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Argonne National Laboratory

**REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT**

July 1968

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ANL-7478

Reactor Technology
(TID-4500)

AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY

9700 South Cass Avenue
Argonne, Illinois 60439

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

July 1968

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Issued August 28, 1968

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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 accordance with budget activities in a way which, it is hoped, gives the clearest, most logical overall view of progress. Since the intent is to report only items of significant progress, not all activities are reported each month. In order to issue this report as soon as possible after the end of the month editorial work must necessarily be limited. Also, since this is an informal progress report, the results and data presented should be understood to be preliminary and subject to change unless otherwise stated.

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

The last six reports issued
in this series are:

January 1968	ANL-7419
February 1968	ANL-7427
March 1968	ANL-7438
April 1968	ANL-7445
May 1968	ANL-7457
June 1968	ANL-7460

REACTOR DEVELOPMENT PROGRAM

Highlights of Project Activities for July 1968

EBR-II

The reactor was operated for 543 MWdt in Runs 29A, B, C, and D in July. The primary purpose was to conduct physics experiments related to the exchange of depleted uranium for stainless steel in Rows 7 and 8. Run 29A was conducted with the stainless steel still in place; 29B was conducted with nearly all the stainless steel subassemblies of Row 7 replaced by depleted uranium; and 29C was conducted with all the stainless steel subassemblies of Rows 7 and 8 replaced by depleted uranium. Run 29D is a continuation of power operation with the depleted-uranium blanket. The cumulated total of EBR-II operation is 18,588 MWdt.

Production of driver fuel in the FCF hot line was limited because of high silicon analyses in a few melt-refining runs and low yields in injection-casting runs. Production continued normally in the cold line.

ZPR-3

In a continuation of the FTR Phase-B critical program, measurements were completed with Assemblies 52D and 52E, and loading was begun for 52F. These variants of Assembly 52 have central sodium regions of different sizes.

ZPPR

Leak testing of seal doors neared completion. Cabling for experiments was connected through the seal pit. The autorod drive and the radial traverse drive were completed and are being tested.

Plutonium and depleted uranium for the initial core were received and stored.

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I. LIQUID METAL FAST BREEDER REACTORS--CIVILIAN

A. Fuel Development--LMFBR

1. Metallic

a. Fuel Element Performance (W. F. Murphy and R. Natesh)

Last Reported: ANL-7460, p. 1 (June 1968).

A new shielded electron microprobe* and a pneumatic transfer system are being installed in conjunction with the Alpha-Gamma Hot-cell Facility in the Metallurgy Division of ANL. The probe will be connected to the hot cell by a pneumatic transfer system capable of handling normal metallographic-specimen mounts. The probe is shielded for 100 Ci of fission-product activity and will be able to handle uranium-plutonium-bearing fuels. A specimen-transfer facility (a shielded compartment for the transfer of a specimen from the pneumatic system to the probe) is also being installed.

2. Carbide

a. Fabrication and Evaluation

(i) Fuel Element Performance (L. A. Neimark)

Last Reported: ANL-7460, p. 2 (June 1968).

Irradiation of twelve elements in Subassemblies XA08 (XO08) and XO15 continued in EBR-II. As of July 11, the burnup increment in addition to that previously reported (see Table I.A.1 of ANL-7460) was 0.1 to 0.3 a/o, which raises the maximum burnup in Subassembly XA08 to 7.2 a/o and in XO15 to 3.6 a/o.

3. Fuel Cladding and Structure--Jacket Alloys

a. Fabrication Development of Fuel Jacket Alloys (N. J. Carson)

Last Reported: ANL-7457, p. 10 (May 1968).

Argonne National Laboratory is participating in a program to compare the fabricability of several vanadium alloys, and to assemble a stockpile of vanadium alloys of known history and composition from which material may be withdrawn by other AEC investigators. Five primary-extrusion billets were assembled by Westinghouse personnel for primary and secondary extrusion at ANL. Four of the billets have arrived at ANL.

*Materials Analysis Corporation Model 450.

The first billet was a composite with a core of V-20 w/o Ti alloy and a smaller section of VANSTAR-8, an alloy developed by Westinghouse. This billet was primary extruded successfully at 1100°C through a reduction of 3.9 to 1. The primary extrusion was cut into seven sections, each of which provided a core for a secondary-extrusion billet. The seven secondary billets were extruded into sheet bar or rod stock at 1100°C. Although both V-20 w/o Ti and VANSTAR-8 alloys extruded well, the VANSTAR-8 alloy required a higher pressure. Both alloys possessed very good surfaces after secondary extrusion (the surface of VANSTAR-8 was in slightly better condition).

A V-15 w/o Cr-5 w/o Ti billet (an alloy developed at ANL) was successfully extruded to the same primary reduction at 1150°C. This extruded billet has been sectioned, and the seven billets are being prepared for secondary extrusion.

B. Physics Development--LMFBR

1. Theoretical Reactor Physics

a. General Fast Reactor Physics (H. H. Hummel)

(i) Multigroup Methods

Last Reported: ANL-7403, pp. 14-15 (December 1967).

(a) Sensitivities of Calculated Interface Currents upon Collapsing of Multigroup Cross Sections in Regions Containing Resonance-scattering Materials. Detailed neutron-flux spectra and fine-flux-weighted coarse-group cross sections for fundamental-mode systems containing resonance scattering materials may be calculated with the Argonne Program ELMOE.* The possible presence of resonance-scattering materials in regions for which the assumption of a fundamental-mode flux may not be satisfactory, for example in reflectors, has led to studies directed toward determining coarse-group cross sections for such regions.

The following reports results of use of coarse-group cross sections which are position- or subregion-dependent. These spatially dependent coarse-group cross sections are obtained by fine-group-flux and fine-group-current collapsing using directly calculated fine-group spatially dependent solutions obtained by use of the modified ELMOE-SNARG composite programs** in the S_2 approximation. Current-weighting is used for energy and spatial averaging of the transport cross sections.

*Rago, A. L., and Hummel, H. H., ANL-7170 (1966).

** Meneghetti, D., Rago, A. L., and Phillip, K. E., "Use of the ELMOE-SNARG Programs to Study Details of Neutron Spectra Through Regions of Resonance-Scattering Media," Resúmenes de los Artículos Presentados en la Conferencia Internacional sobre Utilización de Reactores de Investigación y Computo y Matemáticas de Reactores, Mexico City (1967).

A measure of the effect of employing subregional-dependent cross sections for interface cases is to be found in the comparison of the calculated coarse-group leakages across an interface with the sum of the corresponding directly calculated fine-group leakages. For the interface situation, a two-component system consisting of a slab of aluminum of 30-cm half-thickness as the resonance-scattering material containing an applied flat (inhomogeneous) fission source, and abutted 30-cm-thick iron reflector regions on either side, the directly calculated leakages across the aluminum-iron interface for the indicated coarse-energy intervals are listed in Table I.B.1. For comparison, one, two, and three coarse-group cross-section sets were used in each region. With one cross-section set in each region, the fine-group-weighting solutions were those at the center of the aluminum core region and at the center of an iron reflector region. Two or three cross-section sets in each region indicate use of energy spatial averages over two or three subregions in each region, respectively. Also tabulated is the result of using 50 cross-section sets, i.e., a separate collapsed coarse set at each of the mesh intervals from the center of the aluminum region to the outer boundary of one of the iron reflectors. The spatial mesh was much finer nearer the interface.

TABLE I.B.1. Core Leakage: Aluminum-diluent Core and Iron Reflector

Coarse Energy Group	E_{lower} (MeV)	Fine- Group Calculation ^a	Coarse-group Calculation/ Fine-group Calculation			
			2 Sets ^b	4 Sets ^c	6 Sets ^d	50 Sets ^e
8	0.11-0.18	0.000556	5.89	1.025	1.890	0.990
9	0.067	0.0165	0.840	0.974	0.982	1.000
10	0.0407	0.00926	1.463	1.053	1.014	1.000
11	0.025	0.00776	1.520	1.118	1.115	1.000
12	0.015	-0.0159	1.152	1.097	1.069	1.000

^aSummed from directly calculated fine-group spatially dependent solution employing about 29 fine groups per coarse group.

^bOne coarse cross-section set in each region. Interface at 30 cm.

^cTwo coarse cross-section sets in each region. Subregional outer boundaries are 27 and 30 cm in core region, and 42 and 60 cm in reflector region.

^dThree coarse cross-section sets in each region. Subregional outer boundaries are 18, 27, and 30 cm in core region, and 33, 42, and 60 cm in reflector region.

^eUsing 50 coarse cross-section sets at each of the 50 mesh intervals of the calculation.

Results for a carbon-diluent core region with a nickel reflector are listed in Table I.B.2.

TABLE I.B.2. Core Leakage: Carbon-diluent Core and Nickel Reflector

Coarse Energy Group	E_{lower} (MeV)	Fine-group Calculation ^a	Coarse-group Calculation/ Fine-group Calculation			
			2 Sets ^b	4 Sets ^c	6 Sets ^d	50 Sets ^e
6	0.3-0.5	0.00309	0.312	0.930	0.946	1.000
7	0.18	0.00521	1.016	0.978	1.016	1.000
8	0.11	0.000217	10.7	1.595	1.220	0.992
9	0.067	0.00856	0.591	0.937	0.952	1.000
10	0.0407	-0.00221	-4.37	0.495	0.498	1.005
11	0.025	0.00669	0.985	0.940	0.947	1.001
12	0.015	0.00906	0.740	0.841	0.845	1.000

See footnotes to Table I.B.1 on p. 3.

Results for a ^{238}U -diluent core region with an aluminum reflector are listed in Table I.B.3.

TABLE I.B.3. Core Leakage: ^{238}U -diluent Core and Aluminum Reflector

Coarse Energy Group	E_{lower} (MeV)	Fine-group Calculation ^a	Coarse-group Calculation/ Fine-group Calculation			
			2 Sets ^b	4 Sets ^c	6 Sets ^d	50 Sets ^e
8	0.11-0.18	0.00164	2.64	1.274	1.083	0.999
9	0.067	-0.00824	0.856	0.953	0.992	1.000
10	0.0407	-0.00287	2.26	1.350	1.126	1.001
11	0.025	-0.0102	0.750	0.960	0.983	1.001
12	0.015	-0.00247	1.963	1.196	1.070	1.001

See footnotes to Table I.B.1 on p. 3.

Results for a PuO_2 -fueled core region containing sodium coolant and steel structure and having an abutting sodium region, representing a sodium plenum, are given in Table I.B.4.

TABLE I.B.4. Core Leakage: PuO₂-fueled Core and Sodium Reflector

Coarse Energy Group	E _{lower} (MeV)	Fine- group Calculation ^a	Coarse-group Calculation/ Fine-group Calculation			
			2 Sets ^b	4 Sets ^c	6 Sets ^d	50 Sets ^e
8	0.11-0.18	0.0308	0.962	0.992	0.996	1.000
9	0.067	0.0196	0.997	1.002	1.000	1.000
10	0.0407	0.0150	1.132	1.024	1.010	1.000
11	0.025	0.00876	0.686	0.945	0.971	1.000
12	0.015	0.00971	1.195	1.034	1.020	1.000
13	0.0091	0.00619	0.898	0.970	0.988	1.000
14	0.0055	0.00170	0.877	0.983	0.994	1.000
15	0.0021	0.00214	1.482	1.070	1.057	1.000

See footnotes to Table I.B.1 on p. 3.

2. Experimental Reactor Physics

a. Fast Critical Experiments--Experimental Support (Idaho) (W. G. Davey)

Last Reported: ANL-7460, pp. 12-13 (June 1968).

(i) Neutron Spectrometry

(a) High-count-rate Electronics. Construction of a pulse rejection system which will further improve the count-rate capability of the proton-recoil neutron spectrometer has started. The system will consist of a charge-sensitive preamplifier, a main-shaping amplifier (which is pole zero corrected), pulse-rejection circuits, a linear gate, an active dc restorer, a live-time clock interface, and a coincidence gate division circuit. Any events which occur during the time the baseline of the main amplifier is displaced will be rejected. Dead times created by the system will be accounted for through an interface to the live-time clock.

The dc restorer and preamplifier have been completed, and circuit board layouts are being made for the shaping amplifier, and rejection electronics.

(ii) On-line Computer Applications. Technical evaluations of proposals for the ZPPR computer system have been made, and an order for the system has been placed.

The ZPPR fuel-inventory program for the CDC-1604 is in operation for the ZPPR fuel now being received. In all, 3613 fuel plates have been received, and their storage locations are tabulated on the ZPPR Fuel Inventory Magnetic Tape.

(iii) Integral Studies of Cross Sections. A standardized prescription for the benchmark analyses of ZPR-3 Assemblies 48 and 11 using ENDF/B was drawn up and submitted to members of the Cross Section Evaluation Working Group. This "Phase-IIB" integral data testing is to be carried out with a reissued version of ENDF/B which includes final corrections arising from Phase-I data-testing efforts. Standardized calculational techniques are called for in the analyses in order to enable a meaningful, comparative review of the results from the different establishments. Thereby, errors in cross-section data and in the various processing codes can be detected.

