah was placed under contract on August 14, 1967 for consultant-engineering for the establishment of a reasonable preliminary cost estimate for submission to the AEC in November 1967. This cost estimate is to be independent of the Feasibility and Cost Study to be performed by ANL and the A-E (due February 1, 1968). September 8, 1967 was set as the date when the cell and building configurations to be used for the ensuing cost estimate will be "frozen."

The major technical effort has been that of preparation and study of various layouts and systems involving the cell(s) and associated examination equipment and material-handling devices.

A preliminary estimate was made of the lighting required in the main cell and the consequent heat load. Considering high-level and low-level work-station illuminations of 550 and 400 lumens/sq ft (in-cell), respectively, the estimated heat load is 134 kW.

Preliminary estimates are also being made of the expected leak rates of air into the main cell, based on measured leak rates in the FCF Argon Cell and on those in the Metallurgy Alpha-Gamma Cave.

17. Nuclear Instrument Test Facility Study

Representatives from ANL (Illinois and Idaho), Pacific Northwest Laboratory, and the AEC met to discuss the scope of this task. It was determined that Phase I, up to 700°F testing, would be adequate for the needs of FFTF. However, use of an extremely strong auxiliary gamma source is also desirable. This would add greatly to the cost of the EBR-II nuclear test facility because a complicated handling coffin for the source and instruments would be required.

It was deemed desirable to flux map the gamma and neutron activity in the thimble as soon as possible. Results of these measurements will be evaluated to resolve the need for vertical movement capability of the nuclear detectors during the test program.

B. Physics Development

1. ZPR-3

a. Assembly 50. Work continued on Assembly 50, the third in a series of simple plutonium-fueled assemblies currently being studied on ZPR-3. Assembly 50 has the smallest core and the softest neutron spectrum of the series; achieved through the replacement by graphite of the sodium which was loaded in Assembly 48. The rest of the core loading is the same as Assembly 48.

The uncorrected critical mass (loaded mass) of Assembly 50 was given (see Progress Report for July 1967, ANL-7357, p. 60) as 218.5 kg of $\text{Pu}^{239} + \text{Pu}^{241}$. When corrections have been made for control-rod spiking, partial insertion of one control rod, central gap, and effects of edge irregularity, the corrected critical mass (for a heterogeneous regular cylinder of uniform composition) is 225.6 kg of $\text{Pu}^{239} + \text{Pu}^{241}$.

Experiments completed this month included measurements of the central fission ratio for $\text{U}^{233}/\text{U}^{235}$, $\text{U}^{234}/\text{U}^{235}$, $\text{U}^{236}/\text{U}^{235}$, and $\text{U}^{238}/\text{U}^{235}$. These spectral indices, when compared to measurements made in other assemblies in the series, give a comparison of the neutron fluxes and reaction rates in the various cores.

Central perturbation measurements using small cylindrical and annular samples were also made. The materials measured included Pu (22% Pu^{240}), Pu (4.5% Pu^{240}), U^{235} , U^{238} , Ta, B^{10} , Mn, stainless steel, Cr, polyethylene, Ni, Fe, and iron oxide. In many cases different sample sizes were obtained by varying the thickness of the annuli or the diameter of the cylindrical samples. The size variation was made to investigate the effect of self-multiplication in the fissile samples or the effect of self-shielding in other materials.

Measurements of central reactivity were made using a 2 x 2 x 2-in. sample changer, which will move a 2 x 2 x 2-in. sample of material, or boxes of the same size with individual plates in them, to the center of the core. The experiments included standard measurements of plutonium (both in plate and cylindrical form), depleted uranium, and sodium.

In addition, measurements were made using 2-in. cubical samples formed from the normal core constituent plates, both with and without the plutonium fuel present. The latter measurements, involving removal of plates of fuel, constitute a valuable new series of data since these are amenable for analysis in infinite-plate geometry, correctly describing the heterogeneity of the reactor core.

Of special interest was a U^{235} fission rate traverse and a U^{235} worth traverse across the 2-in. void in which the perturbation measurements were made. These traverses were made to determine the shape of the flux and importance function through the void.

Analysis of the experimental results is in progress and will be reported at a later date.

b. Assembly 48A. Assembly 48A was the reactor loading for the initial set of experiments in support of the Fast Flux Test Facility (FFTF) being designed by Pacific Northwest Laboratory. Measurements of the reactivity worths of rhenium and europium which were substituted for equal volumes of enriched B_4C in drawers in the 1st row of the Inconel reflector were made by replacing an equal volume of stainless steel from the reflector drawers 1 and 2-P-9 with enriched B_4C and then by rhenium or europium. Enough rhenium was available so that a $1/4 \times 2 \times 32$ -in. column could be substituted (16 in. in each half of the reactor). Only enough europium was available for substituting a $1/4 \times 2 \times 6$ -in. column in the front of one drawer. The replacement of 32 in. of stainless steel (2023.8 g) with 32 in. of enriched B_4C (517.7 g B_4C containing 322.7 g B^{10}) caused a loss of 208.8 lh of reactivity. The replacement of 32 in. of stainless steel with 32 in. of rhenium (5236.6 g) caused the loss of only 96.5 lh of reactivity.

In replacing a $1/4 \times 2 \times 6$ -in. column of stainless steel (379.5 g) with an equal amount of enriched B_4C (97.1 g B_4C containing 60.5 g B^{10}) the reactivity loss was 60.1 lh. When the 6 in. of stainless steel was replaced with europium (245.3 g) the reactivity loss was only 30.3 lh.

c. U^{238} Foil Activations. Preparations have been made for two U^{238} activation experiments related to understanding of the Doppler effect in U^{238} .

The first has been run in Assembly 50. In this experiment, the activation (neptunium activity) was measured versus position in a sandwich of two $1/8$ -in. U^{238} plates in a relatively simple, symmetrical drawer loading.

In the second experiment, the change in activation with temperature will be measured by irradiating a foil in a furnace under hot and cold conditions, similar to the previous activation Doppler measurements (see Progress Report for February 1966, ANL-7176), but the furnace will be surrounded by a 0.35-cm-thick annular cylinder of depleted uranium to enhance the effect. A change in activation of about 15 to 20 percent is anticipated between the hot and cold irradiations.

In these experiments care is being taken to provide a simple environment which will permit reasonably accurate comparison with theory by means of the RABBLE code and the activation code being developed by the Idaho Division.

A preliminary study was made in which a sandwich of three 0.005-in. foils was irradiated in Assembly 49 for about 240 Whr and allowed to decay for three days. The gamma spectrum was then analyzed using the Ge(Li) semiconductor spectrometer and a 512-channel analyzer. One of the outside foils had an activity about 10% higher than the other two, indicating asymmetry in the flux. There was an average flux depression of about 5% in the center foil. These results indicate that the technique is capable of good accuracy in the planned experiments.

An interesting sidelight is that the plutonium K X rays due to internal conversion in the β decay of neptunium were identified in the spectrum.

2. ZPR-6

a. Oscillator Sample Changer. A new oscillator sample changer built for the purpose of measuring small perturbations was installed. With this device it was possible to measure reactivity signals with an accuracy of 0.001-0.002 lh. This device is presently being used for measuring the heterogeneity effect on the sodium-void coefficient.

In these measurements the reactivity worth of the sodium in a homogeneous sample (2 x 2 x 2 in.) is compared to its worth in a heterogeneous sample of the same size. The homogeneous samples have the same composition as the rest of the core, which was heterogeneously loaded. The worth of the sodium in the homogeneous sample was determined by observing the relative effects on the system of two samples, one with and one without sodium. The composition of the two samples was closely matched. The worth of the sodium in the heterogeneous samples was determined in a similar manner. The data from these measurements are being analyzed.

b. The Effect of Sodium Voiding upon the Spectrum of Assembly 5. The spectrum at the center of Assembly 5 has been measured with 4π hydrogen-recoil proportional counters in a cylindrical volume of the reactor both with and without sodium. Since sodium is one of the principal constituents of ZPR-6 and ZPR-9 cores, its expulsion can cause substantial reactivity changes which are pertinent to the safety analysis of the system.

The normal atom densities of all core materials are listed in Table XVII. The voiding process consists of removing all of the sodium cans from a central region of 20-cm radius and then adding a sufficient amount of steel to restore material concentrations to values listed in the table.

The absence of sodium reduced the reactivity by 25 lh. Measurements in the "normal" core were made with the reactor subcritical by 170 lh; with the sodium void, the reactivity was subcritical by -195 lh.

TABLE XVII. Atom Densities of Assembly 5 of ZPR-6

Material	Atom Density ($\times 10^{-24} \text{ cm}^3$)
Fe	0.009086
Ni	0.001148
Cr	0.002593
Na	0.009028
C	0.012986
U ²³⁸	0.010633
U ²³⁵	0.001547

In order to compare the effect of the sodium void upon the spectrum, both spectra were averaged over the energy group structure of cross-section set ANL-801 (after normalizing to equal total flux) and group by group differences were plotted. General trends of the calculated and measured results agree; in particular, the large deviations predicted at 3 keV and near 800 keV were observed.

In general, sodium voiding of the central region of Assembly 5 produces easily observed changes in the spectrum which are in qualitative agreement with calculation over the energy region above about 2 keV. The measured (and calculated) changes over this region are not large. The predicted deviations are not, however, generally within experimental error. Consequently, further examination of this discrepancy is required.

3. ZPR-9

Doppler measurements with Assembly 18 are being made with different material environments around the Doppler sample (see Progress Report for July 1967, ANL-7357, p. 65).

The following measurements have been completed:

Central fission ratios for U²³⁵, U²³⁸, and Np²³⁷ were made, with the results given in Table XVIII.

Radial traverse measurements were made using U²³⁸ and U²³⁵ fission counters and a BF₃ counter. The ratios, normalized to 1.0 at $r = 0$, as a function of radius are shown in Figs. 10 and 11. Calculated values were obtained from a one-dimensional diffusion calculation using a cross-section set generated by the MC² code for the central

core material. The discrepancies in the driven region are at least in part due to two-dimensional effects. The actual driver thickness at the position of the traverse was greater than the average driver thickness which was used in the calculations.

TABLE XVIII. Central Fission Ratios in ZPR-9 Assembly 18

Ratio Measured	Fission Ratio
U ²³⁸ /U ²³⁵	0.0232 $\pm$ 0.0007
Np ²³⁷ /U ²³⁵	0.154 $\pm$ 0.005

Fig. 10. Ratio of U^{238} Fission to U^{235} Fission as a Function of Radial Position in ZPR-9 Assembly 18

Fig. 11. Ratio of U^{238} Fission to BF_3 Counts as a Function of Radial Position in ZPR-9 Assembly 18

The worth of a 51.38-g sample of sodium at the center of the ZPR-9 Assembly 18 was measured to be 1.19 ± 0.06 lh/kg, using the Doppler oscillator. The effects of the plate orientation around the sample and of streaming down the center of the oscillator drawer were checked. The values of sodium worth were in agreement within the experimental uncertainties for two different plate orientations tested. The sodium worth was 50 percent greater when a 2-x 1-in. axial streaming path was introduced in the oscillator drawer in line with the sample. The value quoted is with core material loaded into the center of the oscillator drawer to eliminate the streaming path.

4. ZPPR

A detailed analysis of the safety factor for the reactor cell ring beam was made after receiving the data from the 28-day, laboratory-cured, concrete test cylinders. These data indicated that the in-place strength of the concrete ring beam is 4500 psi (design specifications are 3000 psi). Using these data, a calculation of the dynamic strength of the ring beam indicated that the ring beam, as constructed, has a dynamic strength 2.4 times that of the catenary cable. The static strength of the ring beam and cables are also satisfactory. Therefore, the ring beam with cracks (see Progress Report for July 1967, ANL-7357, p. 65) is acceptable.

The structural steel frame of the backup containment of the cell has been erected. About 90% of the corrugated steel shell has been placed around the structure. The sections of the shell are to be welded. Work in the cell is continuing on installation of terminal boxes and on the finishing of the interior cell walls.

The placing of the sand fill over the vault workroom area was started. The in-place fill is being sampled continually by ANL to test for

cleanliness, dryness, and gradation. In addition, the contractor is required to periodically sample and test for cleanliness, dryness, and gradation, and furnish test results to the Commission.

The contractor is reprocessing the fraction of the sand which was rejected for the cell roof. This sand has approximately 75% material in the 20 to 50 mesh range. No other new information has been obtained from the contractor during this period.

Work is continuing on the support wing. At present the reactor cell is 80% complete; the vault workroom 90% complete; and the support wing 95% complete. Overall completion is approximately 85%.

A progress meeting with DRDT, CH, and ANL was held.

The packages for the reactor installation are about 80% complete and are being reviewed. An outline of the reactor operating manual is being reviewed.

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

Personnel Shield Tractor Drives: The two units were received and are being stored.

Control Rod Drive Mounting Plates: One plate has been completed and is being inspected prior to shipment. The second plate was machined improperly and rejected. A new plate has been poured, and machining will start on September 6.

Fuel Control Rod Drawers: The vendor has stated that he has repaired the rejected drawers and has shipped them to be Scottsonized. These were shipped by the vendor without ANL inspection since ANL was not notified in time.

Boron Carbide Plate Fabrication: ANL has received 123 plates which were inspected and accepted. The vendor has approximately 20 plates ready for shipment. A total of 168 plates are necessary for the initial reactor startup.

Alpha Air Monitor: The bid packages were received on August 28 and are being reviewed.

Drawer Conveyor: The drawings are 95% complete and are being reviewed.

Fission Counter and Sample Changer: The drawings are about 80% complete.

5. ZPR Fuel and Nonfissile Materials Procurement

a. Technical Assistance

(i) Fabrication of U-Pu-Mo Alloys. The fuel alloy for ZPPR is a uranium-plutonium-molybdenum alloy containing 25 w/o fissionable plutonium ($\text{Pu}^{239} + \text{Pu}^{241}$) plus molybdenum to suppress the formation of zeta phase. The plutonium used has approximately 89% fissionable atoms, making the total plutonium content of the alloy about 28 w/o and the amount of molybdenum about 2.5 w/o.

There is an interest in repeating the zero-power reactor experiments on recycled plutonium with a fissionable atom content near 70%. This requires a total plutonium content near 36% and was thought to require an additional percent of molybdenum. The additional molybdenum, however, is not desirable from the physicists' point of view.

As part of the program to obtain information about the higher plutonium alloys, two U-Pu-Mo alloy melts were made. Melt R-418 was a U-36 w/o Pu-2.5 w/o Mo alloy and melt R-420 was a U-36 w/o Pu-3.7 w/o Mo alloy. The results were good: the surfaces were smooth and clean; the castings were within ZPPR specifications. However, the shrinkage was not as large as that of the ZPPR reference alloy. If additional casts are required, it will be desirable to rework the molds. Analytical results for the melts are given in Table XIX.

TABLE XIX. Analytical Results for U-Pu-Mo Melts

Melt No.	Major Element (w/o)		Minor Elements (wt. ppm)					
	Pu	Mo	C	O ₂	Y	Fe	Al	Ni
418	35.90	2.51	260	100	30	100	100	50
420	35.80	3.86	350	130	20	100	50	30

The castings were parted on a mill with slitting saws. No differences in machineability were apparent, a somewhat surprising result since information predicted that zeta phase would be present in the low molybdenum alloy. The behavior of these alloys during casting and machining is encouraging, but information on microstructure, heat treatment, and rates of corrosion in air is still needed. After samples were taken to yield such information (see Sect. II.B.5.a.(iii) below), sufficient material to make three 4-in.-long ZPPR core plates remained. They were machined, inspected, and jacketed as an exercise for the ZPPR Resident Inspectors now being trained.

(ii) Danger-coefficient Elements. Twelve danger-coefficient elements were produced during this reporting period. These will be used to measure the relative reactivity of specific isotopic compositions of uranium and plutonium. Each element contained a thin metal fuel plate enclosed in a stainless steel jacket, approximately 2 in. long, 2 in. wide, and either 1/4 or 1/8 in. thick. The sizes and compositions of the core plates produced are given in Table XX.