(iv) Experimental Techniques

(a) Pulsed-source Experiments. The initial phase of the data analysis of the developmental pulsed-source experiments done with ZPR-3 Assembly 51 has been completed. The die-away curves have been fitted with functions of the form $\phi(t) = A_1 + A_2 e^{-\lambda t}$. In some cases, the data were fitted to the sum of two exponentials plus background. The values of the fitted, single-exponential decay constants obtained from the detector in the core, and from the detector in the inner reflector containing nickel and sodium, agree to within experimental error for all degrees of sub-criticality. However, the decay constants obtained from the detector positioned in the outer, iron reflector are consistently smaller than the values obtained from the other detector positions. The reasons for this apparent discrepancy are being investigated.

The ratios A_1/A_2 for all data runs have been formed and indicate that the background from ^{240}Pu spontaneous fission is not a serious problem and does not interfere with the conduct of the experiments. Close to critical, where the pulsed-source experiments are more difficult, the ^{240}Pu source is considerably smaller than the fission source, and consequently the limiting condition will be the normal fission-neutron background.

The justification for a two-exponential fit to the die-away data is being considered.

b. Fast Critical Experiments--Experimental Support (Argonne)

(i) Detector Development (R. Gold)

Last Reported: ANL-7382, p. 92 (September 1967).

(a) Spatially Continuous Neutron-flux Plotting with Spark Chambers. Wire spark chambers yielding signals from which the spark location may be inferred within 0.1 cm have been recently developed for high-energy physics experiments.* With the addition of a fission foil, a single wire counter of this general type may be incorporated in a long tube of modest diameter, suitable for insertion in an instrument channel of an available reactor or critical assembly, and used to obtain the neutron-flux profile with continuous spatial resolution.** In addition to the advantages of resolution and speed, this device is relatively temperature- and radiation-insensitive, and may thus be permanently sited to allow repeated measurements of flux profile.

The Fission foil is self-calibrating: as shown by Bowman,[†] alpha tracks due to the natural alpha activity of the fissionable coating may be registered simply by increasing the applied voltage. The space resolution obtainable with such an instrument can be kept to a few millimeters by optimizing the geometry, such that only fission fragments emitted within a certain small angle to the foil normal are registered. The deionization time required after each spark limits the maximum count rate to the order of $10^3/\text{sec}$.

c. Planning and Evaluation of FFTF Critical Assembly Experiments (D. Meneghetti)

Last Reported: ANL-7460, pp. 28-29 (June 1968).

(i) FFTF Critical Assembly Planning and Evaluation

(a) Calculated Spatial Variations of ^{239}Pu , ^{238}U , and ^{10}B Detector Responses for ZPR-3 Assembly 51. The spatial variations of ^{239}Pu fission, ^{235}U fission, and ^{10}B capture detector responses for ZPR-3 Assembly 51 have been calculated. The 29-energy-group cross-section Set 29001 in one-dimensional diffusion theory was used. Cylindrical geometry with axial leakage buckling was used for radial calculations,

* Gianelli, G., Nucl. Instr. Methods 31, 79 (1964); Charpak, G., Favier, I., and Massonnet, L., ibid., 24, 501 (1963).

** Porges, K. G., Managan, W. W., and Kaiser, W. C., Paper B-15, Second Conference on Neutron Cross Sections and Technology, March 4-7, 1968, Washington, D.C.

[†] Bowman, C. D., and Hill, R. W., Nucl. Instr. Methods 24, 213 (1963).

whereas slab geometry with radial leakage buckling was used for axial calculations. The 29-group detector cross sections used in the core region were obtained from fine-group normal-mode flux averaging for the core composition. Those used in the reflector region were obtained from fine-group zero-buckling fine-flux calculations, using the inner radial-reflector composition and the axial-reflector composition, respectively.

The calculated results are compared with the experimental values (see Progress Report for May 1968, ANL-7457, pp. 20-28) in Figs. I.B.1 and I.B.2.

Fig. I.B.1. Calculated Radial Detector Responses Compared with Experiment

Fig. I.B.2. Calculated Axial Detector Responses Compared with Experiment

(b) Calculated Spatially Dependent Fine-energy Neutron Spectrum for ZPR-3 Assembly 51. Results of neutron spectrum measurements in core and reflector regions of ZPR-3 Assembly 51, the first of a series of critical assemblies in support of the Fast Flux Test Facility (FFTF), using a proton-recoil neutron spectrometer have been reported by Powell.*

*Powell, J. E., Fast-neutron Spectrum Measurements in the Core and Reflector of ZPR-3 Assembly 51, Trans. Am. Nucl. Soc. 11, 217 (1968).

The spatially dependent fine-energy-detailed neutron-flux spectrum using 371 equal-lethargy energy groups in the energy range from 2.1 keV to 1.35 MeV has been calculated by use of the composite* of the ELMOE library of angular scattering data and the Argonne discrete S_N transport code. The one-dimensional cylindrical-geometry S_2 analysis was an inhomogeneous calculation and used the transport approximation. The inhomogeneous calculation utilized an estimated spatially distributed fission source in the core in order to circumvent the need for source iteration. By using fine-energy leakage-absorptions, axial leakage effects were included.

The assembly had a plutonium-fueled core and a two-zone radial reflector. The inner and outer radial reflectors consisted of nickel, sodium, and steel-matrix and iron and steel-matrix, respectively. The cylindricalized core radius used in the analysis was 28.1 cm. The outer radii of the radial reflectors were 47.6 and 64.0 cm, respectively.

Both high-resolution calculated spectra and calculated spectra through a 15% resolution Gaussian energy-window to simulate the experimental measurements at positions approximately corresponding to the four experimental positions reported by Powell are shown in Fig. I.B.3. The calculated positions are at the core center, at 11.2, at 25.2 (near the core-reflector interface), and at 36.2 cm (near the center of the nickel-sodium reflector). For comparison, the experimental data are also shown. The calculated and experimental values have been normalized at the core center for best fit over the energy range of interest. Since a coarse energy structure was used above 1.35 MeV (lethargy of 2.0), the lethargy region corresponding to the energy just below 1.35 MeV contains distortion of the envelope of the flux shape, which tends to make the calculated flux too low near that energy interface. Structures in the spectra are primarily due to Fe, O, Cr, Ni, and Na in the core and to Ni, Na, and Fe in the reflector. The calculated spectrum at the core center is very similar to the fundamental-mode spectrum shown by Powell.

Additional work in the FTR Critical Experiment Program follows under ZPR-3 Operations and Analysis.

3. ZPR-3 Operations and Analysis (W. G. Davey and R. L. McVean)

Last Reported: ANL-7460, pp. 15-22 (June 1968).

a. Mockup Studies. Experiments continued with Assembly 52, the second core of the FTR Phase-B critical program. Assemblies 52D and 52E were completed, and Assembly 52F is now being loaded into the reactor. Analysis of Assembly 51 experiments also continued.

*Meneghetti, D., Rago, A. L., and Phillips, K. E., op. cit. (see page 2).

Fig. I.B.3. Calculated Position Dependent Fine Spectrum Compared with Experiment

(i) Assembly-52 Results. The Assembly-52 data presented here consist of the critical masses of Assemblies 52D and 52E, and the ^{239}Pu reaction-rate traverses in Assembly 52E.

(a) Critical Masses of Assemblies 52D and 52E. Assembly 52D contained a 3- by 11-drawer sodium zone interior to the core material, and Assembly 52E had a 3- by 15-drawer sodium zone which went completely across the diameter of the core region. In both cases the zone extended through the upper and lower axial reflector regions. The reactor loadings which were critical for Assemblies 52D and 52E are shown in Figs. I.B.4 and I.B.5. With the exception of the control-rod positions, both loadings were symmetrical about the matrix interface. Table I.B.5 gives the fissile inventories for these two loadings. The exterior profile of the sodium zone plus core-material zone for 52E approximated a circle with the P-16 matrix position as the center.

Fig. I.B.4. Reference Loading for Assembly 52D

Fig. I.B.5. Reference Loading for Assembly 52E

TABLE I.B.5. Fissile Inventories for Assembly-52D and -52E Reference Cores

Type Drawer	No.	52D Loading 52-49 $^{239}\text{Pu} + ^{241}\text{Pu} + ^{235}\text{U}$ (kg)(b)	No.	52E Loading 52-55 $^{239}\text{Pu} + ^{241}\text{Pu} + ^{235}\text{U}$ (kg)(b)
A(a)	155	161.98	155	161.98
A*(a)	119	94.96	127	101.35
Control (1-U-13)	1	0.79	1	0.79
Control (2-K-14)	1	1.06	1	1.06
Safety	8	17.23	8	17.23
Total		276.02		282.41

(a) A--drawers in even-numbered matrix columns.

A*--drawers in odd-numbered matrix columns.

(b) Includes material in control drawers which are partially inserted in the core.

The critical masses of $^{239}\text{Pu} + ^{241}\text{Pu} + ^{235}\text{U}$ for the two cores were determined to be 275.2 ± 0.5 kg and 282.0 ± 0.5 kg for 52D and 52E, respectively. A new composition for the safety drawers for these assemblies is given in Table I.B.6. The cores are smaller than one would expect from earlier measurements with Assemblies 52A, B, and C, because of a 34% increase in fissile mass contained in the eight safety drawers.

TABLE I.B.6. Composition of Safety Drawers
for Assemblies 52D and 52E

Material	Atom Density ($\times 10^{-24} \text{ cm}^3$)	Material	Atom Density ($\times 10^{-24} \text{ cm}^3$)
$^{239,241}\text{Pu}$	0.004061	O	0.002748
$^{240,242}\text{Pu}$	0.000380	Al	0.000137
^{235}U	0.000032	Fe	0.016227
^{238}U	0.014635	Cr	0.004037
Mo	0.000796	Ni	0.001767
Na	0.004039	Mn	0.000169
C	0.000916	Si	0.000198

The worth of the increased spiking of the safety drawers, above that for Assemblies 52A, B, and C, was measured to be 341.7 lh for two opposing safety drawers in Assembly 52E. The total worth of the spiking for the same two drawers, above that of a normal drawer loading, was measured to be 663.7 lh. The uncertainty in these values is estimated to be $\pm 1\%$. The increased worths for all eight safety drawers were obtained by multiplying these values by four: Thus, the worth of the increased spiking of all eight drawers, above that for Assemblies 52A, B, and C, was taken as 1367 ± 13 lh, and the total worth of the spiking for eight drawers relative to a normal drawer loading was taken as 2655 ± 26 lh. By use of an estimated edge worth of 75 lh/kg, the total spiking correction in Assembly 52E was about 35 kg at the core edge. The correction for Assembly 52D was probably close to this value, because the critical masses and volumes of the two cores did not differ greatly. This estimated number should be considered to be uncertain by about 25%.

(b) Reaction-rate Traverses for Assembly 52E. Reaction-rate traverses were made with Assembly 52E (which had a 3- by 15-drawer sodium zone) with a ^{239}Pu fission counter. The counter was a brass, gas-filled assembly.* It was placed in a stainless steel "boat," 23 in. long, which was traversed down a 5/8-in.-OD stainless steel tube. The boat was a 1/2-in.-OD tube cut down to leave three 1-in. flanges for holding counters in position, and a 1/2-in. strip along the bottom. This was connected by a 1/4-in. stainless steel tube to the traversing mechanism--a mechanical drive, remotely operated from the control room, which provides a travel of 0 to 65.30 in. with a mechanical readout accuracy of about ± 0.03 in.

*See Progress Report for May 1968, ANL-7457, p. 23, Fig. II.B.8.

Three traverses were made axially along the P-16 matrix position and horizontally $1\frac{1}{4}$ in. from the reactor midplane in Half 1 in both the L and P rows. The loading in the core drawers through which the radial traverses were made was the same as for the Assembly-51 radial traverses.* The drawers in the central gap were filled with sodium, except for a $1\frac{1}{2}$ - by 2-in. cavity through which the stainless steel guide tube passed. All other core and test region drawers had their normal loadings. A power level of 3 W was maintained by the autorod during the traverses. The detailed data for each traverse are given in Tables I.B.7 through I.B.9.

TABLE I.B.7. ^{239}Pu Axial Traverse through P-16, Assembly 52E

Counter: ^{239}Pu #R
 Power: 3 W (controlled by autorod)
 Indexer: When center of active area of counter
 (1 in. from counter end) was at core
 centerline, indexer read 35.80

Indexer Position (in.)	Counts/30 sec	Indexer Position (in.)	Counts/30 sec
5.93	33,082	35.93	115,083
6.93	40,626	36.93	114,279
7.93	49,194	37.93	114,333
8.93	57,166	38.93	113,283
9.93	65,193	39.93	111,338
10.93	72,575	40.93	109,867
11.93	78,205	41.93	108,477
12.93	84,348	42.93	105,582
13.93	87,464	43.93	103,638
14.93	90,289	44.93	102,081
15.93	92,660	45.93	99,544
16.93	93,932	46.93	97,162
17.93	93,097	47.93	95,307
18.93	93,305	48.93	93,997
19.93	92,592	49.93	92,662
20.93	92,709	50.93	92,734
21.93	93,104	51.93	93,439
22.93	94,145	52.93	93,901
23.93	96,490	53.93	93,990
24.93	98,412	54.93	93,575
25.93	99,686	55.93	91,917
26.93	101,968	56.93	89,858
27.93	104,166	57.93	86,432
28.93	106,192	58.93	82,001
29.93	108,288	59.93	76,345
30.93	110,483	60.93	69,946
31.93	111,182	61.93	62,397
32.93	113,311	62.93	54,329
33.93	113,951	63.93	45,634
34.93	114,655	64.93	38,027

*See Progress Report for May 1968, ANL-7457, p. 24, Fig. II.B.10.

TABLE I.B.8. ^{239}Pu Radial Traverse in P-row, Assembly 52ECounter: ^{239}Pu #R

Power: 3 W (controlled by autorod)

Indexer: When center of active area of counter
(1 in. from counter end) was at core
centerline, indexer read 38.12

Indexer Position (in.)	Counts/30 sec	Indexer Position (in.)	Counts/30 sec
6.30	10,609	31.30	97,660
7.30	11,224	32.30	99,206
8.30	12,439	33.30	100,048
9.30	16,635	33.50	100,435
10.30	23,608	33.70	101,563
11.30	34,628	33.90	101,761
12.30	48,585	34.10	102,365
12.80	56,183	34.30	104,667
13.30	63,474	34.50	104,862
13.80	70,765	35.30	107,586
14.30	85,683	36.30	109,835
15.30	91,958	37.30	110,854
16.30	103,338	38.30	111,257
17.30	112,326	39.30	110,831
17.80	114,758	40.30	109,960
18.30	117,208	41.30	107,042
18.80	116,485	41.80	103,274
19.30	115,420	42.30	102,280
19.80	111,593	42.80	100,777
20.30	107,291	43.30	100,744
20.80	100,869	44.30	98,456
21.30	92,737	45.30	97,806
21.40	91,471	46.30	96,445
21.60	88,122	47.30	94,048
21.80	84,470	48.30	91,043
22.00	81,438	49.30	86,970
22.20	78,600	50.30	84,224
22.30	77,573	51.30	80,968
22.40	77,251	52.30	78,324
23.30	75,774	53.30	77,514
24.30	78,558	54.30	89,405
25.30	81,901	55.30	104,413
26.30	84,828	56.30	114,389
27.30	87,105	57.30	118,336
28.30	91,632	58.30	116,215
29.30	93,937	59.30	108,943
30.30	96,421	60.30	97,643

TABLE I.B.9. ^{239}Pu Radial Traverse in L-row, Assembly 52ECounter: ^{239}Pu #R

Power: 3 W (controlled by autorod)

Indexer: When center of active area of counter
(1 in. from counter end) was at core
centerline, indexer read 38.30

Indexer Position (in.)	Counts/30 sec	Indexer Position (in.)	Counts/30 sec
6.30	9,550	35.80	94,461
7.30	10,111	36.30	95,849
8.30	10,641	36.80	96,998
9.30	11,708	37.30	98,112
10.30	13,628	37.80	98,803
11.30	20,431	38.30	98,543
12.30	30,805	38.80	98,152
13.30	42,827	39.30	97,796
14.30	54,487	39.80	96,578
15.30	66,785	40.30	96,047
16.30	78,872	40.80	94,328
17.30	89,327	41.30	91,431
18.30	98,048	41.80	87,564
19.30	103,393	42.30	84,650
20.30	103,356	42.80	83,540
21.30	99,911	43.30	82,315
22.30	93,280	43.80	80,506
23.30	83,796	44.30	79,423
24.30	71,689	45.30	78,087
25.30	67,091	46.30	76,180
26.30	69,129	47.30	74,610
27.30	70,770	48.30	72,704
28.30	73,061	49.30	70,486
29.30	75,329	50.30	69,180
30.30	77,384	51.30	69,201
31.30	78,354	52.30	77,794
31.80	79,438	53.30	87,317
32.30	80,107	54.30	95,836
32.80	81,511	55.30	102,533
33.30	82,603	56.30	104,251
33.80	84,370	57.30	102,226
34.30	86,770	58.30	95,467
34.80	90,407	59.30	86,130
35.30	93,557	60.30	74,836

4. ZPR-6 and -9 Operations and Analysis

a. Doppler Effect (C. E. Till)

Last Reported: ANL-7438, pp. 18-20 (March 1968).

Doppler-effect measurements in fast reactor critical assemblies performed at Argonne over the past several years have utilized a reactivity technique in which a small heated sample of fertile or fissile material is oscillated in and out of the center of an unheated (20°C) core. The fast critical assemblies in which these measurements have been made, the Argonne ZPR facilities, are constructed of small plates of fertile, fissile, and diluent materials arranged in stainless steel drawers of 2-in. square cross section. Because of this construction, significant unavoidable heterogeneities occur in the immediate environment of the small Doppler-measurement samples as a result of local configuration of plate material, size, and orientation. These heterogeneities are significant in that they affect the gross flux and adjoint spectra at the samples; in addition, when the resonant material of the sample is present, unheated, in the surrounding core, the heterogeneities affect the microstructure of the flux incident on the sample.

Analyses have been completed of a series of experiments which were performed with a 1-in.-dia x 12-in.-long Doppler sample of natural UO_2 in which the environment of the sample was extensively varied in order to study the sensitivity of the measured Doppler results to the environmental configuration. This sample was chosen because of the high concentration of ^{238}U in the fast assemblies of interest, which emphasizes any heterogeneity effect, and because of the extensive measurements made on this sample in the past.

(i) Experimental Results. The measurements reported here were performed in a zoned fast critical assembly (ZPR-9 Assembly 18) which had an average central core composition typical of a dilute sodium-cooled uranium carbide-fueled (7/1 $^{238}\text{U}/^{235}\text{U}$) fast power reactor. This composition was identical to that used for previously reported ZPR-6 and -9 Doppler measurements. The loading pattern of Assembly 18 is shown schematically in Fig. I.B.6. The drawer pattern for the various core regions is summarized in Table I.B.10. Equivalent outer radii and homogenized regional atomic densities are given in Table I.B.11.

Fig. I.B.6. Loading Pattern for
ZPR-9 Assembly 18

Figure I.B.7 shows the standard environment which has been used routinely in the experiments up to this time. The figure shows a section view normal to the core axis

TABLE I.B.10. Regional Drawer-loading Patterns for ZPR-9 Assembly 18

Core (see Fig. I.B.7)

Type 1: $1/8$ D, $1/2$ Na, $1/8$ D, $1/16$ E, $1/8$ C, $1/8$ D, $1/8$ C, $1/16$ D, $1/2$ Na, $1/8$ D, $1/8$ C

Type 2: $1/8$ D, $1/8$ C, $1/16$ E, $1/8$ C, $1/16$ D, $1/2$ Na, $1/8$ D, $1/2$ Na, $1/8$ D, $1/16$ E, $1/8$ C, $1/16$ D

Type 3: $1/8$ D, $1/8$ C, $1/8$ D, $1/2$ Na, $1/8$ D, $1/8$ C, $1/8$ D, $1/16$ E, $1/8$ C, $1/16$ D, $1/2$ Na

$1/8$ D = $1/8$ -in.-thick depleted uranium plate

Na, E, C = Sodium, enriched uranium, and graphite plates

Buffer

$1/8$ C, $1/8$ D, $1/8$ C, $1/4$ Na, $1/8$ C, $1/8$ D, $1/8$ C, $1/8$ C, $1/8$ D, $1/8$ C, $1/4$ Na, $1/8$ C, $1/8$ D, $1/8$ C

Driver

$1/8$ D, $1/4$ Na, $1/8$ E, $1/8$ C, $1/4$ Na, $1/8$ E, $1/8$ D, $1/4$ Na, $1/8$ C, $1/8$ E, $1/4$ Na, $1/8$ C

TABLE I.B.11. ZPR-9 Assembly 18

Region	Outer Radius (cm)	Atom Densities $\times 10^{-24}$						
		Fe	Ni	Cr	Na	C	^{238}U	^{235}U
Core	25.883	0.009086	0.001148	0.002593	0.009028	0.012986	0.010633	0.001547
Buffer	32.525	0.008825	0.001116	0.002520	0.003917	0.034635	0.009708	0.000020
Driver	40.750	0.010638	0.001345	0.003038	0.007833	0.012986	0.005381	0.006876
Reflector I	54.520	0.004322	0.00054	0.00114	-	-	0.03990	0.000081
Reflector II	80.750	0.077772	0.000891	0.02036	-	-	-	-

Calculated Critical Mass: 581.8 kg ^{235}U .

Measured Critical Mass: 579 kg ^{235}U .

Mass of ^{235}U in Central Core: 116.2 kg.

Core Height: 36 in. (91.44 cm).

Estimated Axial Reflector Savings: 16.1 cm.

Total Assembly Volume (excluding reflector): 477.0 liters.

Volume of Central Core: 192.4 liters.

Fig. I.B.7

Standard Environment
around Doppler Sample

at the reactor midplane with the sample in place. Oscillation occurs normal to the plane of the figure. The drawer pattern shown is repeated vertically. The sample cans, oscillator drawer, and adjacent reactor matrix constitute a built-in stainless steel filter, approximately 1/4-in. thick, which is always present interposed between the core and the sample.

The first variation of the standard configuration of Fig. I.B.7 is shown schematically in Fig. I.B.8 and will be referred to as the revised standard configuration. Details of the sample can and oscillator drawer have been suppressed in the figure. The configuration change consisted of turning all of the plates in the adjacent four drawers so that their flat sides rather than their edges (as in the standard case) face the sample, and then subdividing the plates and rearranging them in the adjacent eight drawers so that each adjacent drawer contained four identical 1/2-in.-thick sections, each containing approximately core-average composition.

Fig. I.B.8. Revised Standard Configuration

The second variation consisted of rotating the adjacent four drawers in the revised standard configuration 90° so that the plate edges face the sample; this will be called the 90°-rotated revised standard configuration.

Environment variations three through six consisted of introducing filter zones around the sample by replacing portions of the revised standard configuration adjacent to the sample. Filters consisting of 1/4-in.-, 1/2-in.-, and 1-in.-thick stainless steel, and of 1/4-in.-, 1/2-in.-, and 1-in.-thick depleted uranium were investigated. The configuration is illustrated schematically in Fig. I.B.9 for the 1/2-in. stainless steel filter. Note that the built-in 1/4-in. stainless steel-equivalent filter zone is present in all cases.

Five additional variations were considered, consisting of the standard, revised standard, 90°-rotated revised standard, 1/2-in. stainless steel filter, and the 1-in. stainless steel filter. In all cases sodium was detected from the core.

Fig. I.B.9. One-half-inch Stainless Steel Filter Configuration

The experimental results for the various configurations are listed in Table I.B.12. Previous measurements had indicated that this

1-in. natural UO_2 sample had a negligibly small expansion effect, so that the numbers in the table represent Doppler changes. The data are summarized in Table I.B.13 where the estimated Doppler reactivity effect between room temperature and 1100°K is normalized to the results obtained for the revised standard configuration. Examination of these results for those variations in which the orientation of the ^{238}U plates in the drawers adjacent to the sample was changed with respect to the sample, viz., the standard, revised standard, and 90°-rotated revised standard, indicate that reasonable plate reorientation does not affect the results more than by about 5% either with or without sodium in the core.

TABLE I.B.12. Experimental Results with 1-in. Natural UO_2 Sample in ZPR-9 Assembly 18
Sample Mass = 1.116 kg Uranium (1.266 kg UO_2)

Sample Environment	Sodium in Core	Date (1967)	Standard Deviation of Each Point (lh/kg U)	Sample Temperature, °K; Reactivity Change (lh/kg Uranium) {Data Normalized to 0.00 at 293°K}			
Standard	Yes	6/23	0.002	293 0.000	510 -0.155	773 -0.289	1058 -0.394
		7/13	0.002	293 0.000	762 -0.289		
		8/25	0.002	293 0.000	756 -0.278	1044 -0.388	
		8/28	0.003	293 0.000	776 -0.284		
Revised Standard	Yes	6/28	0.002	293 0.000	499 -0.157	782 -0.303	1059 -0.418
		8/8	0.002	293 0.000	768 -0.301		
		8/21	0.002	293 0.000	749 -0.291	1044 -0.408	
		8/23	0.001	293 0.000	748 -0.291	1043 -0.404	
		8/23	0.002	293 0.000	748 -0.292	1043 -0.406	
1/4-in. SS Filter	Yes	8/17	0.001	293 0.000	479 -0.151	758 -0.299	1051 -0.408
1/2-in. SS Filter	Yes	7/11	0.001	293 0.000	771 -0.316		
		7/11	0.001	293 0.000	771 -0.318		
		8/11	0.001	293 0.000	505 -0.160	775 -0.320	
		8/16	0.001	293 0.000	760 -0.320	1064 -0.444	
1-in. SS Filter	Yes	7/5	0.002	293 0.000	505 -0.181	786 -0.353	1075 -0.477
90°-rotated Revised Standard	Yes	8/22	0.001	293 0.000	751 -0.289	1045 -0.393	

TABLE I.B.12 (Contd.)