TABLE XX. Sizes and Compositions of Danger-coefficient Core Plates

Number Required	Length (in.)	Width (in.)	Thickness (in.)	Material
2	1.850 to 1.830	1.695 to 1.690	0.050 to 0.047	U ²³³
2	1.850 to 1.830	1.695 to 1.690	0.080 to 0.077	U ²³³
2	1.850 to 1.830	1.695 to 1.690	0.200 to 0.197	U ²³³
2	1.850 to 1.830	1.695 to 1.690	0.050 to 0.047	Pu(22% Pu ²⁴⁰)
2	1.850 to 1.830	1.695 to 1.690	0.080 to 0.077	Pu(22% Pu ²⁴⁰)
2	1.850 to 1.830	1.695 to 1.690	0.200 to 0.197	Pu(22% Pu ²⁴⁰)

Inasmuch as the surface dose rate for the plates containing U²³³ metal ran as high as 15 R/hr, thus constituting a handling problem, the core plates were produced by precision casting rather than rolling. A book mold with a stepped cavity was designed for casting the three thicknesses in one melt. The cavity was stepped with the thicker section on the top. This provided a larger gate and higher head on the thinner cavity on the bottom. The mold plates were made from ATJ graphite. The mold dimensions were made oversize to compensate for a 2.4% shrinkage in the uranium castings.

The same molds were used for the plutonium melts. Since the plutonium did not shrink as much as the uranium, the castings were slightly oversize and required some machining.

The jackets were manufactured by a process similar to that developed for the manufacture of ZPPR-element jackets. The technique by which the elements were assembled and loaded was that used in ZPPR element manufacture.

(iii) Properties of Zero-power U-Pu-base Alloys. As part of the fuel development program for ZPPR, air corrosion tests and metallographic studies have been started, and thermal-expansion properties have been measured on the two new U-Pu-Mo alloys described above in Sect. II.B.5.a(i).

The specimens for tests of air corrosion consist of rectangular sections approximately 0.25 x 0.25 x 0.375 in. cut from the as-cast fuel plate. These were placed in a glass dish and stored in an air-atmosphere glovebox. After a one-month exposure, both alloy samples showed no weight gain and only a very slight surface tarnishing.

The microstructure of the R418 alloy is similar to that of U-20 w/o Pu-2.5 w/o Mo alloy, but the R420 alloy with the additional molybdenum is quite different, indicating that it is probably in a different phase field of the ternary diagram. The different phases present are being analyzed.

Thermal-expansion measurements were made on pieces approximately 0.188 x 0.188 x 1.25 in. cut from the as-cast fuel plates. The coefficients of expansion are given in Table XXI.

TABLE XXI. Expansion Characteristics of U-Pu-Mo Alloys for ZPPR

Alloy Melt No.	R418	R420
Nominal Alloy Composition (a/o) (w/o)	U-34 Pu-6 Mo U-36 Pu-2.5 Mo	U-34 Pu-9 Mo U-36 Pu-3.7 Mo
Mean Coef of Expansion, $\times 10^6/^{\circ}\text{C}$ for 25 to 220 $^{\circ}\text{C}$	19.2	17.4
Mean Coef of Expansion, $\times 10^6/^{\circ}\text{C}$ for 220 to 380 $^{\circ}\text{C}$	26.6	26.2
Mean Coef of Expansion, $\times 10^6/^{\circ}\text{C}$ for 380 $^{\circ}\text{C}$ to Transformation	31.6	31.1
Transformation Range, $^{\circ}\text{C}$	545-600	535-560
Expansion, $\Delta\ell$	0.38%	0.26%
Mean Coef of Expansion, $\times 10^6/^{\circ}\text{C}$ for 600 to 700 $^{\circ}\text{C}$	25.1	24.1

b. ZPPR Fuel Procurement. Shipment was made of 950 kg of ZPPR plutonium from Richland, Wash., to Rocky Flats, Colo. Approximately 600 kg of this plutonium will be used by Dow Chemical Co. for fuel fabrication at Rocky Flats and 350 kg will be trans-shipped to NUMEC for their production. Isochem Inc. completed the analyses of the remainder of the 1330 kg of plutonium that they blended in FY 1967. Isochem has been given an additional order for 1200 kg of plutonium to be blended to 11.5 percent Pu²⁴⁰ during fiscal year 1968. This work will be assumed by the Atlantic Richfield Chemical Corp., which took over the work of Isochem Inc.

Dow Chemical Co. overcame difficulties in making sound castings of the U-Pu-Mo alloy and completed their 50 preproduction fuel elements, of which 47 passed the Dow inspection. Twenty-four of these elements have been received by Argonne National Laboratory for evaluation of the Dow process and for part of the calibration of the gamma-spectrometric technique for Pu^{239} assay. The additional 23 preproduction fuel elements and the remainder of the calibration elements will be shipped early in September. Except for mold design only minor changes were needed in the Dow procedures to manufacture the ZPPR fuel. Argonne National Laboratory is evaluating the elements received from Dow for conformity to ZPPR fuel specifications.

Nuclear Materials and Equipment Corp. completed installation work and first-generation ZPPR tooling. NUMEC started to demonstrate their proposed process and have successfully cast 15-in.-long bars from the U-Pu-Mo alloy. Modifications were found to be necessary in the method that they proposed to cut core plates from the cast bars, and further development is needed of assembly, welding, and inspection procedures. Preliminary outlines of analysis, manufacturing process, and quality-control procedures were submitted by NUMEC to Argonne National Laboratory for comment.

Over 1500 sets of jacket components, 100 each of 5 sizes, and 500 additional sets of the 5- and 7-in. sizes were fabricated at Argonne National Laboratory. The stretch-forming method of making rectangular-section jacket sleeves from tubing worked well. Burrs and distortion from the jacket sleeve cut-off operation required corrective rework which added to jacket costs. Shearing dies were designed to eliminate the burrs and distortion. The end-plug manufacturing operation was retooled to obtain a better fit of the plugs in the jacket sleeves.

C. Component 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.

1. Sodium Quality Measurement

Both laboratory and in-line methods are being developed for determining the content and form of various impurities, particularly of oxygen, carbon, and hydrogen.

a. Laboratory Techniques. This 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.

Facilities for determining oxygen in alkali metals by the amalgamation method have been set up and are in use. The first series of analyses aimed at determining the precision obtainable by this method. Analyses conducted on samples cut from a single frozen bypass sodium line yielded a value of 1.2 ppm for the average deviation. This now broadens our capability for correlating in-line oxygen measurements with oxygen determinations made by the amalgamation method as well as by the vacuum-distillation technique.

Both methods depend on the separation of oxygen (as sodium oxide) from the bulk sodium sample. If any sodium carbonate is present in the sodium, it will also be separated by these methods and will appear along with the sodium oxide residue. Unless accounted for, the carbonate will be interpreted as Na_2O , and hence a higher oxygen indication will be obtained than actually exists in the sample. Experiments are in progress to determine the sodium carbonate concentrations of the residues.

b. In-line Techniques. 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 dip-type vacuum-distillation samplers.

Temperature-control problems that were encountered with the Sodium Analytical Loop have been greatly reduced after locating and replacing a faulty valve in the cold-trap cooling-system line. Stabilization of the room-temperature air conditions in the loop area also has helped to alleviate the earlier problem.

Several dip samples of the loop sodium have been taken and distillations performed to check the sampling technique, vacuum system, and high-frequency heating system. During checkout a porous copper tubing elbow in the vacuum system was located and replaced.

Repeated plugging-indicator runs are being made to determine optimum operating conditions (cooling rate and flow rate). At present, typical plugging-indicator measurements give values of 9.3, 10.0, and 11.2 ppm for the oxygen concentration in the loop sodium. Measurements made on a dip sample followed by vacuum distillation yielded a value of 8.2 ppm oxygen.

The loop line sampler, which permits the withdrawal of a sample for distillation from a dynamic or flowing sodium system, has been completed. Initial tests have begun to check the mechanical or operational features of the device, the vacuum system, and the ability to repeatedly withdraw uniform-sized samples.

2. 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 getters. 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 diffusion-type cold trap that can be submerged in the large primary tank is being developed.

A method to determine gross amounts of sodium oxide in sodium is being investigated. The method would be suitable for postoperation examinations of cold traps to learn the distribution of sodium oxides in the trap. The method involves reacting the Na-Na₂O mixture with water and measuring the hydrogen evolved to determine the ratio of sodium to sodium oxide. The method would be particularly suitable for cold traps designed to be dismantled for such examinations.

3. Reactor Mechanism and Instrumentation

a. Fuel-pin Thermocouple. Thermocouples are being developed to indicate fuel-pin temperatures up to 2800°C. 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 Insulator. The electrical resistivity of such various forms of high-temperature insulators as thoria, alumina, beryllia and hafnia are being measured up to temperatures of ~2800°C to determine their potential usefulness; mechanisms of failure are being investigated.

A third thoria resistivity probe, identified as TR-3 and with the same construction as TR-1 and TR-2 (see Progress Report for June 1967, ANL-7349, pp. 70-71), was tested. TR-3 yielded essentially the same resistivity values as obtained from TR-2. Thus the evidence indicates that the resistivity of 99.9%-pure vitrified thoria is defined by the TR-2 curve given previously (see Fig. 7 of Progress Report for July 1967, ANL-7357, p. 73). At least one more thoria probe of this type is being tested to acquire more temperature-resistivity data and to investigate possible freezing-in of defects (see ANL-7357, p. 73).

Beryllia and hafnia insulators have been ordered. Delivery is expected in October.

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

Several different types of bare-wire W-3% Re/W-25% Re thermocouples were constructed after the mechanical failure of BTC-1 (see ANL-7357, p. 73). Each modified thermocouple mechanically failed within 24 hr of testing at 2500°C.

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

A second high-temperature furnace is being altered to test materials-compatibility samples.

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, eddy current, or other types of flowmeters can be obtained for use in severe temperature and radiation environments.

(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.

The thermal testing of the Alnico alloys is continuing. The irreversible and material effects for Alnico-5, -6 and -8 have been determined for temperatures up to 1300°F and for length-to-diameter (L/D) ratios of 8, 4, and 2.

The magnetometer has been completely checked and calibrated; a reproducibility of less than 0.1% can be expected for magnets having 1000 cgs units of moment (typical of the test magnets). Comparison of moment measurements made in a magnetically clean area remote from laboratory buildings (an open field) and in a steel cave mockup indicated in the latter an error of about -0.2% in the measured value of the magnetic moment.

The clean-area measurements also helped to explain the previously reported anomaly between calculated and measured "too-close" errors (see Progress Report for July 1967, ANL-7357, p. 74). The clean-area measurements indicated that the magnetometer averages the axial

field over a considerably shorter distance than the actual sensor length, thus explaining the differences. Based on the above information, corrections up to 1.4% must be applied to some of the magnet-moment measurements to obtain the greatest accuracy in absolute magnet strengths.

The fabrication of the ceramic furnace pieces is now complete, and the furnace assembly should be finished very soon, at which time measurements of reversible temperature effects on the Alnico alloys will begin. Assembly of the B-H tester is continuing.

(ii) Magnetic-flowmeter Prototype. Prototype magnetic flowmeters are being designed, fabricated, and tested at sodium temperatures up to 1200°F to determine calibration sensitivity, impedance, and field-noise signal levels; temperature compensation schemes will be developed if needed.

The calibration loop has been repaired and is being checked out at 400°F.

(iii) 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.

Work is continuing on the experiment description and hazard analysis. Data for the new gamma-heating rates and neutron-flux distributions in EBR-II are being obtained.

The hot-cell mockup is complete, and magnet-handling procedures are being studied. These handling techniques will be used to evaluate the total capability of the hot-cell test system to determine radiation effects of simulated test samples over a range of temperatures up to 1300°F.

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, and will be tested out-of-pile and in-pile.

(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.

Attempts are still underway to assemble successfully and fill a 1/8-in.-OD subminiature NaK-filled bellows system for pressure transduction. This system will use a 10-ft-long, 0.012-in.-ID capillary tube with a total length of 20 ft.

A 1/8-in.-OD miniature NaK-filled bellows system was actually assembled and filled. This entire assembly had been evacuated at both ends by a vacuum system in preparation for filling. Then NaK was forced into the capillary tube and allowed to flow for a short time out the end of the bellows through a tube. Later this tube was pinched to seal the system.

When gas pressure was applied to the outside of the bellows, no signal was observed from the transducer. Postmortem examination revealed that the NaK fill was not complete.

It is believed that before filling the capillary the entire system should be "baked out" while at vacuum conditions; the NaK then should be allowed to flow through the tube for a longer time than the previous filling. There is suspicion that gas had been trapped in the bellows and did not get out because NaK flow was not carried on long enough.

d. Signal-lead Connectors for Sodium Service. To develop remote connectors for electrical and fluid signal leads for in-core instrumentation, connector lead-wire assembly designs and test procedures will be reviewed, changes made to meet FFTF design criteria, new connector concepts considered, out-of-pile test programs established, and methods for terminating the sensor leads inside the reactor vessel, both above and within the sodium, will be studied and developed.

Work is in progress on a state-of-the-art report. A preliminary design for a test prototype FFTF connector has been initiated; it includes all the required signal leads, including two connections for capillary tubes, one 3/8-in.-ID sodium sample tube, and a 36-pin electrical connector. There is a drain line to remove sodium that might leak from the sample line. Space has been left for a flux-wire duct, the details of which have not yet been determined. The design also includes means for coupling mechanically the upper and lower portions of an entire 35-ft-long FFTF fuel-channel assembly.

e. Failed-fuel Locating Method. To find a method for determining which subassembly within a shutdown reactor core has one or more failed fuel pins, the feasibility of sensing the fission-gas release from the subassembly is being investigated.

Calibration tests are still in progress on equipment to allow very small helium leakrates to be injected into the sodium tank used in the mocked-up fission-gas detection system.

D. Fuel Development

1. Metallic Fuels

a. Performance of Uranium-Plutonium Ternary Alloys. Two fuel elements, of composition 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 remain in EBR-II in subassembly XG05. No significant burnup has been accumulated during the past month. About 70% of the target burnup has been achieved. The status of the irradiations is the same as in Table XVIII, Progress Report for July 1967, ANL-7357, p. 82.

Group M-3 (see Table XVIII, p. 70, Progress Report for May 1967, ANL-7342), consisting of fifteen U-15 w/o Pu-10 w/o Zr fuel elements, is still awaiting insertion into EBR-II for irradiation to 10 a/o burnup maximum.

Group M-4 consists of 40 fuel elements of U-15 w/o Pu-(10-12) w/o Zr with 93% enriched uranium (see Progress Report for June 1967, ANL-7349, p. 76). The fuel pins are being fabricated. All jacket materials are on hand and processing has begun. The fuel diameter (0.163 in.), being larger than that of fuel previously investigated, will enable linear heat ratings of up to 14 kW/ft to be attained.

b. Radiographic Tests on Experimental Fast Reactor Components. Radiographic examinations of four EBR-II irradiation capsules are being conducted to determine the quality of the sodium bond during successive heating and cooling cycles, programmed to simulate reactor environment.

Initially, the capsules were examined when the sodium was at room temperature and only a few shrinkage voids were detected at that time. The capsules were then individually heated to approximately 265°C for 15 min and radiographed. A series of four views were taken of each capsule, which was rotated axially about 35° after each exposure. After radiography of the capsules while hot, they were cooled from the top down, so that the upper sodium froze first. The capsules were radiographed again after the sodium had solidified. The hot testing revealed no detectable shrinkage voids, whereas the cold capsules contained a significantly large number of shrinkage voids. These heating and cooling cycles were repeated three additional times. After the second heating cycle, several large shrinkage voids were detected. Since the void formation was believed due to insufficient heating time, the time was increased to 45 min. No shrinkage voids have been detected after the third and fourth heating cycle, but radiographs of the cold capsules show large numbers of shrinkage voids. Several more cycles will be made in order to show that a reasonably good bond will be maintained.

Neutron radiographs of two sodium-bonded capsules were also taken in order to determine if neutron radiography would reveal voids in the narrow sodium annulus. Voids of the order of 0.625 in. or more do appear to be detectable. Therefore, neutron radiography may also be considered a possible technique for sodium-bond evaluation of irradiated capsules.

2. Oxide Fuels

Additional microprobe examinations planned for vibratorily compacted mixed oxide in element No. SOV-6 (see Table XL in Progress Report for June 1967, ANL-7349, p. 77), were continued with the mounting and preparation of a core-drilled sample. Previous microprobe examinations of a core removed from SOV-6 provided the approximate compositions of metallic and oxidized fission products occurring as discrete particles in the irradiated structure (see Progress Report for May 1967, ANL-7342, p. 71).

3. Carbide Fuels

a. Mixed-carbide Fuel Performance. There has been no change in the status of elements undergoing irradiation in EBR-II (see Table XXI, Progress Report for July 1967, ANL-7357, p. 83). Analysis of element No. SMP-1 for burnup was completed. Values for linear heat rating and cladding temperature, corrected for the analytical burnup result, are shown together with values for VMV-1 and SMV-1 in Table XXII.