Sample Environment	Sodium in Core	Date (1967)	Standard Deviation of Each Point (lh/kg U)	Sample Temperature, °K; Reactivity Change (lh/kg Uranium) {Data Normalized to 0.00 at 293°K}			
90°-rotated Revised Standard	Yes	8/22	0.001	293 0.000	748 -0.281		
		8/24	0.002	293 0.000	752 -0.283	1046 -0.394	
1/4-in. Depleted Uranium Filter	Yes	8/18	0.002	293 0.000	749 -0.263	1043 -0.356	
1/2-in. Depleted Uranium Filter	Yes	7/10	0.002	293 0.000	511 -0.128	796 -0.239	1047 -0.318
1-in. Depleted Uranium Filter	Yes	7/6	0.002	293 0.000	512 -0.089	796 -0.173	1057 -0.233
Standard	No	9/5	0.002	293 0.000	772 -0.199	1043 -0.270	
Revised Standard	No	9/6	0.001	293 0.000	766 -0.205	1043 -0.279	
90°-rotated Revised Standard	No	9/6	0.001	293 0.000	1043 -0.269		
1/2-in. SS Filter	No	9/8	0.001	293 0.000	768 -0.215	1043 -0.296	
1-in. SS Filter	No	9/11	0.002	293 0.000	770 -0.242	1043 -0.320	

TABLE I.B.13. Summary of Experimental Results for Doppler Reactivity Effect with 1-in. Natural UO₂ Sample for 293-1100°K

Configuration	Sodium in Core	Normalized to Revised Standard with Sodium	Normalized to Revised Standard without Sodium
Revised Standard	Yes	1.000	
90°-rotated Revised Standard	Yes	0.958 ± 0.010	
Standard	Yes	0.948 ± 0.010	
1/4-in. Extra SS Filter	Yes	0.995 ± 0.010	
1/2-in. Extra SS Filter	Yes	1.068 ± 0.010	
1-in. Extra SS Filter	Yes	1.138 ± 0.016	
1/4-in. Depleted U Filter	Yes	0.869 ± 0.010	
1/2-in. Depleted U Filter	Yes	0.779 ± 0.011	
1-in. Depleted U Filter	Yes	0.566 ± 0.010	
Revised Standard	No	0.683 ± 0.009	1.000
90°-rotated Revised Standard	No		0.962 ± 0.010
Standard	No		0.969 ± 0.010
1/2-in. Extra SS Filter	No		1.062 ± 0.015
1-in. Extra SS Filter	No		1.144 ± 0.018

(ii) Analytical Results. The extent to which sample-core resonance interactions affect the measured Doppler values is important both in the interpretation of experimental results and in the planning of experiments. The experiments reported here with the various filter zones surrounding the sample were undertaken primarily as an aid in the determination of the magnitude of this effect of sample-core interaction. In conjunction with the measurements, a detailed analysis was made of the revised standard, 1/4-in. stainless steel filter, 1/2-in. stainless steel filter, and 1/4-in. depleted uranium filter cases as well as for a hypothetical reference case in which the core is brought up to the sample surface (i.e., the built-in 1/4-in. stainless steel-equivalent filter is removed). Qualitatively, interposing stainless steel reduces the resonance-interaction effect between sample and core (an effect which increases the Doppler effect) and increases the Doppler range flux at the sample. Thus a balancing effect is to be expected, as is seen in Table I.B.13, where the 1/4-in. stainless steel filter case gives essentially the same result as the revised standard case.

The basic analytical tool used was the RABBLE code, which will permit a detailed resonance absorption calculation in the cylindrical multiregion configuration of the measurement. The integral slowing-down equation is solved by dividing the energy range into many very small (compared to the Doppler width) energy groups and applying collision-probability theory (flat-source) to a series of small annular regions. RABBLE allows arbitrary combinations of resonant and nonresonant materials in the various regions of the problem and region-dependent temperatures. The code provides regional coarse-group flux-collapsed cross sections as output.

Code limitations made it necessary to approximate the core region as a homogeneous region, not too bad an approximation. Further, it was possible to mitigate the error introduced by manipulating the input data in order to adjust the broad-group macroscopic absorption cross section to an appropriate value in the core region. This preserved the gross-source reduction rate.

RABBLE calculations were performed over the energy range from 111 keV to 29 eV with the sample at 293 and 1100°K, with the rest of the reactor at 293°K. The sample-region Doppler cross-section changes and the induced core-region cross-section changes were calculated both with ^{238}U resonances outside of the sample region explicitly included and with these resonances suppressed and replaced by a uniform background absorption cross section. ENDF/B (Category 1, zero iteration) resonance parameters were used with appropriately distributed resonances generated at energy points throughout the unresolved range. Only s-wave Doppler effects were considered on the assumption that the p-wave resonances are so small and strongly overlapped as to have no Doppler effect.

Core-region coarse-group cross-section sets were generated with the MC² code on the basis of the ENDF/B (Category 1, zeroth iteration) microscopic cross-section set. These cross sections together with the RABBLE-generated sample-region cross sections were used in SNARG-1D S₈ calculations to determine the sample-region coarse-group flux and adjoint distributions; the flux was calculated with the sample at 1100°K and the adjoint with the sample at 293°K.

The results of the SNARG flux and adjoint calculations are summarized in Table I.B.14. The expected sample-region coarse-group flux variations are seen. The no-filter case in the table refers to the hypothetical case in which the core begins at the sample surface. Table I.B.15 summarizes the calculated RABBLE sample-region Doppler changes (293 to 1100°K) and the induced core region changes in ²³⁸U cross sections listed per ²³⁸U atom in the sample. For reference, sample cross-section values calculated with suppression of the ²³⁸U resonances outside of the sample region are shown in the last column.

TABLE I.B.14. Summary of Calculated Flux and Adjoint Ratios, Sample/Core, and Calculated Core Flux and Adjoint Spectra

Group	Lower Energy Limit (eV)	No Filter		Revised Standard Case		1/4-in. SS Filter		1/2-in. SS Filter		1/4-in. Depleted Uranium Filter		Core Flux	Core Adjoint
		Flux Ratio	Adjoint Ratio	Flux Ratio	Adjoint Ratio	Flux Ratio	Adjoint Ratio	Flux Ratio	Adjoint Ratio	Flux Ratio	Adjoint Ratio		
8	111,100												
9	67,380	1.011	0.989	1.010	0.983	1.015	0.978	1.020	0.973	1.014	0.947	10.22	3.37
10	40,870	1.012	0.985	1.016	0.979	1.017	0.974	1.018	0.968	1.018	0.933	8.54	3.29
11	24,790	1.007	0.981	1.029	0.972	1.056	0.964	1.093	0.956	1.022	0.965	6.85	3.26
12	15,030	0.988	0.976	0.898	0.972	0.865	0.956	0.833	0.960	0.858	0.903	5.48	3.29
13	9,119	0.985	0.970	1.094	0.962	1.123	0.953	1.153	0.943	1.049	0.881	3.84	3.37
14	4,307	1.002	0.964	1.094	0.951	1.163	0.933	1.250	0.914	1.023	0.956	3.01	3.51
15	2,613	1.147	0.956	1.604	0.942	2.013	0.927	2.613	0.910	1.529	0.845	0.534	3.85
16	2,035	0.934	0.967	0.897	0.958	0.896	0.947	0.920	0.932	0.797	0.886	0.354	3.98
17	1,234	0.900	0.973	0.928	0.959	0.941	0.944	0.961	0.926	0.841	0.895	0.995	4.06
18	961	0.921	0.975	0.980	0.952	1.011	0.930	1.054	0.907	0.886	0.879	0.318	4.18
19	583	0.964	0.972	1.081	0.965	1.172	0.955	1.298	0.943	0.994	0.895	0.370	4.49
20	275	0.985	0.974	1.173	0.956	1.358	0.939	1.641	0.918	1.097	0.894	0.196	4.75
21	101.3	0.996	0.980	1.321	0.961	1.717	0.941	2.416	0.918	1.233	0.892	0.0385	4.82
22	29.02	1.137	0.979	1.178	0.955	1.610	0.931	2.356	0.902	1.248	0.884	0.00303	4.62

TABLE I.B.15. Summary of Calculated S-wave Doppler Cross-section Changes, 293 to 1100°K, for 1-in. Natural UO₂ Doppler Sample

Case:	No Filter		Revised Standard Case		1/4-in. SS Filter		1/2-in. SS Filter		1/4-in. Depleted Uranium Filter		No Resonance in Core
	$\delta\sigma_a$ Sample	$\delta\sigma_a^{(a)}$ Core	$\delta\sigma_a$ Sample	$\delta\sigma_a^{(a)}$ Core	$\delta\sigma_a$ Sample	$\delta\sigma_a^{(a)}$ Core	$\delta\sigma_a$ Sample	$\delta\sigma_a^{(a)}$ Core	$\delta\sigma_a$ Sample	$\delta\sigma_a^{(a,b)}$ Core	
9	0.000489	0.000143	0.000383	0.000042	0.000355	0.000040	0.000343	0.000012	0.000408	0.000052	0.000333
10	0.00169	0.00051	0.00131	0.00018	0.00122	0.00008	0.00117	0.00002	0.00140	0.00020	0.00114
11	0.00482	0.00127	0.00385	0.00057	0.00360	0.00027	0.00349	0.00000	0.00407	0.00052	0.00340
12	0.0117	0.0029	0.00955	0.00099	0.00901	0.00040	0.00878	0.00012	0.00999	0.00110	0.00858
13	0.0252	0.0048	0.0212	0.0017	0.0202	0.0009	0.0198	0.0004	0.0220	0.0021	0.0194
14	0.0576	0.0086	0.0507	0.0030	0.0489	0.0013	0.0481	0.0006	0.0521	0.0037	0.0476
15	0.0971	0.0108	0.0917	0.0045	0.0903	0.0021	0.0897	0.0009	0.0940	0.0055	0.0894
16	0.151	0.070	0.142	0.003	0.141	0.002	0.140	0.000	0.146	0.003	0.139
17	0.175	0.038	0.171	0.002	0.170	0.001	0.170	0.000	0.170	0.002	0.167
18	0.255	-0.010	0.245	0.001	0.243	0.002	0.242	0.000	0.251	0.001	0.249
19	0.356	-0.007	0.354	-0.003	0.353	0.000	0.352	0.000	0.354	-0.003	0.352
20	0.357	-0.014	0.357	-0.001	0.355	0.000	0.354	0.000	0.358	-0.007	0.354
21	0.407	-0.032	0.426	-0.020	0.429	-0.021	0.428	-0.010	0.426	-0.035	0.430
22	0.285	-0.073	0.342	-0.023	0.350	-0.005	0.345	-0.002	0.341	-0.041	0.345

(a) per ²³⁸U atom in sample.

(b) 1/4-in. depleted uranium filter zone.

The calculated sample and core-region Doppler cross-section changes and the calculated flux and adjoint distributions were used in the usual multigroup perturbation formulation to obtain calculated values for the relative Doppler effect to be expected in the various configurations. These values are listed in Table I.B.16 normalized to the no-filter case.

TABLE I.B.16. Calculated Flux and Adjoint-weighted Relative Doppler Effects for 1-in. Natural UO₂ Sample

Environment ^a	With Core Resonances	Without Core Resonances
No Filter	1.000	1.000
Revised Standard	0.973	1.087
1/4-in. Extra SS	0.996	1.157
1/2-in. Extra SS	1.064	1.252
1/4-in. Depleted Uranium	0.857	0.931

Ratios	With/Without Resonances Core
No Filter	1.187
Revised Standard	1.063
1/4-in. Extra SS	1.023
1/2-in. Extra SS	1.008
1/4-in. Depleted Uranium ^b	1.086

^aAll cases except No Filter include built-in 1/4-in. SS filter.

^bWithout resonance; ²³⁸U resonances in the 1/4-in. depleted uranium filter also suppressed.

The calculated extent of sample-core resonance interaction is also shown in Table I.B.16 where the calculated Doppler changes with core resonances are normalized to the case with these resonances suppressed.

These analytical results indicate that, while an approximately 19% resonance-interaction effect is present with no filter between the sample and the core, this is reduced to approximately a 6% effect in the usual measurement configuration and to about 3% with an additional 1/4-in. stainless steel filter.

Figures I.B.10 and I.B.11 demonstrate the agreement between calculation and experiment. Figure I.B.10 demonstrates the way the two effects--flux perturbation and hot-cold resonance interaction--combine to make up the total effect. The figure is normalized such that the ideal case with no flux perturbation or resonance interaction has a Doppler effect of unity. As the figure shows, the hot-cold resonance-interaction effect

decreases rather rapidly with increasing filter thickness, and the flux-perturbation effect increases in a regular manner with increasing filter thickness. The three experimental points relevant to this comparison show good agreement with the predicted trend of the total effect.

Fig. I.B.10. Effect on the Doppler Effect of Steel Filter Extrapolated Back to Zero Steel Thickness; Comparison of Calculation and Experiment

Fig. I.B.11. Effect on the Doppler Effect of Steel and Uranium Filters, 1/4-in. Built-in Steel Filter Always in Place; Comparison of Calculation and Experiment

Figure I.B.11 shows the effects both of steel and uranium filters. The right-hand side of Fig. I.B.11 essentially repeats the information given on Fig. I.B.10, except that the calculations are not extrapolated back to zero filter thickness. The left-hand side of Fig. I.B.11 shows the effect of increasing thickness of the uranium filter, but this differs somewhat from the steel filter case in that a combination of steel and uranium is always present in the uranium filter experiments because of the presence of the 1/4-in. built-in steel filter.

The experiments with the uranium filter were carried out to a full 1-in. of uranium thickness. As the figure shows, the main effect of the uranium filter is to depress the flux, rather linearly with increasing uranium thickness, and this trend is borne out by the experiments at the larger thicknesses of the uranium filter.

In summary, it is believed that the good agreement demonstrated between these very detailed experiments and calculations indicate that a good understanding of these effects has been achieved. It is concluded that while these effects must be considered in designing and interpreting measurements of this kind, by proper experimental design along the lines suggested by the above work the effects can be made rather small and of known magnitude. Specifically, the hot-cold interaction effects, which require very detailed calculations, can be made small, and in fact are normally rather small in the usual experimental environments of reactivity Doppler measurements done to date.

b. Integral Studies of Large Systems (W. Y. Kato)

Last Reported: ANL-7460, pp. 22-25 (June 1968).

(i) Measurements with Assembly 6 of ZPR-6. Analysis of the data obtained with Assembly 6 of ZPR-6 continued. A 22-group cross-section set was generated for Assembly 6 from the ENDF/B data with the MC² code and subsequently spatially averaged with the intrastructure of the real fluxes. The critical mass calculated with this set was 2088 kg ²³⁵U. The measured critical mass was 1817 kg of ²³⁵U. The discrepancy between the measured and calculated critical mass is about 15% ($\approx 1.5\% \Delta k/k$).

The measured central reactivity worths of a few typical samples (nominally 2 x 2 in. with varying thickness) as well as some calculated values are given in Table I.B.17. The third column lists the calculated values with first-order perturbation (FOP) theory and Cross Section Set 224* (values in parentheses were obtained with ENDF/B parameters). A correction to FOP theory incorporating self-shielding inside and flux depression outside the sample, as well as in-group and out-of-group scattering, was made in each energy group** and the results are given in the

TABLE I.B.17. Measured and Calculated Central Reactivity Worths in Assembly 6 of ZPR-6

Material	Sample Weight (g)	Calculated Worth ^a (lh) ^b		Measured Worth (lh)
		First-order Perturbation (FOP)	FOP Corrected for Self-shielding	
Fe	488.0	-0.745 ^c (-0.658) ^d	-0.646 (-0.561)	-0.450 $\pm$ 0.004
Ni	546.0	-1.069 (-0.721)	-0.929 (-0.748)	-0.833 $\pm$ 0.008
Cr	220.73	-0.363 (-0.403)	-0.352 (-0.369)	-0.222 $\pm$ 0.008
Na	51.38	-0.023 (+0.002)	-0.033 (-0.004)	+0.013 $\pm$ 0.008
¹⁰ B	29.29	-33.015 -	-21.945 -	-23.209 $\pm$ 0.010
Ti	90.42	-0.045 -	-0.061 -	+0.026 $\pm$ 0.013
Zr	406.0	-0.464 -	-0.425 -	-0.443 $\pm$ 0.007
Nb	481.2	-5.259 -	-4.290 -	-3.525 $\pm$ 0.007
Mo	599.0	-3.888 -	-3.165 -	-3.115 $\pm$ 0.021
Al	165.33	-0.279 -	-0.263 -	-0.122 $\pm$ 0.010
C	103.0	+0.036 -	+0.042 -	+0.366 $\pm$ 0.002
²³⁸ U	5.913	-0.0211 (-0.0248)	-0.0211 (-0.0248)	-0.0252 $\pm$ 0.002
²³⁸ U	11.739	-0.0420 (-0.0493)	- (-0.0494)	-0.0429 $\pm$ 0.002
²³⁸ U	28.977	-0.1037 (-0.1216)	- (-0.122)	-0.103 $\pm$ 0.002
²³⁸ U	60.227	-0.215 (-0.253)	- (-0.255)	-0.220 $\pm$ 0.002
²³⁸ U	147.123	-0.526 (-0.617)	- (-0.623)	-0.514 $\pm$ 0.002
²³⁸ U	1153.580	-4.132 (-4.838)	- (-3.710)	-3.764 $\pm$ 0.003
²³⁵ U	4.511	- (+0.191)	- (+0.190)	+0.211 $\pm$ 0.002
²³⁵ U	22.950	- (+0.969)	- (+0.986)	+0.969 $\pm$ 0.002
²³⁵ U	61.491	- (+2.596)	- (+2.665)	+2.567 $\pm$ 0.020

^aCalculated with one-dimensional diffusion theory.^b1% $\Delta k/k$ = 458 lh.^cValues obtained from cross-section Set 224.^dValues in parentheses obtained from ENDF/B parameters.*O'Shea, D. M., et al., Twenty-six Group Cross Section, Trans. Am. Nucl. Soc. 7(2) 242 (1964).**Beckurts, K. H., and Wirtz, K., Neutron Physics, Springer-Verlag, New York (1964); also Nakamura, T., Self-shielding Corrections to First-order Perturbation Theory, to be published.

fourth column.. The correction is accurate as long as the multigroup approximation to the neutron spectrum is adequate. For ^{10}B , for which the cross sections are smooth with energy, the calculated value is in good agreement with that measured. For ^{238}U , which has many resonances, the agreement with measurements is good for the thinnest and thickest samples. This may be due to negligible self-shielding in the thinnest sample and to a small contribution from resonance neutrons to the total worth of the large sample.

(ii) Plutonium Alpha Experiments. An experiment to measure the effective capture-to-fission ratio (alpha) for ^{239}Pu by a short-term irradiation in either a pulsed subcritical or critical facility having a soft neutron spectrum, which emphasizes the 10-keV range where the greatest uncertainty in the alpha exists, has been proposed and is under investigation. The basic concept is to produce a sufficient amount (~100 ppb) of ^{240}Pu so that the irradiated sample of plutonium may be analyzed mass spectrometrically for the ^{240}Pu content.

One concept of the experiment is a cooperative one between Argonne and the Linac group at Gulf General Atomic. A small, 46-liter, soft-spectrum core of ^{235}U , ^{238}U , and BeO, having a multiplication of about 1.5, was built on the Gulf General Atomic's STSF facility. This facility is a split-table reactor, similar to ZPR-3, which may be pulsed with the Linac beam. It may be used to build subcritical systems up to about \$10 subcritical with fuel loadings up to 600 kg of ^{235}U . This core would be pulsed with the Linac to produce fission neutrons in the core at a level equivalent to about 3 to 4 kW. A small sample of high-purity ^{239}Pu (~10 ppb ^{240}Pu) and a sample of normal plutonium of known weight and isotopic composition would be irradiated at the center of the core for about 120 hr. This would result in the production of sufficient quantities of ^{240}Pu (~100 ppb) in the high-purity sample for mass-spectrometric determination of the capture rate and sufficient quantities of ^{140}Ba (of 13-day half-life) in the normal plutonium sample for gamma-spectrometric determination of the fission rate. A time-of-flight spectrum measurement would also be made from the center of the core in order to have a definitive knowledge of the spectrum where the irradiation took place. Comparison of the experimental effective alpha and the calculated alpha would help to clarify the discrepancy in the differential alpha measurements.

Production of the high-purity ^{239}Pu sample was accomplished using the Argonne Chemistry Division's mass separator. The preparation of the samples to be irradiated and the fission product analysis are being conducted by the Argonne Chemical Engineering Division. The mass-spectrometric analysis will be carried out using the Chemistry Division's tandem mass spectrometer [see Sect. II.A.2.a(ii)].

Initial intensity checks were made with the STSF assembly late in June. This was accomplished by irradiating small samples of ^{235}U ,

^{238}U , and plutonium in the assembly for short periods of time. It was found that a large fraction of the electron beam from the Linac was lost after it was bent approximately 104° to go into the natural uranium target imbedded in the subcritical assembly. These intensity checks revealed that a current whose magnitude is greater by a factor of about 25 than was available is needed in order to have a sufficient neutron flux to produce 100 ppb of ^{240}Pu in 120 hr. Since a large fraction of the beam is lost after bending, the bending magnet is currently being studied to determine how to improve the amount of electron beam current on the target.

(iii) Comparison of Central Worths in a Full-size Oxide Critical and in a Small Zone. Table I.B.18 contains a listing of central worths measured in Assembly 19 of ZPR-9 and compares these values with central worths measured in ZPR-6 Assembly 6 (see Progress Report for March 1968, ANL-7438, pp. 17-18). The relatively constant values of the worth ratios between the two assemblies indicates that the neutron spectrum in the test zone of Assembly 19 is well matched to the Assembly 6 spectrum.

TABLE I.B.18. Central Reactivity Worths in Assembly 19; Comparison with Assembly 6

Sample Material	Weight of Sample (g)	Can Type	Can Weight (g)	Gross Sample Worth (lh)	Net Sample Worth (lh)	Specific Sample Worth (lh/kg)	Worth Assy 19, ZPR-9 Worth Assy 6, ZPR-6
$(\text{CH}_2)_n$	58.986	Bare	-	17.513 ± 0.009	17.513 ± 0.009	296.9 ± 0.2	1.421
Be	114.308	Bare	-	1.801 ± 0.007	1.801 ± 0.007	15.76 ± 0.06	1.381
^{10}B	29.29	SS	55.73	-39.312 ± 0.070	-39.195 ± 0.071	-1338.2 ± 2.4	1.689
C	102.0	Bare	-	0.459 ± 0.007	0.459 ± 0.007	4.50 ± 0.07	1.254
Na	51.38	SS	56.42	-0.095 ± 0.005	0.020 ± 0.007	0.39 ± 0.14	1.538
Al	165.33	Bare	-	-0.091 ± 0.007	-0.091 ± 0.007	-1.15 ± 0.04	1.746
Fe	488.0	Bare	-	-0.720 ± 0.007	-0.720 ± 0.007	-1.475 ± 0.014	1.600
SS Can	72.39	Bare	-	-0.153 ± 0.005	-0.153 ± 0.005	-2.11 ± 0.07	-
SS Block	480.9	Bare	-	-0.871 ± 0.007	-0.871 ± 0.007	-1.81 ± 0.02	1.643
Ni	546.0	Bare	-	-1.435 ± 0.007	-1.435 ± 0.007	-2.63 ± 0.02	1.725
Zr	406.0	Bare	-	-0.748 ± 0.007	-0.748 ± 0.007	-1.84 ± 0.02	1.688
Nb	481.2	SS	30.10	-5.983 ± 0.007	-5.920 ± 0.009	-12.30 ± 0.02	1.679
Mo	599.0	Bare	-	-5.115 ± 0.075	-5.115 ± 0.075	-8.539 ± 0.125	1.642
Ta	924.7	Bare	-	-19.724 ± 0.007	-19.724 ± 0.007	-21.269 ± 0.008	-
W	1108.0	Bare	-	-8.945 ± 0.007	-8.945 ± 0.007	-8.074 ± 0.007	1.756
Enriched	18.17 U 16.93 ^{235}U	SS	72.603	1.100 ± 0.007	1.254 ± 0.009	69.01 ± 0.50	-
Depleted	1153.8	Bare	-	-6.317 ± 0.007	-6.317 ± 0.007	-5.475 ± 0.006	1.678

c. Heterogeneity Effects (C. E. Till)

Last Reported: ANL-7445, pp. 16-20 (April 1968).