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

	Element Number		
	SMP-1	VMV-1	SMV-1
Fuel Composition, w/o	(U _{0.8} Pu _{0.2})C	(U _{0.8} Pu _{0.2})C	UC-20 w/o PuC
Fuel Synthesis	Arc Melted	Arc Melted	Arc Melted
Fuel Form	Pellets	Vibratorily Compacted	Vibratorily Compacted
Fuel Diameter, in.	0.253	0.253	0.257
Fuel Length, in.	13.4	14.0	14.0
Effective Density, %	81.4	85.9	80.0
Cladding Material	Type 316 SS	Vanadium	Type 316 SS
Cladding OD, in.	0.306	0.301	0.306
Cladding Wall, in.	0.024	0.024	0.025
Max Heat Rating, kW/ft	15.9	25.8	21.1
Max Clad Temp, °C	565	640	625
Max Burnup, a/o	1.91	3.02	2.59

b. Compatibility of (U,Pu) Carbides with Potential Jacketing Materials.

(i) (U,Pu) Carbides versus V-15 w/o Cr-5 w/o Ti. No reaction was observed in a 1000-hr test of V-15 w/o Cr-5 w/o Ti in contact at 800°C with hyperstoichiometric $(U_{0.8}Pu_{0.2})C$ containing approximately 20 v/o $(U,Pu)_2C_3$. This alloy has been tested in both sheet (see Progress Report for October 1966, ANL-7267, p. 38) and rod form from different melts under these testing conditions. Previously (see Progress Report for June 1967, ANL-7349, p. 79), no reactions were observed with single-phase $(U_{0.8}Pu_{0.2})C$.

(ii) Compatibility of Uranium Sulfide with Selected Jacket Materials. In order to compare the interaction of various jacket alloys with ceramic fuel materials other than (U,Pu) carbides, six alloys, previously tested with stoichiometric and hyperstoichiometric (U,Pu)C, were heat treated at 800°C for 1000 hr with uranium monosulfide. The uranium sulfide composition used was slightly hyperstoichiometric in sulfur and contained 0.17 w/o oxygen, resulting in impurity phases of approximately 5.5 v/o U_2S_3 and 3 v/o UOS. After the pellets were sintered at 1800°C, the impurity phases were found as a eutectic like precipitate surrounding relatively large (approximately 100 μ) grains of US.

No changes were observed metallographically in any of the five iron- or nickel-base alloys resulting from contact with the US composition. The only evidence of fuel-jacket interaction in the case of the three iron-base alloys (Type 304 stainless steel, 16-25-6, Haynes 56) and the two nickel-base alloys (Hastelloy-X and Inconel 625) was a small gray band observed on the fuel side of the interface in each specimen. The width of these bands ranged from 3 to 6 μ . The size was evidently not related to the iron or nickel contents of the jacket alloys, since the smallest layers were observed with the alloys having the highest (Type 304 stainless steel) and lowest (Inconel 625) iron-to-nickel ratios. The layers are probably related to the diffusion of oxygen and/or nitrogen from the jacket alloy into the US. Further tests are being run at 900°C with Type 304 stainless steel and Inconel 625.

One vanadium-base alloy, V-20 w/o Ti, was also tested. No reaction zone was found in the V-20 w/o Ti; however, etching produced a more distinct outline of some of the grain boundaries within about 50 μ of the interface than in the remainder of the jacket pellet. There was no gray phase at the US-jacket interface as observed in the iron- and nickel-base alloys. Instead, the U_2S_3 phase appeared to be reduced to US within 5 μ of the interface.

In general, these tests have shown good compatibility between uranium monosulfide and a variety of potential commercial jacket

materials at 800°C. Future testing with $(U_{0.8}Pu_{0.2})S$ will provide a more direct comparison of jacket compatibility with carbide and sulfide fast reactor fuels.

c. Pressing and Sintering of Carbide Fuel Pellets. With an adequate supply of high-purity (U, 15 w/o Pu)C prepared by the fluidized-bed technique now available (see Progress Report for February 1967, ANL-7308, p. 31), emphasis has turned to a study of the techniques for pressing and sintering this material to produce pellets suitable for irradiation testing. Equipment for this purpose has been set up in a glovebox. The irradiation test pellets are to have narrow tolerances on dimensions, density, and carbon-to-metal ratio. To preserve the stoichiometry and purity of the (U,Pu)C powder, the pellets will be prepared in a high-purity helium-atmosphere glovebox adjoining the powder-preparation glovebox. The pellets will be pressed in a semiautomatic hydraulic press;⁴ carbide powder and a binder material will be fed into the die by hand, but pressing and ejection of the green pellet will be done in an automatic cycle. After removal of the binder in a presintering furnace, the pellets will be sintered in a resistance furnace under either vacuum or inert gas atmosphere. Experimental progress to date is described below.

(i). Pellet Density. Since shrinkage of the green pellets occurs during the sintering process, the die diameter must be precisely selected,

Fig. 12. Relationship between Pellet Density and Pellet Diameter before and after Sintering.

D_g = green-pellet diameter,
 ρ_g = green-pellet density (binder-free basis), and
 ρ_s = specified pellet density (85% of theoretical).

the green-pellet density must be carefully controlled, and the sintering conditions must be reproducible in order to prepare pellets of a specified diameter and density. Figure 12 shows the calculated relationships between the pellet density and diameter before and after sintering. From the figure, it may be ascertained that if the green-pellet diameter and green density are in the right relationship to the specified density (85% of theoretical), a variation in the final sintered density of 1% from the specified density will cause an error in the final diameter of 1 mil.

Uranium-15 w/o plutonium pellets were pressed to determine the effect of the binder content on the green density. The stearic acid binder was dry-mixed with the carbide powder; a solvent was not used.

⁴ Enerpak System with 2 double-acting 9-ton cylinders powered by a two-stage electrical pump. Currently, the unit is operated at a maximum pressure of 5000 psi.

The results of the pressing operation are shown in Fig. 13. It is apparent that the green density of the pellets may be adjusted over a sufficiently

Fig. 13: Green Density of (U,Pu)C Pellets as a Function of Stearic Acid Binder Content (60,000 psi pressing pressure)

broad range to accommodate the die sizes available, and meet the pellet-density and -diameter specifications.

In another pellet-pressing experiment with a different batch of feed material, nine (U,Pu)C pellets were pressed under identical conditions of binder content (1% stearic acid), die size (0.252-in. dia), and pressure (60,000 psi). The die was loaded with a fixed volume of powder contained in a scoop. The pellet

lengths varied by $\pm 7.7\%$ from the mean of 0.304 in.; the pellets varied in green density by about $\pm 0.5\%$ from the mean of 7.43 g/cc. The green density was a linear function of the length of the pellet; as expected, the longest pellets had the lowest density. Improvements in measurement of the powder volume required for making a specific size pellet are expected to result in a satisfactorily low variation in green density; weighing of the feed material for each pellet should not be necessary.

(ii) Sintering. The binder appeared to be completely removed from the pellets produced in the above operations by heating in the presintering furnace at 650°C for one hour under vacuum. Weighing before and after presintering indicated 100% removal of the binder. The pellets were then transferred to the sintering furnace.

Because the sintering furnace is located in a glovebox, a coolant is needed to dissipate heat. Since the projected use of the furnace involved the preparation of enriched uranium carbides, criticality considerations indicated the desirability of using a nonhydrogenous coolant. Experiments were conducted, therefore, using perchloroethylene as a coolant. It was found that the use of perchloroethylene limited the furnace temperature to 1500°C. One batch of presintered pellets was sintered at 1500°C under vacuum for 3 hr. However, this temperature was apparently inadequate for densification of the pellets to the required specifications. Even those in the center of the furnace reached a density of only 75% of theoretical.

The sintering furnace has now been changed to water cooling,⁵ which will permit higher sintering temperatures ($\geq 1800^\circ\text{C}$). The preparation of pellets of adequate density should now be possible.

⁵This necessitated lowering the amount of fissionable material that will be permitted in the glovebox.

4. Fuel Cladding and Structure

a. Irradiation of Jacket Material. Tube-burst tests were conducted at 550, 650, and 750°C on specimens of V-15 w/o Ti-7.5 w/o Cr alloy irradiated in EBR-II subassembly XO13 to a nominal fluence of 5×10^{21} n/cm² in the temperature range from 510 to 704°C. The results, including those reported previously, are given in Tables XXIII to XXV.

TABLE XXIII. Tube-burst Properties at 550°C of
V-15 w/o Ti-7.5 w/o Cr Alloy Irradiated in EBR-II
in the Temperature Range from 510 to 704°C

Distance from Reactor Midplane (cm)	Fluence (n/cm ² x 10 ²¹)	Tangential Rupture Strength (psi)	Diameter Increase (%)	Average Diameter Increase (%)
Unirradiated	-	100,100	(a)	(a)
Unirradiated	-	101,300	(a)	(a)
-20 to -10	3.1 to 5.1	113,800	0.8 to 2.3	1.4
-10 to 0	5.1 to 5.8	128,000	1.8 to 2.4	2.0
0 to +10	4.8 to 5.8	118,200	1.0 to 1.3	1.2
+10 to +20	2.8 to 4.8	122,200	0.4 to 5.5	3.6
+20 to +30	1.6 to 2.8	116,000	1.5 to 5.2	3.8
+30 to +40	0.8 to 1.6	112,700	1.1 to 5.7	2.5
+40 to +50	0.3 to 0.8	92,500	0.5 to 1.6	1.0
+50 to +60	0.2 to 0.3	110,500	2.2 to 3.6	2.8

(a) Shattered.

TABLE XXIV. Tube-burst Properties at 650°C of
V-15 w/o Ti-7.5 w/o Cr Alloy Irradiated in EBR-II
in the Temperature Range from 510 to 704°C

Distance from Reactor Midplane (cm)	Fluence (n/cm ² x 10 ²¹)	Tangential Rupture Strength (psi)	Diameter Increase (%)	Average Diameter Increase (%)
Unirradiated	-	97,800	7.4 to 8.8	8.1
-20 to -10	3.1 to 5.1	111,000	1.5 to 3.2	2.5
-10 to 0	5.1 to 5.8	112,000	1.2 to 1.7	1.4
-10 to 0	5.1 to 5.8	115,600	0.7 to 4.6	2.0
0 to +10	4.8 to 5.8	111,600	0.9 to 2.4	1.6
+10 to +20	2.8 to 4.8	99,600	0.4 to 1.6	1.0
+20 to +30	1.6 to 2.8	103,600	0.8 to 3.5	1.8
+20 to +30	1.6 to 2.8	106,200	1.1 to 1.8	1.4
+30 to +40	0.8 to 1.6	99,100	2.0 to 5.5	3.4
+30 to +40	0.8 to 1.6	104,000	1.3 to 4.8	2.5
+40 to +50	0.3 to 0.8	95,600	3.6 to 4.0	3.8
+40 to +50	0.3 to 0.8	100,500	4.2 to 4.9	4.6
+50 to +60	0.2 to 0.3	100,000	2.8(b)	2.8
+50 to +60	0.2 to 0.3	88,000	(a)	(a)

(a) Shattered.

(b) One measurement.

TABLE XXV. Tube-burst Properties at 750°C of
V-15 w/o Ti-7.5 w/o Cr Alloy Irradiated in EBR-II
in the Temperature Range from 510 to 704°C

Distance from Reactor Midplane (cm)	Fluence (n/cm ² x 10 ²¹)	Tangential Rupture Strength (psi)	Diameter Increase (%)	Average Diameter Increase (%)
Unirradiated	-	90,200	6.9 to 8.6	7.8
-20 to -10	3.1 to 5.1	91,600	1.9 to 3.6	2.5
-10 to 0	5.1 to 5.8	97,800	1.5 to 2.7	2.0
0 to +10	4.8 to 5.8	96,900	1.5 to 4.1	2.5
+10 to +20	2.8 to 4.8	93,800	4.6 ^a	4.6
+20 to +30	1.6 to 2.8	93,300	2.7 to 3.0	2.8
+30 to +40	0.8 to 1.6	88,900	1.1 to 2.6	1.7
+40 to +50	0.3 to 0.8	90,700	2.0 to 7.3	4.9
+50 to +60	0.2 to 0.3	89,300	2.2 to 7.0	5.4

^aOne measurement.

The ductility of the specimens is given by the average percent diameter increase of the tubes after rupture (see last column in the tables). The ductility of the irradiated alloy was lower than that of the unirradiated alloy at test temperatures of 650 and 750°C. This is also probably true at a test temperature of 550°C even though the unirradiated alloy shattered to such an extent that no diameter measurements, and hence no preirradiation and postirradiation ductility comparisons, were possible. There was no apparent relationship between ductility and fluence. At a test temperature of 750°C, the postirradiation ductility of V-15 w/o Ti-7.5 w/o Cr alloy was higher than that of material tested at 550 and 650°C.

At all test temperatures the increase in burst strength of V-15 w/o Ti-7.5 w/o Cr alloy due to irradiation was related to fluence. The highest increase in burst strength occurred at a test temperature of 550°C and the smallest increase at a test temperature of 750°C. Additional tests will be necessary on both unirradiated and irradiated specimens in order to determine more accurately the effects of irradiation on V-15 w/o Ti-7.5 w/o Cr alloy.

Previously reported data (see Progress Report for May 1967, ANL-7342, p. 77) indicated that irradiation of V-20 w/o Ti and V-15 w/o Ti-7.5 w/o Cr alloy to a nominal fluence of 5×10^{21} n/cm² decreases the ductility of both alloys from that of the unirradiated material from approximately 25 to 20% at a test temperature of 550°C. In order to determine the effect of grain size and test temperature on the postirradiation strength and ductility of the alloys, tensile tests were conducted at 400 and 550°C with samples of these alloys that had grain sizes ranging from 10 to 35 μ prior to irradiation. Tests at 650°C are in progress. The specimens were irradiated in EBR-II subassembly XO13 to a nominal fluence of 5×10^{21} n/cm² in the temperature range from 510 to 704°C. They were tested at a

strain rate 0.5 mm/min over a gauge length of 25 mm. The results of the tests are given in Table XXVI.

TABLE XXVI. Postirradiation Tensile Properties of V-20 w/o Ti and V-15 w/o Ti-7.5 w/o Cr Irradiated in EBR-II Subassembly XO13 to a Nominal Fluence of 5×10^{21} n/cm²

Alloy (w/o)	Fluence (n/cm ² × 10 ²¹)	Test Temp (°C)	Grain Diameter (μ)	0.2% Yield Strength (kg/mm ²)	Tensile Strength (kg/mm ²)	Uniform Elongation (%)	Total Elongation (%)
V-20 Ti	4.6	400	10	48.0	65.6	14.6	20.2
V-20 Ti	4.6	400	10	50.5	67.4	16.3	23.3
V-20 Ti	4.4	400	20	51.8	66.2	13.1	18.7
V-20 Ti	4.4	400	20	51.9	66.0	13.2	18.4
V-20 Ti	4.2	400	35	51.1	64.6	10.5	15.6
V-20 Ti	4.2	400	35	53.3	65.0	11.2	16.8
V-20 Ti	4.0	550	10	51.4	66.0	11.7	17.8
V-20 Ti	4.2	550	10	47.6	66.6	12.0	17.8
V-20 Ti	3.6	550	20	52.8	65.2	12.8	18.1
V-20 Ti	3.7	550	20	55.4	70.6	15.5	21.2
V-20 Ti	4.0	550	35	52.6	68.9	16.1	21.7
V-15 Ti-7.5 Cr	3.6	400	10	56.4	74.7	13.0	19.0
V-15 Ti-7.5 Cr	3.8	400	10	59.6	76.9	13.9	19.4
V-15 Ti-7.5 Cr	4.3	400	25	63.2	76.6	14.1	18.8
V-15 Ti-7.5 Cr	4.4	400	25	66.7	81.5	15.2	20.3
V-15 Ti-7.5 Cr	4.2	400	35	62.7	76.1	13.7	17.8
V-15 Ti-7.5 Cr	4.4	400	35	62.9	77.2	16.0	19.8
V-15 Ti-7.5 Cr	4.2	550	10	61.0	77.1	11.5	16.8
V-15 Ti-7.5 Cr	4.0	550	25	66.3	79.6	14.6	19.5
V-15 Ti-7.5 Cr	3.8	550	35	65.2	80.0	16.1	19.4

At a test temperature of 400°C, the ductility of irradiated V-20 w/o Ti alloy decreased linearly as the grain size increased from 10 to 35 μ, whereas at a test temperature of 550°C, the ductility increased linearly over the same range of grain size.