(i) Theoretical. The analysis of ZPR-6 Assemblies 5 and 6 required a treatment of the spatial heterogeneity of the plate-type cell associated with these assemblies. The treatment was developed on a broad-group basis using equivalence theory for ^{238}U and ^{235}U . Within this context different approaches were used to account for spatial heterogeneities, and two codes were written to facilitate the preparation of cross sections for homogeneous-cell MACH calculations. One approach was to use MC^2 , with a simple one-absorber region, one diluent-region cell with the approximate σ_p of the actual cell pattern, to compute energy-averaged

broad-group cross sections. These cross sections are correct in-plate cross sections and were used in a SNARG cell calculation to compute the spatial dependence of the broad-group fluxes. A code was then written, for use with the CDC-3600, to punch a new MACH-format cross-section deck, using the MC² MACH cross-section cards and the SNARG fluxes as input data. The punched cross-section deck incorporates spatial effects in the cell and can be used in overall MACH calculations for the core. The code can be used to obtain linear flux-weighted cross sections

$$\frac{\overline{\Sigma_{i \rightarrow j}}}{\Sigma_{i \rightarrow j}} = \frac{\int \Sigma_{i \rightarrow j}(\vec{r}) \Phi_i(\vec{r}) d\vec{r}}{\int \Phi_i(\vec{r}) d\vec{r}} \quad (1)$$

or bilinear (adjoint) weighted cross sections

$$\frac{\overline{\Sigma_{i \rightarrow j}}}{\Sigma_{i \rightarrow j}} = \frac{\int \Sigma_{i \rightarrow j}(\vec{r}) \Phi_i(\vec{r}) \Phi_j^*(\vec{r}) d\vec{r}}{\int \Phi_i(\vec{r}) \Phi_j^*(\vec{r}) d\vec{r}} \quad (2)$$

The bilinear weighted cross section of Eq. (2) must be used in a special version of MACH:

(MACH/SPECIAL) L50707,

as the use of bilinear weighting necessitates the use of a full $\chi^{(i \rightarrow j)}$ matrix instead of the usual MACH scalar χ_j .

A code, SIGP, was also written to determine the approximate ²³⁸U and ²³⁵U cross sections, within the resonance-energy range, in a heterogeneous cell without having to make MC² or SNARG calculations. The code calculates both the in-plate cross sections for use in cell calculations such as with SNARG, and the cell-averaged cross sections (quantities not directly available from MC² calculations) for use in homogenized-cell calculations such as done with MACH. The basic cross sections in the resonance region for each absorber plate are shielded through the usual σ_p formulation after finding the Dancoff shielded σ_p of each individual absorber plate using the two-sided E₃ formulation of Meneghetti.[†] In order to determine cell-averaged cross sections, the code needs a relation for the flux shape in the diluent regions, which was determined using the usual narrow-resonance, flat-source assumptions. With these assumptions the flux shape between two black absorber plates separated by distance d is given as

[†]Meneghetti, D., ANL-7320, p. 377 (1966).

$$\Phi(x) = \frac{1}{2} \left\{ \int_0^x dx' E_1(x - x') + \int_0^{d-x} dx' E_1(x') \right\}. \quad (3)$$

The code permits calculation of the cell-averaged cross section by use of Eq. (3) for the flux shape in the diluent region at resonant energies and assuming a flat flux in the diluent region at a nonresonant energy.

Calculations for Assemblies 5 and 6 of ZPR-6 showed the final cell-averaged cross section calculated using the SNARG fluxes for linear flux weighting and the cross sections obtained from the simple SIGP code to be in excellent agreement. Further, a RABZPR calculation over one resonant energy group showed good agreement with the broad-group cell-averaged cross section determined with the above codes.

(ii) Experimental. Knowledge of the spatial fine structure of the flux through a unit cell in the heterogeneous plate-type criticals is necessary in order to average the broad-group cross sections over space to account for heterogeneity effects.* Measurements bearing on the adequacy of calculational procedures to predict the flux fine structure were made by measuring the fission rates in ^{235}U and ^{238}U as a function of position in a unit cell at the center of Assembly 6 of ZPR-6.

In one measurement depleted uranium foils (0.2% enrichment, 5 mils thick and of 5/8-in. diameter) and ^{235}U foils (93% enrichment, 4.5 mils thick and of 3/8-in. diameter) were placed between adjoining plates throughout the cell at corresponding locations about the center of the two halves of the ZPR-6 machine. In another experiment, a $2 \times 2 \times 1/16$ -in. plate of 93% enriched uranium and a $2 \times 2 \times 1/8$ -in. plate of depleted uranium (0.2% enriched) were replaced by packets of foils equivalent to the two plates in weight and enrichment. All foils were irradiated for one hour at a power level of about 50 W and subsequently counted for fission products.** In the case of the packets, only 5/8-in.-dia foils punched out from the center of the $2 \times 2 \times 0.005$ -in. foils comprising the packets were counted. This procedure eliminated any corrections for edge effects. A few foils were radiochemically analyzed for absolute fissions. In a third experiment, the reactivity worth of ^{235}U as a function of sample thickness was measured.

The results are given in Figs. I.B.12 and I.B.13. The solid and dashed lines were calculated with the SNARG code† with SN order of 32

*Meneghetti, D., "Calculational Studies of Sodium-void Reactivity Variations due to Their Slab Heterogeneities in Fast Criticals," Proc. Inter. Conf. on Fast Critical Experiments and Their Analysis, ANL-7332 (1966).

**Stanford, G., The Codes NURF, COMBO, and TWOSOURCE for Processing Foil Counting Data, ANL-7356 (1967).

†Duffy, G. J., et al., SNARG-1D A One-dimensional Discrete-ordinate Transport Theory Program for the CDC-3600, ANL-7221 (1966).

using the modified Gaussian quadratures* and a 22-group set generated for Assembly 6 with the MC² code from ENDF/B data.** The results show: (1) the peaking of the high-energy fluxes as measured by ²³⁸U fission rates and the depression of the low-energy fluxes in the ²³⁵U plate that combine to give a net negative heterogeneity effect for the ²³⁵U plates, (2) the fission rate in ²³⁵U within the plate is flat (see Fig. I.B.13), (3) general agreement between calculated and measured spatial distribution of the fission rates, and (4) the calculated fission rate in ²³⁸U per fission in ²³⁵U is about 16% lower than the measured value regardless of the position within the cell. Independent measurements also showed that the reactivity worth of ²³⁵U decreased by 10% as the thickness is increased from 5 mils to that of a 1/16-in.-thick plate.

Fig. I.B.12. Fine Structure in Fission Rates in Cell of Assembly 6 of ZPR-6

Fig. I.B.13. Fission Reaction Rates in Depleted and Enriched Uranium Foil Packets Simulating Plates in Assembly 6 of ZPR-6

5. ZPPR Operations and Analysis (W. G. Davey and P. I. Amundson)

Last Reported: ANL-7460, pp. 25-27 (June 1968).

a. Basic Studies of Large Plutonium Systems

(i) Program. The analytical evaluation of the worths of a fueled safety rod and a boron safety blade has been completed. The worth curves have been used to determine safety- and control-rod positions for Core I. The positions selected are given in Table I.B.19. The numbering scheme selected for ZPPR matrix tubes is used in the position column: the first digit indicates reactor half (1--stationary; 2--movable); the next two digits indicate vertical columns (these read left to right on Half 1 and right to left on Half 2); the last two digits indicate the horizontal row (numbered from top to bottom).

*Meneghetti, D., Discrete Ordinate Quadratures for Their Slab Cells, Nucl. Sci. Eng. **11**, 295-303 (1962).

**Toppel, B. J., et al., MC²--A Code to Calculate a Multigroup Cross Section, ANL-7318 (1967).

TABLE I.B.19. Selected Control- and Safety-rod Positions and Calculated Worths

Description	Position	Radius (cm)	Calculated Worth (% $\Delta k/k$)
Poison Blade	139-30	40.325	0.374/2
Poison Blade	134-44	40.325	0.374/2
Poison Blade	239-30	40.325	0.374/2
Poison Blade	244-31	52.248	0.265/2
Poison Blade	234-44	40.325	0.374/2
Poison Blade	130-43	52.248	0.265/2
Fueled Safety Rod	141-36	23.744	0.1105
Fueled Safety Rod	133-38	23.744	0.1105
Fueled Safety Rod	241-36	23.744	0.1105
Fueled Safety Rod	233-38	23.744	0.1105
Fueled Safety Rod	143-41	41.077	0.080
Fueled Safety Rod	131-33	41.077	0.080
Fueled Safety Rod	243-41	41.077	0.080
Fueled Safety Rod	231-33	41.007	0.080
Fueled Control Rod	141-39	25.596	0.1075
Fueled Control Rod	233-35	25.596	0.1075

1.013% $\Delta k/k$

0.762% $\Delta k/k$

1.775% $\Delta k/k$ Total

0.215% $\Delta k/k$

Approximately 672 kg of plutonium fuel pieces have been received, checked for dimensional tolerances and contamination, and stored in the ZPPR vault.

b. Doppler Coefficients

(i) Design. The drawings of the Doppler drive system and the Doppler sample support and containments are being reviewed for cost estimates, and the bid packages are being prepared. All major commercially supplied items have been received except for the drive motor and control, and the ion pump and control.

Two independent safety analyses have been performed for the Doppler drive and capsule system. These have shown that the system design is adequate from a reactor-safety standpoint.

(ii) Equipment. The bids for the compressor systems are being reviewed. No satisfactory bids have been received for the special-design aftercooler. The entire cooling system is being reviewed for a possible complete change in design concept.

With the completion of the Doppler-drive-system design, work has begun on the design of the thermocouples and heater wires for the Doppler sample. Work has also begun on design of the remote control for equipment which has been received.

c. Reactor Equipment Development

(i) Experimental Equipment. The preparation of reactor experiments continues. The status of items of equipment under development is as follows:

(a) The mechanical system for the perturbation-sample changer has been received, and is being assembled and inspected. Work on the associated control system has been started.

(b) Special fuel pieces that accept a horizontal, radial, traverse tube have been constructed. The initial order of 50 elements will be received and inspected shortly.

(c) A safety analysis of the axial traverse-drive system is in progress preparatory to accepting a bid for construction.

(d) The experimental console in the control room is being assembled. General experimental equipment has been installed. Special control and readout systems will be installed as they are completed.

(ii) Automation of Data Acquisition

(a) Digital Position Indicators: Two additional digital position indicators are under construction. One unit will be used for position indication of rods or experiments, and the other will be used in conjunction with the data-acquisition and -recording system.

(b) Buffer Scaler Register: Design and circuit-board layout has been completed for a prototype buffer scaler register. These units will provide a general-purpose interface between experiments and the ZPPR computer.

(c) Autorod Control: Testing is continuing on the autorod control system. A redesign of the temperature-control oven appears to be required to increase the oven temperature. This is a simple redesign and should not delay completion of the project by more than one week.

d. Heterogeneity. Evaluation of data taken from a ^{238}U -foil exposure in AFSR together with an efficiency calibration for the 11-cc germanium detector has given values for the fission yields of eight mass chains in the heavy-fragment peak and eight chains in the light-fragment peak. They have

been normalized and compared to Katcoff's compilation. Generally, the agreement is quite good. Some cases show a definite discrepancy, such as mass chain 99.

Figure I.B.14 shows our data normalized to Katcoff's by a least-squares method for four mass chains: $^{140}\text{Ba-La}$, ^{132}I , ^{97}Nb , and ^{103}Ru .

Fig. I.B.14. Relative Fission Yields in ^{238}U

the workroom and the cell, and the two seal doors D73 and D74 in the emergency escape tunnel from the cell.

The sealing functions of the doors were tested several times to assure repeatability of the seals. For each test the door was opened and closed by operating the normal cam mechanisms, and separate leak-rate measurements were made. Tests of door D74 indicated an average leak rate of 320 cc/min with a test pressure of 5 psig. A check on the door itself was made by use of Freon tracers in the test air and a leak detector. No discernible leak was found in the door and its seals. Many minor leaks were detected around the test frame and test cover. There was constant difficulty in getting the test frames and test covers acceptably leaktight.

Seal doors D74 (leak rate 320 cc/min), D65 (550 cc/min), and D69 (187 cc/min) were tested satisfactorily. Seal door D73 (220 cc/min)

Narrowing of the light-fragment peak may be due to the softer neutron spectrum in our measurements and also uncertainties in the decay-scheme data.

Compilation of decay-scheme data on major fission products is underway. Essential data are being put on punch cards for computer input and eventual use with programs for peak identification and isotope-yield calculations.

6. ZPPR Construction (H. Lawroski)

Last Reported: ANL-7460,
pp. 27-28 (June 1968).

The major activity during July was the testing of seal doors. These include the two large seal doors D65 and D66 to the workroom from outside and from the control room, the two seal doors D67 and D69 in the access tunnel between

was also tested satisfactorily, but a combination test of the bulkhead and the door was necessary owing to leakage between the door test frame and the bulkhead. Measurements indicated that the major leakage was through the bulkhead.

Approximately 45 tons of depleted uranium and uranium oxide were received and stored in the blanket-storage area of the cell. Checking of fuel into the vault continued.

Equipment for testing the filtering capability of the containment structure was assembled and is ready for use. Procedures for performing the tests were written.

The status of the installation of the instrumentation and control system for ZPPR is as follows: five cabinets for use in reactor experiments were installed complete with cooling blowers and voltage regulators; five nuclear-instrument module bins with their power suppliers were installed; the multipoint temperature recorder and its patch panel for selecting 24 of 60 thermocouples for recording were put into place along with the thermocouple-reference junctions; the cabinet for a digital multiplexer and for the counter preamps was installed in the cell.

Extensive cabling for experiments was connected through the seal pit in the reactor cell. This included cabling for five TV cameras, experimental control cables, computer data-link lines, and experimental coax.

The autorod drive and its control system were completed and are being tested. Fabrication of the radial traverse drive was completed, and the drive is now being tested. The sample changer was delivered from the manufacturer.

7. ZPR Materials

a. Development of Fabrication Techniques (J. E. Ayer)

Last Reported: ANL-7457, pp. 31-33 (May 1968).

(i) Oxide Rod Elements. The Plutonium-Materials Fabrication Group expects to load 25 oxide-rod elements during early fiscal year 1969. The manufacture of these 25 elements constitutes a pilot run that will be used as a basis for the development of a procurement specification. The necessary fuel-element hardware is on hand and has been prepared for loading. Early in July about 2 kg of $(U_{0.7}Pu_{0.3})O_2$ and $1\frac{1}{2}$ kg of $(U_{0.85}Pu_{0.15})O_2$ in the form of pellets were received from Dow Chemical Co. (Rocky Flats). These pellets are now being weighed and gauged preparatory to loading.

(ii) Instrumented Traverse Elements. Two hundred and fifty instrumented traverse elements are required for ZPR experiments by the end of fiscal year 1969. Fabrication procedures have been developed during the pilot production of the first 50 elements. The fuel elements are now being inspected to meet standard ZPR-plate specifications where applicable. The results of the inspections will be used to establish specifications and quality-assurance criteria for the production of the additional 200 instrument traverse elements.

b. Thin U-28.3 w/o Pu-2.5 w/o Mo Fuel Elements. Sixty fuel elements of nominal 1/8-in. thickness are needed as shims for weight adjustments in ZPR experiments. The elements must meet all specifications pertaining to standard ZPR fuel plates, with the exception of thickness and weight. A mold, crucibles, and the fuel material necessary for fabrication were received; the design of the fuel-element hardware was determined; and the jackets and end plugs were ordered. Casting runs to determine mold dimensions and feasibility of casting the thin plates (about 0.090 in. thick) have been successfully completed.

C. Component Development--LMFBR

1. Sodium Technology Development

a. Engineering Development

(i) On-line Monitoring Development and Evaluation (S. B. Skladzien)

Last Reported: ANL-7460, pp. 31-33 (June 1968).

When tests of the response of the in-line instruments in the Sodium Analytical Loop were hampered by clogged cooling tubes on the cold-trap shell, the cold trap was removed from the loop and the 1/4-in.-dia tubes were replaced with 3/8-in.-dia tubes. The improved heat transfer to the cold-trap shell should make the cold trap more efficient and allow operation at lower temperatures.

The line sampler was removed and cleaned of accumulated sodium and sodium oxide. To improve the operation of the line sampler, the clearance that forms the annulus between the sample cup and overflow rim is being reduced so as to decrease leakage of sodium past this annulus.

Progress continues on the data-acquisition system for computer-programmed use. All raw data will be computed by means of punched-tape input to get usable data in final form of temperature, flow, ppm O₂, etc.

2. Reactor Mechanism and Instrumentation

a. Materials Evaluation

(i) Sodium Effects on Wear Properties (E. S. Sowa)

Last Reported: ANL-7460, pp. 33-35 (June 1968).

Wear tests were run with molybdenum test blocks in contact with carbonyl shafts in sodium and in argon. At 300°F, sodium exhibits lubricating action, so less wear is encountered than in argon. At 500°F, sodium "wets" the surface and accelerates wear until well into the 700°F range, where wear rates in both argon and sodium decrease significantly (see Fig. I.C.1). The plateau is believed to result from the formation of MoO₃, which serves as a lubricant. The low wear rate persists in sodium through the 1200°F range. However, wear in argon sharply increases at 1200°F, probably because of rapid removal of the oxide from the surface (MoO₃ has a high vapor pressure and sublimates at 1292°F). The sodium could prevent such removal by formation of sodium molybdate.

Fig. I.C.1. Wear of Molybdenum Blocks against a Carboloy Shaft in Sodium and in Argon

Data from wear experiments conducted to date are being processed to determine whether they can be fitted to a semiempirical wear equation.

An apparent correlation has been discovered between wear rate and the recorded curves of motor torque (which is directly proportional to the coefficient of friction) in each temperature range. When wear is increasing considerably, the motor torque curves "wander" noticeably, as is appropriate when larger fragments are being detached from the matrix of bearing material. The significance of this observation lies in the possibility of using torque data to monitor for incipient breakdown of bearings.

b. Instrumentation Development

(i) Out-of-core Neutron Flux Detection System (Intermediate Range Neutron Flux Monitor) (G. F. Popper)

Last Reported: ANL 7457, pp. 40-42 (May 1968).

Wide-range and intermediate-range circuits are being developed and will be tested with detectors installed in a special thimble in EBR-II. Environment and performance requirements for FFTF and future LMFBRs serve as development and test goals so that suitable systems for neutron-flux measurement will be available for such reactors.

(a) Detectors and Cables. The Westinghouse WX-31353-25 and the Boston Insulated Wire Co. Bostrad-22 cables have been tested up to 800°F in both air and dry helium (<5 ppm H_2O). Approximately 20 ft of each cable were exposed to the uniform high-temperature environment, ~3 ft were in a temperature gradient to room temperature, and the remainder was exposed to room temperature; the temperature profile approximates closely the conditions expected in the modified EBR-II "O" thimble.

The coaxial-type 0.187-in.-OD Westinghouse cable has a stainless steel center conductor, alumina insulation, and is hermetically sealed in a Type 304 stainless steel sheath; with a high-temperature HN-type connector at each end, it is 25 ft long overall. It is rated at 850°F with 1000 V dc applied; the capacitance was not to exceed 70 pf/ft, and the insulation resistance was not to be less than 5×10^{13} ohms/ft at room temperature.

The Bostrad-22 sample (BIW part #5734-C-G24) is a 0.270-in.-OD triaxial cable, is insulated with quartz braids, and has pure-nickel braided shields and stranded center wire. It is not hermetically sealed. We attached a Westinghouse WL-23096A high-temperature HN-type connector to the detector end and a standard low-temperature HN-type connector to the preamplifier end. The cable is 29.5 ft long and is rated at 1500°F with 1500 V dc applied. The capacitance was not to exceed 20 pf/ft, and the insulation resistance was not to be less than 1×10^{13} ohms/ft at room temperature. Its nominal impedance is listed as 75 ± 5 ohms.

The cables were installed in a "tube"-type furnace, and then one cable was attached alternately to either the electronic system that will be used for the high-temperature EBR-II tests (see Progress Report for March 1968, ANL-7438, pp. 28-29) or to a capacitance bridge. The capacitance bridge is a General Radio Co. Type 716-C, which allows measurements to be made with an accuracy of ~0.2% or 0.2 pf, whichever is greater, at 1 kHz and 1 MHz. At each test temperature, we recorded cable-insulation resistance with voltages from 100 to 800 V applied when possible, cable capacitance at 1 kHz and 1 MHz, preamplifier output pulse shape and size, and system noise.

The furnace was filled with dry helium for the first series of tests and with normal room-temperature air for the second series. No attempt was made to dry the air because the air is not dried in the EBR-II thimbles. The results of these two series of tests are shown in Figs. I.C.2, I.C.3, and I.C.4. Very few differences in cable performance were found between the helium and air environments.

Fig. I.C.2. How Temperature Changes Insulation Resistance in Westinghouse WX-31353-25 Cable

Fig. I.C.3. How Temperature Changes Insulation Resistance in Bostrad-22 Cable

Fig. I.C.4. How Temperature Changes Cable Capacitance

present in the air could not possibly have entered the Westinghouse sheath-type cable so quickly, even if the hermetic seal at the end was not present or was defective. Secondly, in dry helium between 200 and 700°F, cable or connector breakdown occurred with voltages as low as 380 V dc, but with air, 800 V dc could be applied without any evidence of breakdown at all temperatures except 800°F. Because of the very high packing density in the Westinghouse-type cable, breakdown was attributed to the connector construction. These factors are being studied further in relation to their effect on system performance in the EBR-II thimble test.

Cable capacitance did not change significantly with either environment or temperature. The Westinghouse cable capacitance did increase significantly at 700 and 800°F. The capacitance of the Westinghouse cable at 800°F in air with 1 kHz applied was measured as 63.1 pf/ft, which is ~18% greater than at room temperature. The capacitance of the Bostrad cable increased almost linearly with temperature (~6% at 800°F) with 1 kHz applied and appeared to level off or even decrease at the highest temperatures with 1 MHz applied.

Very low apparent insulation resistances were measured (see Figs. I.C.2 and I.C.3) at temperatures below 300°F, where it is believed that the moisture in the air significantly affects the high-temperature connectors; above 300°F, there was only a slight improvement in the insulation resistance when using air compared to helium. There are two reasons why the connector is suspected as the cause of the lower apparent insulation resistance rather than the cable itself. First, the insulation resistance of both cables decreased four decades in about 2 min after room-temperature air was allowed to enter the furnace after the cable had been exposed to an ambient-temperature vacuum for a short time to remove the helium. Although the Bostrad cable could be expected to behave this way because it is not hermetically sealed from the environment, the small amount of moisture

With a pulse of standard shape and size applied to the cable, the preamplifier output did not change with either type of cable, temperature, or environment (as determined by an $\sim 2\%$ accurate oscilloscope). The noise level in the furnace, which in all cases was below the alpha-pulse threshold, was found to be related to external electrical noise and not to test temperature.

Because the insulation resistance of the Bostrad-22 cable was ~ 50 times greater and its capacitance was ~ 3 times less than that of the Westinghouse cable at 700°F in air, the Bostrad cable has been selected as the first candidate to be used with the WX-4036 detector in the EBR-II high-temperature tests.

The gamma-pulse, dc, and rms sensitivities of both the Bostrad and Westinghouse cable-detector (WX-4036) combinations have been determined in gamma fluxes of $\sim 10^4$ and $\sim 10^6$ rad/hr from cobalt-60; thermal-neutron sensitivities have also been determined at fluxes of $\sim 10^4$ and $\sim 10^8$ n/cm²-sec.

None of the high-temperature guarded detector-cable combinations ordered from Westinghouse and Reuter-Stokes have been received as yet. The cost estimates for the manufacturers to test these detectors at high temperatures have been received and are being evaluated.

Because of the relatively high estimated cost for Westinghouse to modify the two WX-31369 fission counters, a decision to proceed or defer this work has not yet been made.

(b) Circuits. The Gulf General Atomic 10-decade neutron monitor was installed in the EBR-II. The standard low-temperature General Electric NA-04 fission counter supplied with the system was installed in a "J" instrument thimble. The entire system has been checked and is operating satisfactorily; there was no indication that extraneous electrical noises interfered with the system performance. As installed, the detector is estimated to be in an equivalent thermal-neutron flux of about 3×10^{10} n/cm²-sec at 45 MWt; before system performance can be evaluated further, the flux must be reduced by about a factor of 4 by withdrawing the detector to a position of lower neutron flux. After the ordered high-temperature guarded fission counters are received and tested, this monitor system will be used to evaluate the performance of an intermediate- or wide-range channel capable of being used with detectors operating continuously at 700°F .

A Hewlett Packard Model 7562A logarithmic converter (voltmeter) has been purchased for evaluation as a true rms log-N intermediate-range channel. This instrument has a dynamic range of 80 db (8 decades of neutron flux) and a voltage-range input of from 1 mV to 10 V or from 10 mV to 100 V (selectable by a panel switch); it has a frequency response of 50 Hz to 100 kHz with a ± 1 -db accuracy and a minimum

slewing speed of 60 db/sec. This meter, which is being used and evaluated as a part of the electronic system assembled for the cable-detector tests, is performing adequately.

(c) Miscellaneous. The alpha rms response of the test system has been predicted accurately by using known pulse parameters. The same techniques will be used to check the gamma and neutron rms responses.

A report entitled "Summary Review of Reactor Instrument Systems for Measuring Neutron Flux Out-of-Core in Liquid Metal-Cooled Fast Breeder Reactors (LMFBR)" is about 80% drafted. It reviews and evaluates information on various reactor instruments, systems, and concepts for measuring neutron flux out-of-core in terms of their availability or the research and development essential to their usefulness in the typical LMFBR out-of-core environment.

(ii) Fission-Gas Pressure Transducer Evaluation and Development (J. R. Folkrod)

Last Reported: ANL-7460, p. 35 (June 1968).

(a) Null-balance System. The performance and endurance tests on the ANL null-balance pressure transducer continue. The present test consists of cycling temperature between 400 and 900°F at a constant pressure of 350 psig; tests will continue at different constant pressures.

For these tests, the furnace is started at 400°F and then is heated slowly (over a 1-hr period) to 900°F, at which it is held for 1 hr. Then the furnace is turned off and a fan turned on to blow air through the furnace and cool the transducer to 400°F in 2.5 min. Thus a complete cycle requires 2 hr. The null-balance pressure transducer has shown no adverse effects in 25 cycles at 350 psig.