There were slight decreases and increases in tensile strength, respectively, at test temperatures of 400 and 550°C as the grain size was increased from 10 to 35 μ. The yield strength of irradiated V-20 w/o Ti alloy increased with increase in grain size at both test temperatures. The maximum increase in yield strength occurred for irradiated material with a grain size of 20 μ.

The ductility of irradiated V-15 w/o Ti-7.5 w/o Cr alloy did not change significantly with grain size at a test temperature of 400°C. At a test temperature of 550°C, the ductility increased linearly as the grain size increased from 10 to 35 μ. At 550°C, the increase in ductility was greater for an increase in grain size from 10 to 25 μ than from 25 to 35 μ. The strength of irradiated V-15 w/o Ti-7.5 w/o Cr alloy increased with increased grain size at both test temperatures. There was no significant gain in strength of the irradiated alloy for an increase in grain size from 25 to 30 μ.

The test results indicate that the postirradiation strength and ductility of V-20 w/o Ti can be increased by increasing the preirradiation grain size to $35\ \mu$ (except at the 400°C temperature), and that the postirradiation strength and ductility of V-15 w/o Ti-7.5 w/o Cr alloy can be increased by increasing the preirradiation grain size to $25\ \mu$. No further beneficial effect on the strength and ductility of V-15 w/o Ti-7.5 w/o Cr alloy is obtained by increasing the grain size from 25 to $35\ \mu$.

b. Development of Refractory-metal Alloys for Service in Oxygen-contaminated Sodium. The static, oxygen-refreshed testing of promising vanadium alloys at 450°C has been completed after about three-month exposures. All samples lost weight in both the 175°C and the 110°C cold-trapped test loops, the rate of weight loss being approximately linear. The corrosion rates (linear slope) are given in Table XXVII for these experiments.

TABLE XXVII. Corrosion of Vanadium Alloys in 450°C Sodium

Alloy ^a	Corrosion Rate (mdd) ^b	
	Cold Trap at 110°C	Cold Trap at 175°C
V-20Ti	0.85	1.9
V-15Ti-7.5Cr	2.1	1.4
V-15Cr-Ti (annealed)	3.6	6.8
V-15Cr-5Ti (50% cold work)	3.8	7.8

^aNumbers in alloy designations indicate weight percentages.

^bmdd = milligrams per decimeter-day.

The dynamic experiment at 450°C is continuing. The weight changes (as usual) are smaller in this system than in the refreshed static system with the same cold-trap setting. Two differences in the systems, besides velocity, have been noted: (a) the refreshment (or cleanup) rate is much higher in the dynamic system; (b) the dynamic loop is built of a stabilized stainless steel (Type 318) whereas the static systems are built of Type 304 stainless steel. Thus carbon transfer from the steel may be playing a role in the static refreshed testing of vanadium alloys.

E. 1000-MWe Plant

1. Contract Management and Liaison

The Laboratory is coordinating the activities of the contractors to assure that all studies of large liquid-metal-cooled fast breeder reactors

for central-station power plants are being conducted in accord with the established scope and common ground rules and background information.

a. Westinghouse. Final requirements for the reactor envelope, and the core and blanket fuel subassemblies have been completed. The tradeoff study for modular versus nonmodular core has begun.

Conceptual design of the pipe-loop-type heat exchange and pumping equipment is nearing completion, and costs are being estimated. Turbine definition, costs, and heat balances have been developed for the sodium reheat and nonreheat cycles. The architect-engineer has begun working on plant layout and cost estimates for the primary system, containment, and fuel-handling hot cell.

b. Atomics International. Atomics International has established the areas of work that constitute Tasks I, II, III, and IV, including the scheduling of the parametric and tradeoff studies. Work plans for Task I have been completed; development of work plans for the other tasks is continuing. Progress has been made in establishing a reactor concept for Task I, based on minimum research and development requirements.

c. Combustion Engineering. Parametric studies for both modular and cylindrical cores with mixed uranium and plutonium carbides have been completed. The cylindrical core with a modest height-to-diameter ratio has been selected as the reference core. A design-layout drawing of the reactor vessel, including internals, has been completed. Work is continuing on conceptual designs for the fuel subassembly and the fuel-handling scheme.

d. Babcock & Wilcox. All written work for the Task I and concept-system description reports are technically complete. Task II engineering work and tradeoff studies are continuing. Plant-arrangement drawings are progressing for the reference concept. The spoiled versus unspoiled-geometry tradeoff study is nearing completion. Work continues on physics methods development.

e. General Electric. Work progressed in the areas of core and safety analysis, nuclear design, core and primary-system mechanical design, and plant and systems design. GE is evaluating both conservative and advanced-type core designs; optimization of core and plant systems is underway. Other major activities include establishment of possible safety and design criteria for conservative and advanced designs and the performance of conceptual cost studies.

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Development of a Numerical Program for Doppler Broadening

A program has been written to provide numerical data for Doppler-broadened cross sections at various temperatures. The present program includes a revised version of the TEMPO⁶ code and several additional options. The program requires punched decks of unbroadened cross sections. As an option, single-level unbroadened cross sections σ_c , σ_s , and σ_f can be calculated for use in the broadening routines. Broadened and unbroadened cross sections may also be punched by the program. A plotting routine is also included which allows the user to plot and superimpose any two curves representing, for example, the broadened and unbroadened cross sections of interest.

This program is to be used as a subroutine in the analysis of cross sections represented by multilevel Breit-Wigner resonances.

B. Reactor Fuels and Materials Development

1. Behavior of Reactor Materials

a. Corrosion Inhibition by Dissolved Getters in Liquid Sodium.

Analyses of several samples of stock sodium before and after circulation through a cold trap containing excess oxygen were reported (see Progress Report for July 1967, ANL-7357, pp. 102-103) to show that such cold trapping can lower the concentration of calcium from several hundred wt ppm, and that of strontium from 25 to 50 wt ppm, to less than 10 wt ppm in both cases, but the concentrations of potassium are not greatly affected. The sodium so purified is acceptable for circulation through dissolved getter (DG) loops. Distillation of used sodium reduces the iron content and also was expected to make the sodium suitable for use in a DG loop.

Results of spectroanalysis of sodium distillate indicate that, regarding metallic impurities of present interest, the distilled product is acceptable for use in the DG program. The program is thus proceeding, use being made of "model tank" sodium, distilled, hot trapped, and cold trapped in the tantalum still. Analyses for nonmetallic impurities of interest are being obtained.

A preliminary corrosion test was made of V-20 w/o Ti and of commercially pure nickel samples in 650°C static sodium containing 1000 wt ppm of magnesium. After a four-day exposure the samples had

⁶Gregson, K., and James, M. F., TEMPO--A General Doppler Broadening Programme for Neutron Cross Sections, AEEW-M 518 (1965).

gained weight slightly (about 0.25 mg/cm^2) and displayed an uneven, gray surface discoloration. No effect was seen that would lead to modification of the loop tests planned for these materials.

b. Dissolution Kinetics in Liquid-metal Systems. A sodium-purification system consisting of a one-liter-capacity vacuum distillation still and a hot-gettering vessel has recently been placed in operation. The apparatus is of double-wall construction. The inner container in contact with the sodium is made of tantalum. The Type 304 stainless steel outer vessel was required to prevent oxidation of the tantalum. Purified helium is used as the heat-transfer medium between the two vessels.

After the initial period of heating to the distillation temperature ($\sim 550^\circ\text{C}$), one liter of sodium can be transferred from the boiler to the condenser in approximately 2 hr. The distillation process is monitored by a temperature record of the boiler section. The characteristic abrupt rise in temperature of the boiler section indicates when the process is completed.

Twenty liters of sodium have been processed. Samples before and after purification have been submitted for spectrochemical analysis for metallic impurities. The initial results indicate that the limits of detectability for metallic impurities by the spectrographic method are typically too high to assess quantitatively the purity of either the starting or the purified sodium. An effort is being made to obtain more accurate analyses of the sodium samples.

c. Studies of Corrosion in Lithium at Elevated Temperatures. Samples, equipment, and a high-vacuum furnace are being readied for a study of the behavior of silicon-tantalum protective layers on tantalum in lithium-silicon solution at 1000°C and hotter. The protective layer was not identifiable by an X-ray powder method, for the layer material consists of a Ta-Si-W compound that is not available in the ASTM Index.

2. Chemistry of Irradiated Fuel Materials

a. Distribution of Gamma-emitting Fission Products in Mixed-oxide Fuel. Cross-sectional cuts of a UO_2 -20 w/o PuO_2 fuel element (SOV-6)⁷ irradiated in EBR-II to an estimated burnup of 2.9 a/o (U+Pu) have been radially sampled to determine the distribution of gamma-emitting fission products across the fuel element (see Progress Report for July 1967, ANL-7357, pp. 104-105).

The samples were dissolved and analyzed for uranium by a fluorophotometric technique and for plutonium by alpha counting. The counts per minute of each of the fission products per milligram of heavy

⁷Data on fabrication and metallographic examination were reported in Progress Reports ANL-7308, p. 28, and ANL-7317, p. 52.

metal (U + Pu) were calculated and are given in Table XXVIII. These data indicate essentially the same findings that were obtained from the data normalized to cerium-144 (see ANL-7357, p. 104).

TABLE XXVIII. Distribution of Various Fission-product Activities in Irradiated Mixed-oxide Fuel (SOV-6)

Sample No.	Location of Cross Section	Distance ^a (in.)	Weight U + Pu (μg)	Counts/min/mg (U + Pu)						
				Ce ¹⁴⁴ (x 10 ⁻⁴)	Ce ¹⁴¹ (x 10 ⁻³)	Ru ¹⁰³ (x 10 ⁻³)	Ru ¹⁰⁶ (x 10 ⁻²)	Cs ¹³⁷	Zr ⁹⁵ (x 10 ⁻³)	Nb ⁹⁵ (x 10 ⁻⁴)
6	4 in. from bottom	0.081	58.0	1.68	5.46	1.81	7.3	<20	3.43	1.74
7		0.057	116	1.62	4.92	1.58	5.60	<20	3.00	1.25
8		0.043				plutonium analysis not complete				
9		0.025	68.2	1.74	5.39			2.76 x 10 ²	3.28	1.44
10	6 in. from bottom	0.006	116	1.65	5.06	2.22	13.0	2.39 x 10 ³	3.24	1.42
1		0.074	70.2	1.02	5.58	1.89	7.7	<30	3.64	1.74
2		0.060	49.2	1.77	5.42	1.83	6.26	<30	3.23	1.41
3		0.043	116	1.64	5.05	2.21	14.7	35	3.05	1.25
4		0.020	80.8	1.77	5.57	2.29	13.2	5.6 x 10 ²	3.38	1.48
5	9 in. from bottom	0.009	110	1.64	4.96	2.23	13.3	1.30 x 10 ³	3.18	1.29
11		0.072	118	1.58	4.84	1.79	5.8	<20	3.12	1.47
12		0.050	132	1.55	4.71	2.13	11.0	24	2.78	1.14
13		0.038	132	1.57	4.80	1.78	9.2	57	3.09	1.30
14		0.023	102	1.58	4.94	2.14	11.4	4.64 x 10 ²	3.08	1.34
15		0.009	28	1.64	5.10	2.30	10.4	1.05 x 10 ³	3.03	1.33

^aDistance from periphery of fuel element.

No major differences were observed in the behavior of any of the fission products across the three sections of the fuel element. In each cross section, the counts per minute of the fission products per milligram of heavy metals (U + Pu) indicated that no migration of cerium or zirconium occurred. There appeared to be some slight enrichment of niobium at the center of the pin. Migration of ruthenium to the outer edge of the columnar grain region was noted, but it appeared to move very slowly, if at all, in the outer zones of the element. The difference in the ruthenium-103 and -106 ratios can be accounted for by the difference in the half-lives of the two isotopes. Cesium-137 was not detected in any of the inner samples and was just detectable at the edge of the columnar grains. However, its concentration increased by an order of magnitude from the edge of the columnar grain region to the equiaxial grain region and by another order of magnitude to the undisturbed outer portion.

b. Application of Laser Microsampling to Analysis of Fuels.

A small laser attached to a microscope is being used as a microsampling device. Samples obtained by this technique should be useful for the analysis of very small areas of metallurgical specimens. Three methods of analysis are being considered: alpha pulse analysis, gamma spectrometry, and neutron activation. The sampling technique and initial tests of its application to the analysis of irradiated material have been described (see Progress Report for April 1967, ANL-7329, p. 62).

Six laser samples were taken across the radius of a 4-mm-dia UO₂ fuel pin (6.8 a/o burnup, 13% enriched). Craters approximately 50 μ in diameter were formed at each location, corresponding to a sample weight of ~0.5 μg.

The specimens were counted for ~18 hr using a solid-state alpha pulse-height analyzer. Energy peaks obtained at 5.14, 5.48, and 6.1 MeV corresponded to the peaks for Pu^{239} - Pu^{240} , Pu^{238} - Am^{241} , and Cm^{242} , respectively.

The above specimens were also analyzed gamma-spectrometrically using a lithium-drifted germanium detector. In the five samples taken from the body of the fuel pin, the isotopes that were identified from the scans were Ce^{144} , Pr^{144} , Ru^{106} , Zr^{95} , Nb^{95} , Cs^{134} , and Cs^{137} ; few major differences in fission product ratios were noted. However, the activity in the sample from the cladding-fuel interface region was principally cesium.

Successful application of the laser sampling technique to neutron activation analysis could provide a method of analyzing inactive specimens for about 70 different elements. Tests are being made to determine the feasibility of using neutron activation for the analysis of these extremely small samples (3×10^{-9} to 1×10^{-3} g). Initial tests were made with samples of pure manganese metal that weighed $\sim 10^{-8}$ g. Seven samples were irradiated in CP-5 for 30 min and then counted for Mn^{56} using a 256-channel analyzer and a sodium iodide detector. The counting rates that were obtained are considered adequate. These initial results indicate that samples prepared by the laser sampling technique may be suitable for neutron activation analysis of some inactive materials.

3. Basic Fabricability

a. Determination of Elastic Moduli of High-temperature Materials by Ultrasonics. A new furnace that can be utilized for measuring ultrasonic velocities in rods of Type 304 stainless steel after use in EBR-II is being modified to enable rods to be tested at high temperatures over a range of diameters. For tests at temperatures above 1100°C a tungsten buffer rod will be coupled to short specimens. The data to be obtained will be used in deducing the elastic moduli of the sample material.

A specimen holder has been received and vacuum tested. Specimens designed for the old furnace can now be used in the new furnace. The new specimen holder has increased the versatility of the present equipment.

Data to calculate the elastic moduli of EBR-II stainless steel 304 specimens are being compiled. A series of three runs up to 1200°C will be made on the four EBR-II specimens, and from these data the elastic moduli of each will be calculated.

b. Development of a Neutron-image Intensification System. Further work on the track-etch method of detecting thermal-neutron images⁸ is being done in order to study possible methods for contrast enhancement. Emphasis

⁸Berger, H., and Kraska, I. R., A Track-etch, Plastic-film Technique for Neutron Imaging, Trans. ANS 10, No. 1, 72-73 (1967).

at present is being placed on the preparation of thin, uniform, cellulose nitrate layers, since this is a first requirement for the enhancement techniques being considered. Some progress has been made in floating thin layers on water and in pressing these layers between glass plates.

c. Development of a Nonreactor Neutron Source for Radiography.

Initial experiments with an Am-Cm-Be neutron source to determine the characteristics of several moderator materials are in progress. A neutron source is now being studied in a light-water moderator in the Building 200, M-Wing cave facility. Plans are to compare the light-water moderator with a BeO moderator experimentally, in order to obtain results to check a computer calculation of the thermal-neutron intensities that should be obtained in several moderator materials.

d. Passive Ultrasonic Test Techniques. The feasibility of using lithium niobate as the piezoelectric element in a transducer probe was investigated. This piezoelectric material is particularly attractive as part of an ultrasonic system to detect incipient boiling in a reactor since its Curie temperature is about 1200°C. An ultrasonic system with a lead zirconate-titanate transmit probe and a lithium niobate receiving crystal (1-cm cube) was set up to evaluate the crystal. Thin brass plates, attached to corresponding faces of the crystal with a rubber band, served as electrodes. The most sensitive axis to longitudinal wave motion was determined by observation of the pulse amplitude of the received signals. The crystal will be mounted in a stainless steel housing prior to additional tests.

e. Radiation Dosimetry. Experimental attempts to produce light fluorescence from gamma-radiated lithium fluoride crystals by subjecting them to ultrasonic vibrations have been made. Initially, 4- and 8-MHz lead zirconate-titanate transducers were attached to lithium fluoride crystals with Eastman 910 adhesive. After the assemblies had received a gamma dosage of about 500 R, a continuous wave signal was applied to each transducer. No fluorescence was observed or could be detected by a photomultiplier circuit from either assembly. Subsequent tests with a 2-in. diameter, 1.9-MHz barium titanate transducer and an alcohol couplant were also unsuccessful. Further tests are planned.

C. Engineering Development

1. Development of Master-Slave Manipulator Systems

a. Electric Master-Slave Manipulator, Mark E4A. The slave arm of the Mark E4A has been tested for heating of the motors and force output capabilities as a function of time. Figure 14 shows stator copper temperature and force output of the Z motion as a function of time. Each phase of each of the four motors was connected to 150 V, 60 cycle, 2-phase power at time zero. This power was applied to the Y, Z and the

two elevation motions. The regular cooling fan was on. The load capacity at room temperature (28.5°C) was about 70 lb but rapidly decreased, crossing the 50-lb value in about 2.3 min.

Fig. 14. Behavior of Stator Copper Temperature and Force Output of Mark-E4A Slave Arm with Time

In air the force levels at about 42 lb and can be kept at this value indefinitely with the copper temperatures of the hottest motor at 170°C and the average of the four motors at 153°C . In argon, the continuous load capacity is 37 lb with the voltage reduced to 143 V on each phase instead of 150 V.

Additional cooling tests will be carried out with different ambient temperatures in both air and argon. Also tests will be made on the X motion because it will be cooled with a separate fan.

The "motion exerciser" which will apply six simultaneous motions to the master arm has been completed in design and is being fabricated. This test device will give typical, but ac-

celerated, tests to the entire arm. During the test the master and slave arms will be connected in their normal manner through the servo amplifiers and the slave arm will be "holding" a mass load.

b. Low-inertia Servo Motors for Manipulators. The special 2-pole, 2-phase low-inertia motor has been wound and assembled. Preliminary tests indicate the motor is performing moderately close to design expectations. A gear box for testing this motor in a force-reflecting servo system has been fabricated.

2. General Heat Transfer

a. Boiling-sodium Experiment. To provide the understanding of liquid-vapor behavior that is essential for analyzing the steady-state behavior and the safety of sodium-cooled fast reactors for large central-station power plants, the heat transfer and two-phase flow characteristics of boiling sodium are being investigated to temperatures of 2100°F and pressures of approximately 8 atm.

(i) Niobium-1% Zirconium Loop. During heat-loss calibrations at a sodium temperature of 1525°F , the tantalum shutters used for condenser heat-loss control were rotated to a position about 30° from fully closed and became stuck. The cause was traced to the drive chain; the chain was lengthened to its maximum diameter to allow for thermal expansion. While

the vacuum facility was in a dismantled condition, several minor repairs and modifications were completed in preparation for boiling runs.

b. 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.

(i) Counterflow Double-pipe Heat Exchangers. A draft of a topical report⁹ describing both analytical and experimental portions of the research has been completed. Further analytical investigations continue (see Progress Report for July 1967, ANL-7357, pp. 115-116).

(ii) Nonsymmetrical-duct Heat Exchangers. Construction of the small heat-transfer loop to be used for experiments with sodium in model heat exchangers continues. Duct work for the air cooler has been installed, fabrication of the electrical heaters is about 75% complete, and the first test section has been temporarily installed for the purpose of checking placement and thermocouple installation at various locations along its length.

(iii) Liquid-metal-heated Steam Generators. Relocation and redesign of the mercury loop that had been used for counterflow double-pipe heat exchanger research has begun. The equipment is being converted to perform experiments with simple simulations of liquid-metal-heated steam generators.

c. Fog-flow Heat Transfer and Fluid Flow. To determine the utility of a particular "fog flow" model of a boiling fluid, and to determine and correlate quantities needed for applying the model (e.g., turbulent-diffusion and droplet-deposition-rate coefficients), simplified versions of the model are being applied to situations for which there are experimental data, so as to verify further that the model is a reasonable one. Theoretical and experimental research are being conducted on specific aspects to determine their range of applicability.

The related literature is being surveyed after a 2-yr inactivity.

3. General Direct Conversion

a. Liquid-metal Linear Generator. To determine the actual efficiencies of the condensing injector, a key component of a liquid-metal magnetohydrodynamic power-conversion cycle, so that the attractiveness

⁹Brown, R. W., Jr., Analytical and Experimental Studies of Double-pipe Counterflow Liquid-metal Heat Exchangers, Ph.D. Thesis, Notre Dame/Argonne National Laboratory.

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.

(i) Interfacial Transport Processes. To obtain a basic understanding of the heat-, mass-, and momentum-transport processes that occur in the film-flow generator and condensing injector, measurements are being made of local and total condensation rates, film thickness, condensation length, and temperature distribution in the film. Data will be taken over turbulent-laminar and turbulent-turbulent conditions of vapor-liquid flow.

Loop construction has been completed and calibration of the instrumentation is essentially complete. Preliminary tests indicate that the loop is operational.

4. Engineering Mechanics

a. Core Structural Dynamics. In the study of the parallel flow-induced vibration of cylindrical rods, experimental spectral-density curves of rod displacements and surface pressures were utilized to construct first-mode harmonic response functions for the rod, corresponding to several mean axial flow velocities. The results demonstrated an anticipated increase in system damping with increasing mean flow velocity (see Progress Report for July 1967, ANL-7357, pp. 122-124).

System damping is important in that an objective of this study is to predict mean amplitudes of displacement and/or stress. This requires knowledge and understanding of the energy dissipation mechanisms as well as development of mathematical model representations for them. Additionally, if we can control (that is, eliminate or attenuate) the magnitude of the energy contained in the pressure fluctuations, as it tends to increase with increasing flow velocity, then the increase in system damping with increasing flow velocity may be used to decrease vibration amplitude with increasing flow velocity.

To obtain preliminary indications of the variance of system damping with mean flow velocity, consider the first-mode harmonic response functions which were constructed from experimental spectral-density curves, normalized at zero frequency. The magnitude of the peak is equal to $1/R_1$, or the reciprocal of the first-mode damping coefficient R_1 . Values of R_1 have been computed as a function of mean axial flow velocity.

Assume that the overall system damping can be represented as a superposition of an effective viscous damping, corresponding to "no flow," and damping due to flow. Then we can write

$$R_1(\bar{u}) = 2\zeta_1 + f(\bar{u}), \quad (1)$$

where ζ_1 is the effective viscous damping factor at "no flow" and $f(\bar{u})$ is the flow damping function. The effective viscous damping factor at "no flow" was obtained from log decrement tests in water and found to be $\zeta_1 = 14.8 \times 10^{-3}$. This implies $R_1(0) = 0.03$.

Paidoussis' model¹⁰ gives a linear relationship for the flow damping function of the form

$$f(\bar{u}) = 2K_1\bar{u}, \quad (2)$$

where

$$K_1 = \frac{\rho_f d C_f + \pi \rho_f d^2 \int \alpha_1(\bar{x}) \alpha_1''(\bar{x}) d\bar{x}}{4M_1 \omega_1}$$

and ρ_f is fluid mass density, d is cylinder diameter, α_1 is fundamental mode shape, ω_1 is fundamental natural frequency, C_f is friction drag coefficient, and M_1 is generalized mass. The coefficient K_1 has been evaluated for cases of flow and "no flow."

Future work in the damping study will involve (1) obtaining experimental results and correlations over a larger range of flow velocities and (2) developing and evaluating mathematical models representing the energy dissipation mechanisms.

D. Chemistry and Chemical Separations

1. Fluoride Volatility Process

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

(i) Development of Decladding Procedures. Work has begun on a laboratory-scale investigation of the fluid-bed decladding of Zircaloy by chlorine to form volatile ZrCl_4 . The advantages of using chlorine (instead of HCl used in a previously developed process) are:

- (1) Hydrogen is not generated.
- (2) The carbon steel from the fuel-rod fixtures reacts with chlorine to form FeCl_3 , which can be vaporized and thus separated from the spent fuel.
- (3) Efficiency of reagent utilization and decladding rate are relatively high.
- (4) Reagent cost is lower for chlorine than for anhydrous hydrogen chloride.

¹⁰Paidoussis, M. P., Vibration of Flexible Cylinders with Supported Ends, Induced by Axial Flow, Proc. Inst. Mech. Engrs. 180 (Part 3J) (1965-1966).

In the first set of experiments, uranium dioxide and chlorine are being reacted in the temperature range 200-600°C to determine the minimum temperature and minimum partial pressure of chlorine at which UO_2Cl_2 forms. Sintered UO_2 pellets, of 0.5-in. dia by 0.5-in. high and weighing ~16 g, are exposed to a chlorine flow of 250 ml/min for 3 hr at one of the following temperatures: 200, 300, 400, 450, 500, 550, or 600°C. The reaction progress is determined by observing weight changes of the UO_2 pellet and by identifying the physical and chemical properties of any reaction product formed. Experiments up to 500°C have been completed. In the range 400-500°C, surface scaling has been observed, possibly due to UO_2Cl_2 formation. These scales have been submitted for X-ray analysis to reveal their composition.

Two 4-run factorial experiments have been planned to study the influence of chlorine concentration, reaction temperature, and gas velocity on the rate of Zircaloy decladding and on the rate of chlorination of carbon steel springs. These experiments will be carried out in a 1.5-in.-dia fluidized-bed reactor.

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

(i) Engineering Development for UO_2 - PuO_2 Fuel: Engineering-scale Alpha Facility. The major objectives of experiments under way in the engineering-scale alpha facility are to demonstrate PuF_6 handling in engineering-scale equipment and to determine the feasibility of fluoride volatility flowsheets. In run UBr-16, the newly installed BrF_5 feeding system was utilized. Activated alumina and soda lime traps were employed to remove UF_6 - BrF_5 - Br_2 from the fluorinator off-gas stream. (The trapping system used in this facility is intended to allow ready disposal of the UF_6 and bromine fluorides. This method would not, of course, be employed in an industrial operation since it does not permit recovery of UF_6 .) In this run, ~4.4 kg of UO_2 was processed in a bed consisting of (1) ~6200 g of alumina from previous fluorination experiments and (2) ~650 g of -325 mesh alumina that had not been previously exposed to fluorine.

In the oxidation step, the UO_2 was contacted with 11-20 v/o oxygen at 450°C for 8 hr. The resulting U_3O_8 -alumina mixture was next fluorinated for $13\frac{1}{2}$ hr with once-through 10 v/o BrF_5 at 300°C and a gas velocity of 0.65 ft/sec; nearly all of the uranium volatilized from the reactor. The final step, fluorination with ~85 v/o fluorine in nitrogen in the range 300-550°C, was intended mainly to prefluorinate the alumina fines fraction and thereby minimize its reaction with active volatile fluorides when it is used in later experiments.

In the BrF_5 step, two interruptions were caused by blockages of the gas flow, mainly in the off-gas trapping system. This was

corrected by introducing a NaF trap for UF_6 just upstream from the alumina-soda lime system. The sintered nickel fluorinator filters were replaced with new, smaller Huyck sintered nickel fiber filters to initiate testing of this new filter material. The run was then completed without interruptions.

(ii) Engineering Development of Fluid-bed Fluorination

Procedures: Process Development Studies for Uranium Dioxide Fuels.

A series of nine engineering-scale experiments has been completed to determine the effects of six independent variables on the fluorination of UO_2 with BrF_5 (see Progress Reports for November 1966 through July 1967). The variables studied were (1) bed temperature, (2) BrF_5 concentration, (3) fluidization gas velocity, (4) quantity of UO_2 charge, (5) particle size of the inert alumina bed material, and (6) the presence or absence of inactive fission products. In these runs, fragmented UO_2 pellets were oxidized to powdered U_3O_8 in an oxygen-nitrogen gas stream, and then the U_3O_8 fines were fluorinated to volatile UF_6 by means of BrF_5 in nitrogen.

Data are being processed. Results have been obtained on the uranium concentrations in grab samples taken from the bed at the end of the oxidation step. The uranium concentrations in these samples were compared with the concentrations expected if the charge uranium were distributed homogeneously. Grab-sample analyses show that either (1) the use of -100 mesh alumina rather than 48 to 100 mesh alumina in the fluidized bed or (2) the presence rather than the absence of fission products significantly decreases segregation of uranium in the bed and elutriation of uranium from the bed. For the runs with -100 mesh alumina and with fission products present, the uranium concentration in the samples closely represents the calculated concentration of uranium if all uranium charged were distributed homogeneously in the bed. In these runs, the ratio of calculated uranium concentration to measured uranium concentration was about 1.1. A variable that had no significant effect on the homogeneity of uranium distribution in the bed was weight of UO_2 charged (4.4 kg versus 2.2 kg). Since the 4.4-kg UO_2 charge was associated with twice as much alumina as the 2.2-kg charge, it follows that the deeper alumina bed had no effect on the homogeneity of uranium distribution in the bed.

It is necessary to know the plutonium content of the pellets for accountability purposes. A method for plutonium analysis based on measuring the weight of input UO_2 - PuO_2 and the uranium-to-plutonium ratio in bed samples is being considered. In this method, only the uranium and plutonium need be homogeneously mixed; it is not necessary that homogeneity be obtained with respect to other constituents of the bed such as alumina. In this evaluation, U^{235} was used as a stand-in for plutonium.

In each of the nine runs of the series, a mixture of uranium dioxide pellets of two enrichments was used--10% of the UO_2 charge consisted of depleted uranium (0.221% U^{235}) and 90% consisted of normal uranium (0.730% U^{235}). The samples analyzed were taken from the bed at the end of

the oxidation step. Their U^{235} and U^{238} concentrations were determined. The ratio of U^{235} to U^{238} in the final bed samples varied from 6.59×10^{-3} to 6.92×10^{-3} as compared with 6.79×10^{-3} for the calculated mean value. These data suggest that if plutonium behavior is the same as that of U^{235} , the plutonium content of the input pellets can be determined to at least $\pm 2\%$ from the Pu/U ratio of the final grab sample during the oxidation step. These results agree quite well with results obtained in a similar earlier study (see Progress Report for August 1965, ANL-7090, pp. 49-51).

2. Compact Pyrochemical Processes

Studies are being made of a pyrochemical process for ceramic (e.g., oxide or carbide) fast breeder reactor fuels clad with stainless steel. The process applies salt-transport separations and liquid metal-molten salt extractions to the separation of fissile and fertile fuel constituents from the fission products. A conceptual flowsheet for the process has been described previously (see Progress Report for September 1966, ANL-7255, p. 30).

a. Process Chemistry of Molten Salt Systems. A system consisting of a molten $CaCl_2$ -20 m/o CaF_2 salt mixture and a liquid Cu-33 w/o Mg alloy containing calcium as the reductant is being considered for use in the oxide-reduction step of the process. Because the salt in this system becomes a waste stream, it is desirable to decrease its plutonium content to the lowest possible level. Therefore, measurements are being made of the distribution coefficients (ratio of w/o in salt to w/o in metal) of plutonium and selected fission product elements between the salt mixture and the metal alloy. Measurements of the plutonium distribution coefficient were made at 700 and 800°C with calcium contents in the alloy ranging from 0.1 to 6 w/o. The results are presented in Table XXIX along with previous data on the distribution of cerium in the same system (see Progress Report for July 1967, ANL-7357, p. 92). The distribution coefficient of plutonium at 700°C ranged from 2.4×10^{-4} to 2.6×10^{-5} for calcium contents in the alloy of 0.1 to 6 w/o. At 800°C the distribution coefficient ranged from 7.6×10^{-5} to 2.9×10^{-5} for the same calcium contents. These values indicate that for all conditions studied the loss of plutonium by distribution to the reduction salt would be less than 0.01%. This salt-alloy system appears to be very promising for the reduction step of the process.

The distribution coefficients of barium in the $CaCl_2$ - CaF_2 /Cu-Mg-Ca reduction system were determined at temperatures between 775 and 850°C to determine its behavior in the process. The distribution coefficient of barium in this system is approximately 700, with no apparent variation over the temperature range. Thus, as with rubidium (see Progress Report for July 1967, ANL-7357, p. 92), essentially all of the barium appears in the salt phase of the system and would be removed with the waste salt.

TABLE XXIX. Plutonium and Cerium Distributions
between CaCl_2 - CaF_2 Salt and
Cu-33 w/o Mg Alloys Containing Calcium

w/o Ca	Temp ($^{\circ}\text{C}$)	K_d (a)
<u>Plutonium</u>		
0.1	800	7.6×10^{-5}
0.1	700	2.4×10^{-4}
1.0	800	3.6×10^{-5}
1.0	700	8.1×10^{-5}
6.0	800	2.9×10^{-5}
6.0	700	2.6×10^{-5}
<u>Cerium</u>		
1.0	850	1.5×10^{-2}
1.0	825	1.1×10^{-2}
1.0	800	1.0×10^{-2}
1.0	775	0.8×10^{-2}
5.0	850	0.8×10^{-2}
5.0	850	1.0×10^{-2}
5.0	825	0.8×10^{-2}
5.0	800	0.6×10^{-2}
5.0	775	0.5×10^{-2}
5.0	825	1.1×10^{-2}
5.0	800	0.7×10^{-2}
5.0	775	0.5×10^{-2}

(a) Distribution coefficient (K_d) = $\frac{\text{w/o in salt}}{\text{w/o in metal}}$

b. Process Chemistry of Liquid Metal Systems. The solubility of neptunium in liquid cadmium is being studied to investigate possible methods for the recovery of neptunium from fast breeder reactor fuels by pyrochemical processes (see Progress Report for June 1967, ANL-7349, p. 86). The solubility of neptunium in liquid cadmium over the temperature range 330 to 450°C may be represented by the equation: $\log (a/o \text{ Np}) = 6.054 - (4384/T)(^{\circ}\text{K})$. The heat of solution, -20.1 kcal/mole, is intermediate between the heats of solution of uranium and plutonium, which are -20.8 and -19.6 kcal/mole, respectively. The solubility values for neptunium in cadmium and the heat of solution suggest that the solid phase in equilibrium with this solution is NpCd_{11} . The nature of the solid phase will be investigated further by X-ray diffraction measurements.

c. Laboratory-process Development. An experiment was performed to investigate the salt transport of plutonium using the alternate-pour, dual-crucible technique (see Progress Report for February 1967, ANL-7308, p. 37). The plutonium was transported between a Cu-33 w/o Mg

donor alloy and a Zn-6 w/o Mg acceptor alloy via a 50 m/o MgCl_2 -30 m/o NaCl -20 m/o KCl salt mixture. In this experiment the position of the agitator in relation to the salt-metal interface proved to have a strong influence on the rate of approach to equilibrium. When the agitator was positioned at the interface, nearly complete equilibrium was obtained, resulting in a high rate of plutonium transport. Twenty cycles were used; however, in the first eight cycles the agitator was not placed at the interface, resulting in a substantially reduced transfer rate. The experiment was terminated when 99% of the plutonium had been transferred out of the donor alloy. The concentration of plutonium in solution in the acceptor alloy was lower than had been estimated by extrapolation of solubility data obtained at lower temperatures. The information concerning the agitator location and plutonium solubility in Zn-6 w/o Mg will be used to improve subsequent plutonium salt transport experiments.

d. Engineering Development. A system has been constructed for testing continuous liquid metal-molten salt contacting equipment. The system (see Fig. 15) is contained within a large argon-atmosphere enclosure, and is designed so that various types of contacting equipment can be installed in and removed from the system remotely. The equipment components, fabricated of Type 304 stainless steel, are designed to operate at a maximum temperature of 600°C . The components include a centrifugal pump, pump sump tank, constant-head tank, flowrate-measuring tank, and interconnecting piping and heaters.

The salt is pumped from the pump sump tank up to the constant-head tank, and it then flows by gravity through the metal-salt contactors to the flowrate-measuring tank and back to the pump sump tank. The flow of salt is controlled by valves which direct the desired proportion of the total salt flow through the test section of the system. The remainder of the flow returns directly from the constant-head tank to the pump sump tank.

Fig. 15. Schematic Diagram of Continuous Metal-Salt Contactor Test System

Initial experiments performed with the system will have as their objectives the demonstration of continuous uranium transport, the determination of uranium transport rates, and the investigation of the effectiveness of different continuous metal-salt contactor designs. The effectiveness of different designs will be compared with respect to

agitation and phase separation. The contacting equipment that will be tested will be designed so that it can be duplicated later from tungsten or molybdenum-tungsten alloys.

Shakedown testing of the system components has started. The pump sump tank was charged with 200 lb of a 50 m/o MgCl_2 -30 m/o NaCl -20 m/o KCl salt mixture which was circulated through the system at 600°C for 4 hr. All equipment components performed well. Further tests will be carried out to establish the reliability of all components before a metal phase is charged to the metal-salt contactors.

IV. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

1. Planning and Administration

a. Safety Analysis: The Preliminary Safety Analysis Report (PSAR) for AARR was originally submitted to the AEC on April 6, 1966 (see Progress Report for April 1966, ANL-7204, p. 65). Since then, extensive modification has resulted from reviews by AEC and design changes, mainly the change from the original stainless steel core to the High Flux Isotope Reactor (HFIR) core and internals.

It is expected that the first meeting with the AARR subcommittee of the AEC's Advisory Committee on Reactor Safeguards (ACRS) will be held next month and that the full committee will review the AARR in October. The seismic design criteria report has been transmitted to the AEC.

2. Core Research and Development

Because the revised AARR design uses fuel elements, control elements, control-drive mechanisms, and many reactor internal structures identical to their counterparts in HFIR, AARR core studies involve only transient heat-transfer effects and development of structures to support the HFIR-type components in the AARR vessel. The structural design is complicated by new seismic-load criteria and by the precision needed to accommodate AARR's six blind tubes and two through tubes that are to be used for experiments.

a. Heat-transfer Studies. To verify the adequacy of the thermal design of the core for the AARR conditions (4.4-MW/liter peak power density and 2×10^6 Btu/hr/ft² peak heat flux at 100 MW), steady-state safety margins are being verified by analysis of operational malfunction and inadvertent reactivity additions.

(i) Heat-transfer Analysis. The thermal performance of the core is being analyzed with respect to steady-state operation, slow and rapid transients, and shutdown-decay-heat removal and emergency cooling.

The computer code TRANTEMP, being developed for the transient heat-transfer analysis of the hydraulic rabbit, has been found in all major respects to be free of error. The preliminary results for heat-transfer analysis previously reported (see Progress Report for May 1967, ANL-7342, pp. 109-111) are unchanged. Several improvements have been made to provide greater versatility and simplicity when using the code: (a) the flow rate as a function of time can now be entered in tabular form

as input instead of as a special subroutine; (b) the heat-generation rate can be accommodated in the same manner as the flow rate; (c) an option now permits printed output of only a few selected points instead of requiring printed output of all computed results. Another improvement being developed for the code is the inclusion of a subroutine which will more quickly and efficiently determine, by a numerical procedure, the steady-state (initial) temperature profiles for use as initial conditions in any transient calculation. Heretofore, these steady-state profiles were obtained by allowing a pseudotransient problem to converge under constant flow and heat-generation conditions, beginning with arbitrary initial temperatures. Such convergence took up to 10 min of computer time; the new method requires only a few seconds of computer time.

The thermal analyses of core structures and reflector to determine adequacy of design have progressed to the point where calculational models using the THTB code have been set up and made operational on the CDC-3600. The beryllium reflector has been represented by the three separate calculational models, selected to cover as large a region as possible considering the restrictions of the code. The regions considered by each model are depicted in the composite section extending from the core center to the outside diameter of the beryllium. The models represent the actual geometry used in constructing HFIR and specified for AARR. Model A considers the aluminum liner, as well as the inner and center cylinders of beryllium; 80 nodes are used and a symmetrical region was selected. Model B, which covers the outer cylinder of beryllium and the semipermanent beryllium, uses 88 nodes; a symmetrical region was selected, which required moving the designated hole. Model C pertains to the permanent beryllium and uses a total of 250 nodes; a symmetrical region was selected. Preliminary calculations have been made for these models, but the results have not been analyzed in sufficient detail to report at this time.

Two separate models of the reflector shroud and pedestal are being considered in preliminary plans for thermal analysis. These models are vertical cross sections through the gussets that support the beryllium. One model, for the portion of the reflector shroud and pedestal with the long gusset, has been set up and is operating on the CDC-3600. The other model is of the portion with the short gusset.

(ii) Transient Heat-transfer Tests. To provide a means by which the calculated effects of reactivity additions on core heat transfer and hydrodynamics can be compared with a physical model of fuel-plate and coolant behavior, tests are being conducted with an electrically heated flow-channel section in which power is varied with time; voids and pressure pulses during transient boiling are among the effects measured.

The test equipment consists of a high-pressure heat-transfer loop, test section, pressure shroud, and associated instrumentation.

During an experimental run, the power generation will be varied with time to fit a preprogrammed shape while the following measurements are recorded: total power input, axial power-generation shape, average void fraction near the exit plane, inlet and exit water temperatures, inlet and exit pressures, temperature on the outside surface of the test section, and coolant flow.

Assembly of test equipment has been under way for several months. Difficulties with the test section, because of its complicated nature, have delayed actual testing. A third test section has been assembled and is ready for installation into the loop. The preceding two sections failed: the first due to a fracture in the silver-solder joint, and the second because the Sauresien cement used to hold the thermocouple wiring in place intruded into the test-section cavity.

b. Core Structure Development. To verify the fabricability of the structures needed to support the core precisely in relation to the experiment facilities, prototype core components are being fabricated and assembly is being prooftested; hydraulic interactions and redesign to meet seismic-load criteria are being evaluated.

The package of drawings on the upper control-rod drive magnet has been assembled in preparation for fabrication. The specification and a set of preliminary drawings have been prepared for the lower-drive components. Preliminary drawings have been completed for the fuel and shield-support assembly, the vessel adapter, and the reflector shroud and pedestal.

3. Component Development

a. Beam and Through Tube Development. Development work on experimental facilities includes extensive stress analyses, tests of rupture, bending, collapse, and radiation heating of typical beam-tube sections, and prototype tests under simulated operating conditions.

The concept used for cooling the beam tubes has been revised. Previously, with a core utilizing stainless steel, the beam-tube cooling lines were routed from the primary coolant piping to a pair of auxiliary pumps and continued through the pool to the beam-tube nozzles. To avoid a complete loss of flow from failure of an auxiliary pump, a valved bypass line was provided to allow a reversed flow under the action of reactor pressure drop if required. This system allowed the possibility of separate control of beam-tube water temperature and chemistry, but entailed the expense and complexity of auxiliary pumps, instrumentation, and pool penetrations. Furthermore, it required investigation of partial-flow conditions, malfunctions of the bypass-valve, inertial time for flow reversal, and adequacy of the beam-tube cooling-water jacket under the external pressures accompanying the reversed-flow condition. With the adoption of the HFIR

core, this system was no longer required and was replaced by simply connecting the lines from the beam-tube nozzles to the main reactor outlet line at a point within the pool, thus providing "reversed" flow on a normal, continuous basis. This revision in flow direction requires a reanalysis of the stresses in the cooling-water jacket.

An investigation of the structural adequacy of the jacket is under way. The effects of supports of both "dimple" and longitudinal "spline" types separating the jacket and the beam tube are being studied. An approximate analytical solution indicating the effects of such spacers on beam-tube temperatures has been developed. The results for "spline" spacers about 1/4 in. wide and "dimple" spacers of 1/4 in. dia indicate that tube temperatures would be increased by only about 18 and 4°F, respectively, even when it is assumed that the beam-tube surface is completely insulated over the area of contact. The spacer may be used as an integral part of the jacket and in loose contact with the tube. Either of these spacer types, therefore, is acceptable from the standpoint of peak beam-tube temperatures.

4. Instrumentation and Control

To ensure that overall control concepts are properly constituted and to verify and optimize system performance, the performance of reactor instrumentation and control dynamics are being studied for various operating conditions.

a. Pump-induced Pressure Resonances in Primary-coolant System.

A preliminary analysis was made to estimate the potential for the occurrence of significant pressure resonances in the AARR primary-coolant system. The main primary-coolant pumps were assumed to be the excitation source. Impetus for this study arose from consideration of the likelihood of this phenomenon being a major contributor to the premature failure of control-rod guide bearings in the HFIR.

In the analysis a mathematical model represents the hydraulic characteristics of the AARR primary-coolant loop; an appropriate CDC-160A FORTRAN program was used to solve the model equations. It was assumed that the pump impeller vanes cause perturbations in the nominal values of discharge (and suction) pressure and flow rate which vary sinusoidally with time. These perturbations are transmitted through the primary-system coolant.

Briefly stated, the primary-coolant system was modeled as a set of interconnected components consisting of distributed-parameter elements (pipes); and lumped-parameter elements (reactor pressure vessel, core zone, heat exchangers, strainer and pumps). Calculations were made for a wide range of assumed impressed frequencies to determine

multiple-resonance points. Transmission of pressure variations through the system structures was neglected in the analysis.

(i) Description of System Model. A schematic representation of the primary coolant system, as used in these calculations, is depicted in Fig. 16. Convenient electrical-circuit symbolism is used in this diagram because of the direct analogy between the equations which describe the transmission of small-amplitude pressure and flow variations through compressible fluids, and those which describe current and voltage transmission through an electrical line.¹¹

Fig. 16. Schematic Representation of AARR Primary-coolant System

Substantial simplifications have been employed in the representation of the actual primary-system coolant loop. These have largely been made to yield tractable mathematics, but enough detail is retained to provide meaningful results.

Major assumptions made in the development of the mathematical model are: (a) the total time derivative of fluid velocity is equal to its partial time derivative; (b) fluid flow resistance (frictional pressure loss) is a linear function of fluid velocity; (c) local pressure losses due to valves, nozzles, pipe bends, and vessel-inlet strainer are treated as contributors to the distributed hydraulic resistance of each associated pipe section; (d) sections of the pressure vessel above and below the core and the shell sides of the heat exchangers are assumed to be Helmholtz resonators, with appropriate¹² hydraulic capacitance; (e) the hydraulic impedance of the reactor core zone and primary pumps is considered to be purely resistive, and the heat-exchanger complex is assumed to have resistance and capacitance but no inductance; (f) the hydraulic resistance

¹¹Bronwell, A. B., and Beam, R. E., Theory and Application of Microwaves, McGraw-Hill Book Co., Inc., New York and London (1947); Rich, G. R., Hydraulic Transients, McGraw-Hill Book Co., Inc., New York (1951).

¹²Binder, R. C., Advanced Fluid Mechanics, Prentice-Hall, Englewood Cliffs, N. J. (1958).

of the heat exchangers is treated as contributing equally to the distributed resistance of the inlet and outlet piping of the heat-exchanger complex.

(ii) Method of Calculation. The amplitude and phase angle of the pressure fluctuations, relative to their respective values at pump discharge, were calculated at various points within the primary-coolant loop. Primary interest is centered on the pressure-variation amplitude existing within the reactor vessel for an excitation frequency corresponding to design-point operation of the primary pumps.

The overall system impedance, i.e., that existing at the pump discharge, was evaluated by successive system reduction in terms of impedance calculations. Inasmuch as the terminal point of the primary-coolant system is considered to be at the primary pumps, which location is also considered to be the sending point, a closed-loop situation is encountered. Under these conditions, it was assumed that the total discharge fluctuations would be comprised of the pump open-loop output summed vectorially with the fluctuations transmitted through the entire primary system. An iterative procedure starting with the open-loop calculation was attempted. It was found on the first iteration that the feedback signal transmitted through the system back to the pump was negligible, so the open-loop results were considered correct.

Because of the algebraic detail contained in the impedance and pressure equations, and the number of excitation frequencies to be considered, a FORTRAN program for the CDC-160A was written to facilitate the calculations. With this program a complete set of results may be computed for each frequency using ~5 sec of computer time.

(iii) Results. Pressure-fluctuation amplitudes in the reactor-vessel regions, relative to pump-discharge values, are plotted against angular frequencies of excitation in Fig. 17. At locations beyond the vessel, calculated amplitudes are insignificant and thus are not shown. The values of amplitude ratio are plotted in decibel (db) units defined as

$$\text{db} = 20 \log_{10} |P_j/P_p|,$$

where P_j is amplitude of pressure fluctuation at system location j and P_p is amplitude of pressure fluctuation at pump discharge.

Fig. 17. Pressure-fluctuation Amplitude vs. Excitation Frequency

The results represent the entire frequency range considered

and are based on values of system constants appropriate to the current preliminary design of the AARR. Salient features of these results are: (a) resonances occur in the upper and lower reactor-vessel zones at approximately the same frequencies; (b) the first resonance occurs at an angular frequency of ~ 26 rad/sec; the next resonance peak occurs at ~ 170 rad/sec; subsequent resonance peaks occur at angular frequencies of $\sim [170 + n(165)]$ rad/sec for $n = 1, 2, 3, \dots$; (c) the amplitude of the pressure fluctuations in the lower portion of the vessel is more than 40 db (factor of 100) less than that in the upper portion of the vessel for all angular frequencies exceeding 30 rad/sec; (d) for angular frequencies exceeding 170 rad/sec, the envelope of the resonance curves for the upper portion of the vessel has a slope of ~ -1.0 (-20 db per frequency decade); for the lower portion of the vessel, the corresponding slope is ~ -2.0 (-40 db per frequency decade); (e) feedback effects due to transmission through the entire system are completely negligible.

Fig. 18. Pressure-fluctuation Amplitude for Design-point Frequency Range in Pressure-vessel Region above Core

Figure 18 presents data contained in Fig. 17 (for the vessel region above core) limited to and expanded about the excitation frequencies of prime interest. These frequencies are those corresponding to impeller-vane action with the pumps operating at normal rotational speed. At a pump speed of 1760 rpm, the vane frequency will be ~ 921 rad/sec for a 5-vane impeller and ~ 1105 rad/sec for a 6-vane impeller. At these angular frequencies, the system operating points are very

near the minima of the pressure amplitude curves, as indicated in Fig. 18. The corresponding pressure amplitudes are ~ -33 db, so that the vessel pressure variation would be approximately only 2.8×10^{-2} times that existing at pump discharge. Operation at these conditions is highly desirable, but certainly cannot be guaranteed. Small deviations in pump speed and/or location of resonant frequencies would cause an increase in the amplitude of vessel pressure fluctuations; operation at the resonance points results in an amplitude ratio of ~ 0 db, and hence the pressure fluctuation amplitude in the vessel would be approximately equal to that existing at pump discharge.

A few calculations were performed to determine sensitivity of the calculated results to changes in system constants. Only those constants considered to possess large uncertainty factors were varied; changes were made "one-at-a-time," with all other constants assigned their design values. The changes made, effected by multiplying the design values by selected factors, are listed in Table XXX.

TABLE XXX. Changes in System Constants

Case	Description of Constant Changed	Multiplying Factor
1	Primary-pump internal resistance	2
2	Primary-pump internal resistance	0.5
3	Resistance coefficient of all pipes	1.5
4	Resistance coefficient of all pipes	0.667
5	Length of all pipes	1.2

For cases 1 and 2, no discernible change in the calculated pressure variation in the reactor vessel was obtained. This agrees with the statement regarding attenuation of perturbations such that feedback effects at the pumps are negligible.

For cases 3 and 4, there is no significant change in the frequencies at which resonances occur. The amplitude of the pressure variations in the reactor vessel changes discernibly only in a narrow frequency range about the resonance points. The magnitude of these changes is ~ 2 db, which is considered negligible.

Thus, of the changes in system constants considered, only changing the length of the piping system yields a significant change in the calculated pressure variations. The resulting change is mainly in the shift in frequencies at which resonances occur.

(iv) Summary and Conclusion. The major results and conclusions derived from this study are: (a) a potential for significant pump-induced pressure resonances exists in the primary-coolant system of the AARR; (b) under normal operating conditions and for the system design constants used, resonances occur in the pressure vessel at angular frequencies separated by ~ 165 rad/sec; (c) calculated results are essentially insensitive to large changes in nominal values of pump internal resistance and piping flow resistance; (d) in the range of frequencies of primary interest, i.e., those in the neighborhood of the impeller-vane frequencies under normal operation (1760 rpm), the pressure amplitude in the upper portion of the vessel, relative to that at pump discharge, varies between ~ 0 db and -33 db; (e) coolant transmission of pressure pulsations originating at primary-pump discharge is negligible beyond the upper region of the reactor vessel; (f) pressure-pulsation magnitudes in the reactor vessel, originating at the suction side of the primary pumps, are expected to be small relative to those originating at pump discharge.

The result of this analysis, i.e., the possibility of significant pressure pulsations in the upper region of the reactor vessel, is based on preliminary design concepts for major components of the primary coolant loop. Final design will include consideration of minimizing this effect.

V. NUCLEAR SAFETY

A. Fuel Meltdown Studies with TREAT

1. Fast-neutron Hodoscope for TREAT

Work is underway on the fast-neutron hodoscope system (see Progress Report for August 1966, ANL-7249, p. 78) at TREAT to expand the system capabilities and provide digital readout and recording capabilities.

All fundamental elements of the digital readout and recording system have now been tested satisfactorily. In order to provide a time resolution of 1 msec and a recording duration of the order of 20 sec, it has been necessary to introduce new lamps into the digital-display system. Tests of NE-2U neon lamps were made at TREAT. The main advantage of neon lamps is that at high (50%) duty factors they do not show the light output deterioration characteristics of the 6977 vacuum triodes previously used.

Fig. 19. Single Frame from 5000-frame/sec Photographic Record of Readout. Each lamp flash corresponds to a binary bit of a 4096-count scaler; 56 scalars can be read out simultaneously in about 0.2 msec.

With a high-speed rotating-prism camera running at 5000 frames/sec, the results shown in Fig. 19 were obtained. Synchronization was arranged in such a way that a 100 microsecond lamp flash occurred during the proper 68 microsecond interval when the camera shutter was open.

Now that full binary four-bit integrated circuit scalars are available, the peak instantaneous recording capability of the photographic system is over 10^9 bits/sec, which greatly exceeds current magnetic storage techniques. At the same time the cost is much less for photographic recording.

Testing and programming effort is continuing to convert the photofilm record into computer-compatible magnetic tape by means of the CHLOE system.

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

The annular linear-induction pump (ALIP) incorporated into the

design of the Mark-II sodium loop as a replacement for the Faraday conduction electromagnetic pumps used in the Mark-I integral loops has been subjected to further tests in the components test loop. The tests were undertaken to obtain empirical data and to permit a comparison to be made of the performance of various configurations possible with the multipole polyphase pump.

The four-pole version of the ALIP unit was modified by grading the stator coil turns in successive phases over the inlet and outlet poles of the pump. With 60 turns in the largest coils of the pump, the inlet pole coils were wound with 20 turns for the first phase, 40 turns for the second phase, and 60 turns for the third; the coil grading of the outlet pole was a mirror image of that in the inlet pole.

Tests with sodium were performed with the graded 4-pole ALIP using the 3/4-in.-dia solid Armco iron and the laminated soft iron central cores, at temperatures ranging from 400-932°F, and with a 1/2-in.-dia laminated core at a temperature of 400°F.

Data from the component test loop during experimental runs were recorded on punched tape by the DYMEC data-acquisition system for subsequent reduction by computer. Parameters recorded included temperatures, differential pressures, sodium flow in gallons per minute, input power in kilowatts, power conversion efficiency, and Mark-II flowmeter outputs in millivolts and centimeters per second of flow velocity. During all tests, the stator yokes and coils were cooled by forced convection of Dow-Corning 200 silicone fluid.

The grading of inlet and outlet pole coils resulted in a noticeable increase in peak efficiency of the pump. In the graded pump equipped with the 3/4-in. solid core operating at 15 psi pressure and 15 gpm flow with a sodium temperature of 300°F (which is adequate for most of the experiments proposed for the near future), a power input of 1.75 kW was required and yielded an efficiency of 5.67%. The ungraded pump required 3.1 kW and yielded an efficiency of 2.75%. Thus an overall increase of efficiency by more than a factor of 2 was obtained. It is anticipated that grading the inlet pole coils alone, while leaving the outlet pole coils ungraded, to compensate for the rapidly fringing fields resulting from high sodium velocities, will further increase the pump efficiency.

The transition from the solid Armco iron core to the lower-density laminated core yielded no perceptible change in pumping efficiency. This resulted because of a decrease of some 30% in the magnetic volume in the laminated core, which is compensated by a decrease in eddy-current losses. Since the central-core eddy-current losses will be desirable at loop startup to liquefy the sodium in the pump annulus, the ALIP to be used in actual experimental use of the Mark-II loop probably will be fitted with the solid iron core.

Elevation of the sodium temperature to the anticipated pretransient Mark-II loop temperature of 500°C resulted in a decrease in output pressure of approximately 1/3, and a drop in output flow of 1/4. However, the drop in efficiency with temperature lies well within the range of compensation obtained by increasing the power input to the pump.

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

1. Measurements of High-temperature Properties

A method to determine expansion coefficients of ceramics at very high temperatures has been described (see Progress Reports for November 1966, ANL-7279, pp. 92-93, and for May 1967, ANL-7342, pp. 115-116). Since a tantalum carbide crucible is used in measuring the coefficient of expansion for uranium carbide, it is necessary to know as accurately as possible the expansion coefficients of the tantalum carbide and the graphite rod used to measure the distance between the bottom and the cover of the crucible, relative to the surface of the uranium carbide.

The expansion of the ATJ graphite rod could be calculated from the manufacturer's data.¹³ Expansion coefficients of tantalum carbide are given in the literature, but vary considerably. Therefore, this coefficient was repeatedly measured between 900 and 2600°C, with results given in Table XXXI.

TABLE XXXI. Expansion Coefficients of TaC

Temp Range (°C)			Temp Range (°C)		
From	To	α (°C ⁻¹ x 10 ⁶)	From	To	α (°C ⁻¹ x 10 ⁶)
25	500	6.32	20	2380	8.20
25	1000	6.67	0	1000	6.50
25	1500	7.12	0	1200	6.65
25	2000	7.64	?	2625	8.02
25	2500	8.40	25	2400	8.19
25	800	8.20			

Figure 20 shows average values of α from 25°C to temperatures ranging from 850 to about 2650°C. The estimated uncertainty in the measurements is approximately $\pm 7.5 \times 10^{-7}$.

¹³The Industrial Graphite Engineering Book, "Thermal Expansion," Tables 5B.02.01 and 5B.02.02.

Fig. 20. Measured Average Linear Expansion Coefficients of TaC from 25°C

in either an excursion or loss-of-coolant accident and result in an explosive generation of steam. Neither experimental nor theoretical information has been available for the calculation of heat transfer and void formation (pressure generation) that occurs when small particles at very high temperatures are dispersed in water. Thus, a study of forced-convection heat transfer from spheres in water is being carried out.

The transient heat transfer from a heated metal sphere moving through water has many aspects in common with transient heat transfer from a heated metal sphere moving through liquid sodium; studies with sodium were discussed in the Progress Report for July 1967, ANL-7357, pp. 133-137.

In the first transient heat-transfer experiments in water, a heated 1/4-in.-dia nickel sphere was pushed or pulled through the water by means of a gravity-operated swinging arm or pendulum (see Progress Reports for June 1965, ANL-7071, p. 64, and July 1966, ANL-7245, p. 66). The present experiments have been conducted with a motor-driven swinging-arm apparatus utilizing a 1/4-in.-dia silver sphere attached to the end of a 1/16-in.-dia sheathed thermocouple which serves as the arm as well as the temperature-measuring device. The motor-driven swinging-arm apparatus is very similar to that used for the studies in sodium, and allows a greater range in combinations of velocity and immersion time than does a free-swinging pendulum, since the velocity of the sphere is independent of arm length.

The angular velocity of the sphere is determined through the use of a perforated timing wheel attached to the back of the motor shaft. The wheel separates a light source and a photoconductive cell; as the wheel turns, light intermittently falls upon the photo cell and a blip occurs on a recorder chart. This information, together with the arm length (distance from the center of sphere to the center of the axis of rotation), the distance from the axis of rotation to the water surface, and the temperature history

2. Pressure Generation due to Particle-Coolant Energy Transfers: Transient Heat-transfer Studies in Water

In order to analyze adequately the consequences of an incident in a water-cooled thermal reactor in which hot fuel materials may be dispersed into the water coolant, a knowledge of the manner in which large amounts of energy are transferred from these particles to the liquid coolant is needed. Such a process can occur

of the sphere as it moves through the water, is sufficient to calculate the length of the sphere path through the water, the linear velocity of the sphere in the water, the time of the sphere in the water, and the average temperature of the sphere before it enters and after it leaves the water. From these the heat-transfer rate and average temperature of the sphere are calculated.

An example of heat-transfer rates determined from recent experiments is shown in Fig. 21 as a function of the average temperature of the sphere during its pass through the water.

Fig. 21

Heat Flux from Swinging-arm Experiment with 1/4-in.-dia Silver Sphere in Room-temperature Water (Sphere velocity = 13.9 ft/sec)

The heat-transfer rate curves have the classic shape for boiling liquids.¹⁴ The results can be interpreted as indicating the initiation of nucleate boiling at average sphere temperatures of about 125°C, with an increased rate of heat transfer up to about 350°C (for room-temperature water). At this point, transition to film boiling causes a decrease in the

¹⁴Nukiyama, S., Maximum and Minimum Values of Heat Transmission from Metal to Boiling Water under Atmospheric Pressure, J. Soc. of Mech. Eng. (Japan) 36, 367 (June 1934); Farber, E. A., and Scorah, R. L., Heat Transfer to Water Boiling under Pressure, Trans. ASME 70, 369 (May 1948).

heat-transfer rate. The effect of heated water is to lower the temperature of the transition to film boiling (designated as the "burnout" or DNB--Departure from Nucleate Boiling--point).

At temperatures up to about 125°C, the data agree well with an empirical expression developed by Vliet and Leppert¹⁵ that correlates heat transfer by forced convection from spheres to water and air:

$$\text{Nu}(\text{Pr})^{-0.5} = 2.7 + 0.12(\text{Re})^{0.66}, \quad (1)$$

where Nu, Pr, and Re are the Nusselt, Prandtl, and Reynolds numbers, respectively (see ANL-7357, p. 135, for further discussion of this equation as related to heat-transfer studies in sodium).

Values of the heat-transfer rate at the burnout (DNB) point from 1/8-in.-dia tubes to water flowing at a velocity of 11 ft/sec have been determined by Vliet and Leppert.¹⁶ At water temperatures of 22, 43.5, and 60.5°C, they found burnout-heat-transfer rates of 290, 244, and 203 cal/(cm²)(sec), respectively. The corresponding values of burnout-heat-transfer rates we obtained (which are, of course, for spheres) are 365, 220, and 145 cal/(cm²)(sec). This is good agreement considering the different natures of the experimental approaches involved.

A theoretical equation for the heat transfer from a sphere is

$$q/A = 1.13 \left(\frac{U_{\infty} \rho_{\ell} C_{p\ell} k_{\ell}}{D} \right)^{1/2} (T_{\ell} - T_B), \quad (2)$$

where ρ_{ℓ} , $C_{p\ell}$, and k_{ℓ} are the density, specific heat, and thermal conductivity of the liquid, U_{∞} and D are the velocity and diameter of the sphere, and T_{ℓ} and T_B are liquid surface and bulk liquid temperatures, respectively (see Progress Report for November 1966, ANL-7279, p. 90 for application of this equation to heat transfer studies in sodium). This equation represents an analytic solution of the theoretical expression previously developed which considered heat to be transferred across a vapor film from the sphere to the bulk liquid phase by conduction and radiation, with the heat being used partly to form vapor and partly to heat the bulk liquid for the case where the liquid was subcooled. Equation (2) was derived for the special case in which both vapor generation and thermal radiation are negligible. This case corresponds to extensive subcooling in which nearly all of the heat is transferred into the bulk liquid. In the derivation, it was assumed that the flow around the sphere was frictionless and could be described by potential flow theory, and that the distance of penetration of heat into the liquid was small as compared with the sphere radius.

¹⁵Vliet, G. C., and Leppert, G., Forced Convection Heat Transfer from an Isothermal Sphere to Water, Trans. ASME (Series C), J. Heat Transfer **83**, 163 (May 1961).

¹⁶Vliet, G. C., and Leppert, G., Critical Heat Flux for Subcooled Water Flowing Normal to a Cylinder, Trans. ASME (Series C), J. Heat Transfer **86**, 68 (Feb 1964).

A comparison between the experimental data and the theoretical curve from Eq. (2) is included in Fig. 21. The data for room-temperature water fall higher than the theoretical curve at temperatures near the burnout (DNB) point. This indicates that Eq. (2), which assumes a forced-convection process (negligible vapor formation) with superheating of the liquid near the sphere-liquid interface, does not completely account for the observed heat-transfer rates. Apparently the bubble action due to nucleate boiling contributes to the higher rates, although superheating of the liquid probably plays an important role.

Future experiments will involve varied water temperatures and sphere velocities, and perhaps larger spheres. Studies will be aimed at determining the conditions relating to water superheating and vapor formation; the effects of dissolved air will also be examined.

Erratum

In the Reactor Development Program Progress Report for July 1967, ANL-7357, p. 136, Fig. 25, the equation of the solid line in the figure should read:

$$Nu = 2 + 0.39(RePr)^{0.5}$$

C. Reactor Control and Stability

To ensure that large liquid-metal fast breeder reactors can be designed to operate safely over their intended life spans, methods are being developed for foreseeing potential hazards, to provide information for designing systems as free of hazards as possible, and for measuring system dynamical performance.

1. Reactor Stability Design Criteria

To provide stability information and develop guidelines for the safe design of large fast reactors, analytical models are being formulated and refined to account for typical internal feedback effects under actual operational conditions, and the Lyapunov's and other related stability theories are being extended to analyze space-time reactor stability and to develop design guidelines.

For this newly initiated task, a critical evaluation of the various methods now available for studying spatially dependent reactor dynamics has been completed. The objective for this evaluation was to determine which method is most suitable in the analyses of stability and control for the various type of spatially dependent fast reactors. A survey and status report has been completed.

The subsequent phases of this task will include: (1) the formulation of the mathematical model of the LMFBR systems, (2) the application of the extended stability theory of Lyapunov and other related stability theories to the spatially dependent reactor systems, (3) the investigation of finding specific Lyapunov functionals for the LMFBR systems, (4) the establishment of possible preliminary reactor design criteria from the stability analysis, and (5) the comparison of these results with the results obtained from other known methods and computer simulations.

2. Transfer-function Techniques

To determine how methods that have been used to study system stability from transfer-function measurements of small reactors, such as EBR-II, need to be modified for the various geometries of large fast reactors, the definition of spatially dependent transfer functions and the interpretation of the measurements will be studied for various models of large fast reactors.

For this newly initiated task, a survey of theoretical as well as practical concepts and applications of spatially-dependent transfer functions for large reactors has been completed. The objective is to determine the feasibility of applying these concepts in the analyses and measurements of spatially-dependent transfer functions for large fast reactors. A survey and status report has been completed.

The subsequent phases of this task will include (1) the spatial effects due to the presence of various types of detectors and reactivity inputs, (2) the spatial effects due to the relative position of the input and the detection devices, (3) the development of practically feasible experimental techniques, (4) the application of these experimental techniques in the stability and control analyses of spatially dependent reactor systems, and (5) the verification of the method from computer simulations. Finally, experiments will be performed on existing facilities to test the method.

D. 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. Methods can then be developed for predicting critical flow velocities, metastability, superheat, velocity ratios, and flow structure in high-velocity two-phase flow.

Previous theoretical predictions¹⁷ were based on assumptions of thermodynamic equilibrium, which appear to be invalid for critical flow at low qualities ($x < 0.02$). The complexity of the problem in the low-quality region is demonstrated by the expression

$$G_c^2 = \frac{-g_c k}{x \left. \frac{\partial v_g}{\partial P} \right|_{H_0} + v_g \left. \frac{\partial x}{\partial P} \right|_{H_0} - x \frac{v_g}{k} \left. \frac{\partial k}{\partial P} \right|_{H_0}}, \quad (1)$$

where G_c is the critical flow rate in $\text{lb}_m/\text{sec-ft}^2$, g_c is the gravitational constant, $32.17 \text{ lb}_m\text{-ft/lb}_f\text{-sec}^2$, H is stagnation enthalpy in Btu/lb , k is velocity ratio, P is pressure in lb_f/ft^2 , v_g is vapor specific volume in ft^3/lb_m , and x is mixture quality. Equation (1) is the general equation for two-phase critical flow derived by Henry¹⁸ under the assumptions that momentum of liquid $\ll$ momentum of vapor $\rightarrow$ small qualities $x < 0.02$, that the liquid phase is incompressible, and that $v_g \gg v_f \rightarrow$ small pressure ($P < 300 \text{ psia}$) v_f is the liquid specific volume, ft^3/lb_m . Although v_g and $\partial v_g / \partial P|_{H_0}$ can be approximated by their corresponding equilibrium values without introducing significant error, experimental data and analysis indicate that x and $\partial x / \partial P|_{H_0}$ would be in gross error if evaluated as equilibrium quantities.¹⁹ Also, evaluations of k and $\partial k / \partial P|_{H_0}$ as proposed¹⁷ do not appear to agree with experimental observations.²⁰

In view of the above discussion, a new variable N has been introduced to formulate the derivatives in Eq. (1) in terms of equilibrium quantities:

$$N = \frac{x}{x_E k} = \frac{\alpha v_f}{x_E (1 - \alpha) v_g}, \quad (2)$$

¹⁷Fauske, H. K., Contribution to the Theory of Two-phase, One-component Critical Flow, ANL-6633 (1962); Moody, F. J., Maximum Flow Rate of a Single Component, Two-phase Mixture, Trans. ASME, J. of Heat Transfer 87-C, 134 (1965); Levy, S., Prediction of Two-phase Critical Flow Rate, *ibid.*, 53.

¹⁸Henry, R. E., A Study of One- and Two-component Two-phase Critical Flow at Low Qualities, Ph.D. Thesis, University of Notre Dame (1967).

¹⁹Fauske, H. K., Metastable Two-phase Flow of a Saturated Fluid, presented at JSME 1967 Semi-International Symposium, Tokyo, Japan (Sept 1967).

²⁰Fauske, H. K., Two-phase, Two- and One-component Critical Flow, Proc. Symp. on Two-phase Flow (University of Exeter, Devon, England), Vol. 1 (1965).

where x_E is the thermodynamic-equilibrium exit quality. Substituting Eq. (2) into Eq. (1) gives

$$G_c^2 = \frac{-g_c}{N \left[x_E \frac{\partial v_g}{\partial P} \Big|_s + v_g \frac{\partial x_E}{\partial P} \Big|_s \right] + v_g x_E \frac{\partial N}{\partial P} \Big|_{H_0}}, \quad (3)$$

where s is entropy in $\text{Btu}/\text{lb}_m\text{-}^\circ\text{F}$. The quantity N has been investigated experimentally by measuring the axial void profile in the approach to critical flow. In the pressure range 50-150 psia, the measurements show the value of N at the choking condition can be correlated as

$$N = 20x_E \quad (4)$$

and that the derivative can be approximated as

$$\frac{\partial N}{\partial P} \Big|_{H_0} \approx 0. \quad (5)$$

A detailed description of the measurements is given by Henry.¹⁸

By means of the approximation given by Eq. (5), Eq. (3) reduces to

$$G_c^2 = \frac{-g_c}{N \left[x_E \frac{\partial v_g}{\partial P} \Big|_s + v_g \frac{\partial x_E}{\partial P} \Big|_s \right]}. \quad (6)$$

Comparison of Eq. (6) with

$$G_{cHE}^2 = \frac{-g_c}{x_E \frac{\partial v_g}{\partial P} \Big|_s + v_g \frac{\partial x_E}{\partial P} \Big|_s}, \quad (7)$$

obtained for the well-known homogeneous equilibrium model, shows that the only difference between the two formulations is the term N . This suggests that N represents the deviations from equilibrium due to both slip between the phases and metastability.

Combination of Eqs. (4) and (6) gives

$$G_c^2 = \frac{-g_c}{20x_E \left[x_E \frac{\partial v_g}{\partial P} \Big|_s + v_g \frac{\partial x_E}{\partial P} \Big|_s \right]}, \quad (8)$$

which may also be expressed as

$$\frac{G_c}{G_{cHE}} = \sqrt{\frac{1}{20x_E}} \quad (9)$$

Figure 22 shows the predictions of previous investigators and Eq. (8) along with the data reported by Henry.¹⁸

Fig. 22. Comparison between Eq. (8) and Previous Theoretical Predictions¹⁷ and Experimental Data¹⁸

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 influences of pressure, heat flux, dissolved gas, oxide content, surface characteristics, system pressure-temperature history, and ionizing radiation.

Typical sodium-temperature traces from a series of experiments in which only the heat flux was changed are shown in Fig. 23. At the lowest heat fluxes, the boiling is seen to be quite unstable, with repeated periods of free-convection superheating, sudden nucleation, and steady boiling. As the heat flux is increased, the boiling becomes quite stable; once the sodium nucleates, boiling continues indefinitely.

Fig. 23. Behavior of Sodium Temperature during Test Runs; Maximum Previous System Pressure was 1.08 atm at 1700°F.

The effect of system pressure on the transient sodium temperature was also determined; boiling is quite unstable at low pressure but becomes more stable as the pressure is increased. In all cases, pressure transients in the sodium involved considerable oscillation, apparently due to alternate collapse and growth of vapor. The magnitudes of the pressure pulses show considerable scatter, varying from 2 to 100 psi in duplicate runs. This scatter in the pressure is quite typical of that resulting from vapor collapse, also observed in water and other fluids in previous experiments.

It is apparent that sodium boiling is most unstable at low pressures and low heat fluxes but becomes stable as these two quantities are increased. In Fig. 24, the regions of stable and unstable boiling in the heat flux-pressure plane are mapped.

Fig. 24. Heat Flux vs. Pressure Plot, Showing Stable and Unstable Boiling Regions.

assembly of the instrumentation for the eddy-current interface detectors is continuing. Calibration of the sodium-immersed thermocouples has started. General instrumentation is also being calibrated and installed.

3. Electron-bombardment-heater Test Facility

Electron-bombardment-type (EBH) heaters, which simulate reactor-heated fuel-pin clusters of various configurations, are being developed, fabricated, and tested in a large evacuated vessel.

The ion pump has been received and installed. The vacuum chamber has been pumped down to 5×10^{-9} Torr. Several minor modifications are being made to the vacuum system. The fabrication of the first EBH test section is complete and is being installed in the vacuum chamber.

4. 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.

a. Structural Deformation of a Shell. The possible existence of a "characteristic" deformation length associated with the structural deformation of a shell under an impulsive pressure loading is being studied (see Progress Report for July 1967, ANL-7357, pp. 143-144). The cylindrical shell has been approximated by an infinite beam on an elastic foundation, leading to the following equation describing the deflection due to an arbitrary load distribution:

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.

The programmed electron-beam heater has been installed and debugged. Replacements have been ordered for several components that were damaged during transit. Final

$$EI \frac{d^4 y}{dx^4} + ky = p(x), \quad (1)$$

where y is deflection, EI is flexural rigidity, k is elastic constant for the "foundation" (k is EH/r^2 for a shell), h is shell thickness, r is shell radius, and $p(x)$ is distributed pressure loading.

Deflection equations for concentrated loads and uniform loads of finite length are given in the literature. It can be shown that zeroes in deflection do exist and that their location is independent of the magnitude of the load. In fact, for a uniform load of finite length, corresponding to the total integrated load with parabolic decay, the zeroes are approximately 3.86 shell radii apart. This result is of interest since it is close to the deformation length of approximately four radii demonstrated experimentally by Proctor.²¹

Various assumed pressure distributions were considered to represent the spatial load distribution resulting from a point charge. However, it was possible to obtain only a closed form solution for the exponentially decaying distribution:

$$p(x) = 2P_0 e^{-\lambda \frac{x}{r}}. \quad (2)$$

The deflection equation obtained by solving Eq. (1) with the distribution given by Eq. (2) is

$$y(\bar{x}) = \frac{2P_0}{k} \left[\frac{1}{4 + \left(\frac{\lambda}{\bar{r}}\right)^4} \right] \left\{ 4e^{-\lambda \frac{\bar{x}}{\bar{r}}} - 2\left(\frac{\lambda}{\bar{r}}\right) e^{-\bar{x}}(\cos \bar{x} - \sin \bar{x}) + \left(\frac{\lambda}{\bar{r}}\right)^3 e^{-\bar{x}}(\cos \bar{x} + \sin \bar{x}) \right\}, \quad (3)$$

where β^4 is $k/4EI$, $\bar{x}$ is βx , and $\bar{r}$ is βr . Note that the deflection shape is independent of the magnitude of the load P_0 .

With the assumption that the exponentially decaying distribution can be used as a first approximation to the actual distribution, Eq. (3) was studied further. In particular, it can be shown that zeroes in the deflection $y(\bar{x})$ do not exist for $(\lambda/\bar{r})$ less than or equal to 1.0. "Zeroes" do occur for $(\lambda/\bar{r})$ greater than 1.0. In Fig. 25 deformation curves corresponding to several different decay rates can be compared with an experimentally

²¹Proctor, J. F., Explosion Containment for Nuclear Reactor Vessels, Nuclear Safety 7(4), 459-468 (1966).

determined deformation curve²² normalized to one at $\bar{x} = 0$. Figure 25 is intended for qualitative comparisons only since (1) the experimental curve represents a plastic deformation, whereas the theoretical results give an elastic deflection, and (2) the cylinder from which this experimental result was obtained contained a nozzle. From Fig. 25 it can be seen that the theoretical and experimental curves are of the same general shape and that a selected exponentially decaying distribution may be a satisfactory first approximation to the actual load distribution.

Fig. 25. Theoretical Deflection Shapes for Various Load-distribution Decay Rates and Experimental Correlation

Future work involves considering a time- as well as spatial-dependent loading and a plastic analysis.

E. TREAT Operations

Two vibratory-compacted mixed urania-plutonia samples were irradiated for the General Electric Company.

Two metal-water reaction samples of Zircaloy-2-clad, vibratory-compacted urania fuel were irradiated in transparent capsules to obtain additional information on the behavior of this type of fuel under transient conditions which do not cause gross failure of the cladding.

²²Wise, W. R., Jr., and Proctor, J. F., Explosion Containment Laws for Nuclear Reactor Vessels, NOLTR 63-140 (Aug 1965).

A temporary neutron collimator and the lead shielding required to obtain neutron radiographs of individual capsules from EBR-II subassembly XO11 were installed in and adjacent to the reactor shield. The permanent neutron collimator and the additional neutron shielding designed for the TREAT reactor radiography facility will be installed after radiographing the capsules from subassembly XO11. Good-quality neutron radiographs have been obtained of several capsules from subassembly XO11 and one sample of Power Burst Facility fuel previously irradiated in TREAT.

F. Safety Features Technology--Containment

To define safety factors and to reduce the cost of guarding against the consequences of major nuclear accidents, analytical and experimental studies are being conducted of the absorption of blast energy by new structural techniques and the use of analytical techniques of blast propagation through such reactor media to obtain one-dimensional codes for material behavior in the elastic and plastic regions.

1. One-dimensional Code for Analyzing Containment Structures

One-dimensional computer codes are being developed to predict the response of LMFBR primary-containment structures to core excursions ranging in magnitude from a few thousand watt-seconds to hundreds of megawatt-seconds. This work will provide design guidance and a realistic basis for the safety analysis of LMFBRs.

The formal solution of the transient response of a multiregion containment structure to a low-energy core excursion has been obtained by means of finite integral transforms. To obtain numerical results, the corresponding eigenvalue problem must be solved. This shall be accomplished by means of digital and analog computers for a wide range of material parameters.

2. Two-dimensional Code for Analyzing Containment Structure

The objective and scope of this task are the same as for the one-dimensional case, but the complexity of the problem is increased considerably.

Although a one-dimensional computer code provides physical insight, the results hold quantitatively for only spherically symmetric excursions.

To consider other geometries, the computer code must be extended to include two spatial dimensions, such as an axisymmetric core excursion enclosed by a finite, right circular cylindrical containment structure. To

study the latter, a code is to be developed to treat the propagation of axisymmetric shock waves through the core and the containment structure by means of hydrodynamic theory. As a first approximation, the medium under consideration is treated as a compressible fluid. The differential equations of shock hydrodynamics will be approximated by finite difference equations. The von Neumann-Richtmeyer artificial viscosity method will be used to treat shock discontinuities.

The choice of different coordinates has been studied. In one dimension, the use of Lagrangian coordinates has the advantages of (1) resulting to a simpler form of differential equations and (2) providing a more convenient way of numerical integration. But this is not true in two dimensions, in which the differential equations in Lagrangian coordinates become very complex. It seems to offer no advantages over the use of the Eulerian coordinates. However, because of the interfaces, which are inherent in the multiregion geometries, the Lagrangian coordinates have been chosen despite their complexity.

The formal solution for the elastic case has been obtained by means of finite integral transforms for symmetric and asymmetric internal pressure pulses. Solution of the corresponding eigenvalue problems are necessary for the numerical evaluation of the formal solutions. Again, this shall be accomplished by means of digital and analog computers.

VI. PUBLICATIONS

Papers

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Partition of Metals between Liquid Metal Solutions and Fused Salts

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P. A. Nelson, D. K. Butler, M. G. Chasanov, and D. Meneghetti

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M. Petrick and J. J. Roberts

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Decay Constant for the Spontaneous Fission of U^{238}

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ANL Reports

ANL-7270

ANALYSIS OF HIGH CONVERSION CRITICAL
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E. M. Pennington

ANL-7326

RABBLE, A Program for Computation of Resonance
Absorption in Multiregion Reactor Cells

P. H. Kier and A. A. Robba