The pressure difference between the reference gas and the 350-psig process gas has been ~15 psig at 900°F; as the transducer is cooled, the pressure difference goes to zero. To generate further data on the relationship between temperature and pressure difference, the furnace will be allowed to cool and heat slowly during the 8-hr working day. This experiment has just begun. During the rest of the day, the automatic controls will continue to heat the transducer slowly and cool it quickly.

c. FFTF Instrumentation

(i) In-core Flowmeter Development (G. F. Popper)

Last Reported: ANL-7457, pp. 42-43 (May 1968).

(a) Magnet Thermal and Mechanical Tests. The magnetometer furnace test apparatus has been used to determine the reversible temperature effects up to 1200°F of Alnico-5, -6, and -8 rod magnets having

an L/D (length-to-diameter) ratio of 8. The results are shown in Fig. I.C.5. The changes are reported in terms of B_i , the intrinsic induction ($B_m - H_m$), rather than the more conventional B_d , the residual induction of the magnet.

Fig. I.C.5. Reversible Changes in Remanence of Rod Magnets with an L/D of 8. The bars show the maximum deviation from the mean for all samples tested.

As predicted, the smallest changes occur in the Alnico-8 alloy; Alnico-6 undergoes the greatest changes. Before the measurements were made, the magnet samples were temperature-stabilized to 1200°F.*

Table I.C.1 summarizes the material temperature effects on the residual induction, B_d , of Alnico alloys. These data are used only to estimate the initial magnitude of flowmeter flux density; accurate initial measurement of the actual flowmeter flux density is always possible and necessary.

The irreversible temperature effects will be summarized in a future report.

*Standard Procedure for Temperature Stabilizing Alnico Permanent Magnets, ANL Specification ANL-HTI-LMFBR-406, June 1968.

TABLE I.C.1. Temperature Effects in Alnico Permanent Magnets

Magnet Type	L/D Ratio	1000°F			1100°F			1200°F			1300°F			1400°F		
		Flux (Gauss)		Flux Change (%)	Flux (Gauss)		Flux Change (%)	Flux (Gauss)		Flux Change (%)	Flux (Gauss)		Flux Change (%)	Flux (Gauss)		Flux Change (%)
		Initial	Final		Initial	Final		Initial	Final		Initial	Final		Initial	Final	
Alnico-5	8	11,630	11,630	0	11,850	11,850	0	11,550	11,550	0	11,850	7,980	-32.7	11,730	3,630	-69.0
	4	8,500	8,500	0	8,550	8,250	-3.5	8,740	5,630	-35.6	8,580	3,300	-61.5	8,530	1,230	-85.5
	2	3,780	3,780	0	3,770	3,380	-10.3	3,570	2,090	-41.5	3,870	1,080	-71.3	3,530	394	-88.9
Alnico-6	8	10,070	10,070	0	9,930	9,930	0	10,730	10,730	0	10,100	10,100	0	10,130	5,700	-43.7
	4	7,800	7,800	0	7,830	7,650	-2.3	7,880	5,890	-23.3	7,870	4,550	-42.1	7,760	2,030	-73.8
	2	3,940	3,940	0	3,910	3,330	-14.8	4,020	2,060	-48.7	4,090	1,460	-64.3	3,950	646	-83.6
Alnico-8	8	8,190	8,190	0	8,080	8,080	0	8,100	8,100	0	8,220	8,220	0	8,140	8,140	0
	4	7,400	7,400	0	7,150	7,150	0	7,130	7,130	0	7,170	7,100	1.0	7,240	6,310	-12.8
	2	5,040	5,040	0	4,520	4,520	0	4,620	4,620	0	4,650	4,320	-7.1	4,630	3,120	-32.6
	1	2,100	2,100	0	1,640	1,640	0	1,700	1,700	0	1,810	1,730	-4.4	1,500	1,080	-28.0

(b) Prototype Permanent-magnet Flowmeter Tests. Procurement specifications have been developed for obtaining commercial permanent-magnet flowmeters for in-core use. Further progress awaits industry response to the request for bids on production of in-core flowmeters for the EBR-II instrumented subassembly.

(ii) Failed-fuel Location Method Development (F. Verber)

Last Reported: ANL-7460, pp. 35-36 (June 1968).

(a) Gas-disengagement Tests. The water-loop mockup was modified to include a gas loop and blower, and a 5/8-in.-dia cup was installed on the bottom of the 3/16-in.-ID bypass tube. Then, with water flowing at ~6.35 ft/sec (~15.5 gal/min) in the 1-in. tube, helium was injected in the water stream below the gas-disengagement chamber, and a mass-spectrometer leak detector was used to detect helium in the gas loop. Injections of 1 cm³ of helium produced no detectable helium readings in the gas loop, but 2-cm³ injections were detected in 4-6 sec; a 5-cm³ injection was detected in 6 sec.

A water pump has been added to the model. Similar gas-disengagement tests at greater water flows are in progress. All tests with the model are being performed with a once-through flow of water to permit more straightforward interpretation of data (a recirculating system would accumulate injected gas in the system due to gas entrainment, particularly at the higher liquid velocities). The data show successful disengagement of the injected helium gas; further tests and evaluations will determine the size and location of the gas-disengagement chamber bypass tube for the sodium loop, or whether the bypass tube can be omitted.

(b) Gas-injection Procedure for Sodium Loop. Tests with a Geiger-Mueller counter show that a krypton-85 concentration of ~0.0002 μ Ci/ml of helium will be readily detectable through the 0.001-in.-thick Mylar window in the monitor cell to be used in the gas loop. This measurement was used to determine the size of the gas-injection capsules (see Progress Report for June 1968, ANL-7460, p. 35) to be used with a cover-gas volume of 16 liters and the assumption that 1% of the gas in the

smallest capsule will enter the cover-gas system. Accordingly, the larger capsules will provide more gas if needed to demonstrate the gas-disengagement objective of the experiment.

3. Fuel Handling, Vessels, and Internals--Core Component Test Loop (R. A. Jaross)

Last Reported: ANL-7457, p. 45 (May 1968).

After the test fixture was modified to prevent sodium from bypassing the FFTF prototype fuel tube, the Core Component Test Loop (CCTL) has operated continuously 530.2 hr with only 2.8 hr of downtime because of minor electrical problems. The CCTL has achieved the 1060°F operating temperature desired for the testing of the FFTF prototype fuel tube, but not at the full flowrate of 400 gal/min because the lower face seal of the pump is leaking oil externally. The leakage rate is 45 ml/hr when the pump is operating at 400 rpm and increases with pump speed. Therefore, the pump is being brought to its full speed (950 rpm) in steps, with all pump oil being accounted for at each step. This conservative procedure is being followed in an effort to conclude the test of the initial FFTF prototype fuel assembly without having to shut down to replace the shaft seal.

Temperature profiles have been measured for the pump case, vapor-trap bodies, loop freeze plug, and vessel support stands. Strain gauges have been read at temperatures up to 1060°F. There has been no open circuit associated with 60 welded full-wave bridge gauges or 30 Rokide flame-sprayed single-element gauges. The statistical results of the piping strain gauge readings are not complete.

The new-type vapor traps* in the inert-gas piping connecting the test vessel, pump, and expansion tank have required no downtime or maintenance. There has been no increase in inert-gas pressure (an indication of plugging) since operation began June 28.

D. Systems and Plant--LMFBR

1. 1000-MWe Plant

a. Contract Management, Technical Review, and Evaluation (L. W. Fromm)

Last Reported: ANL-7460, pp. 36-37 (June 1968).

The work scopes for the five additional design-development studies were drafted and transmitted to AEC-RDT and the LMFBR Program Office for review and comment.

*Kimont, E. L., Engineering Development of a Practical 1200°F Sodium Vapor Trap, Trans. Am. Nucl. Soc. 8(1) 147 (June 1965) Abstract.

The ANL Quarterly Evaluation Report (January 1-March 31, 1968) was transmitted to AEC-RDT.

(i) The Babcock & Wilcox Co. Subcontract. Preparation of conceptual system design descriptions (CSDD) is nearing completion; they will be presented in the first three volumes of the report on Tasks II and III. Four draft CSDDs have been sent to ANL. All drawings of the reference-plant arrangement and component details have been completed.

The description of the parametric studies has been drafted; it will be in Volume 5 of the report on Tasks II and III. The Task-VA and -VB safety work has been reviewed internally by B&W, and will appear in Volume 3 of the report on Tasks II and III. Under Task IV, a list of all system R&D items was made; it categorizes the items by type of R&D required.

Supplemental contract agreement No. 2 has been approved by all concerned.

(ii) Westinghouse Electric Corp. Subcontract. Supplemental contract agreement No. 2 has been approved by all parties and is in effect.

Optimization studies of Task III are complete and the design conditions of the optimum plant have been established. The Task-III parametric-sensitivity work is nearing completion.

Rough drafts of 12 CSDDs of the optimum plant have been written. The reactor development portion of the Task-IV R&D program has been formulated and is being drafted. The report on Task-V safety work has been completed and is being edited.

(iii) General Electric Co. Subcontract. The report on Task I and a proposal for additional safety work have been received. The Task-III parametric studies have been completed and the report is being written. Final data have been generated for the Task-II reference design and the report is being drafted. Task-V safety work has been completed.

Material is being generated for the Task-IV report.

(iv) Combustion Engineering, Inc., Subcontract. Volumes I and II of the Task-II and -III report have been received by ANL. Volume I contains CSDDs of the reference concept, and Volume II contains static design and performance analyses. Volume III will describe the safety studies.

The contract supplemental agreement that expands the Task-IV R&D definition effort has been approved and is in effect.

Task-IV work on components, fuels, and materials has been directed toward defining the information needed for the R&D program.

(v) Atoms International Subcontract. Major Task-III analytical activities have been completed. Basic data have been generated for the parametric studies, but interpretation of this information is continuing. The DBA

and other-accident analysis and the nuclear uncertainty analysis are complete. Most of the plant arrangement drawings have been prepared; work on the process and instrumentation drawings is being resumed after having been delayed. Comparison of the reference reactor concept with a modular core is underway.

A supplemental contract agreement, which modifies Tasks I, II, and III and expands Task IV, has been drafted and transmitted to AEC for comment and approval.

E. EBR-II

1. Research and Development

a. EBR-II System Design Descriptions (E. Hutter)

Last Reported: ANL-7460, p. 38 (June 1968).

In addition to drafts of Volume II, Primary System, submitted previously, drafts on the following topics have been submitted for review: reactor vessel assembly, neutron shield, primary auxiliary pump, primary tank assembly, primary tank heating, and shutdown cooling.

b. Reactor Experimental Support--Reactor Analysis and Testing (R. R. Smith)

Last Reported: ANL-7460, pp. 38-47 (June 1968).

(i) Nuclear Analysis and Safety

(a) Preliminary Results of Static Tests Conducted during Run 29. In order to obtain additional information on hysteresis effects, measurements of power-reactivity decrement (PRD) were made during Runs 29A and 29C, when the reactor power was cycled between the following levels: 0 to 25, 0 to 41.5, and 0 to 45 MWt. In Run 29B the reactor power was cycled between 0 and 45 MWt only. The results of the PRD measurements are summarized in Table I.E.1.

TABLE I.E.1. Summary of PRD Values for Runs 29A, 29B, and 29C^a

Power Range (MWt)	Reactivity Change (lh)		
	Run 29A	Run 29B	Run 29C
0 to 25	34.0	-	48.0
25 to 0	-37.3	-	-51.6
0 to 41.5	48.5	-	79.0
41.5 to 0	-51.2	-	-79.2
0 to 45	51.5	57.5	84.4
45 to 0	-53.5	-59.7	-83.7

^aAll values normalized to an 11.00-in. rod-bank position.

The reactivity losses associated with the substitution of depleted uranium for stainless steel in Rows 7 and 8 amounted to 355 and 225 lh, respectively. The total of 580 lh compares favorably with

the value of 580 to 610 lh estimated from the results of similar measurements made at the Run-24-Run-25 interface when depleted-uranium sub-assemblies were replaced with stainless steel. In estimating the value of 580 to 610 lh, the effects of differences in plutonium content were considered.

Values of 8.3, 7.5, and 8.4 lh were established for Runs 29A, 29B, and 29C, respectively, for the reactivity change noted between the following conditions: 41.7 MWt with 100% flow, and 22.5 MWt with 54% flow.

With the stainless steel reflector, a reactivity gain of 4 to 6 lh occurred when the flow was reduced from 100 to 75% at 22.5 MWt (negative flow coefficient). With the depleted-uranium blanket the corresponding change in flow produced a reactivity loss of approximately the same magnitude (positive flow coefficient).

(b) Kinetics Studies of Run 29. All kinetics tests scheduled for Runs 29A, 29B, and 29C were completed. These included the following: rod drops, trapezoidal reactivity insertions (through the programmed movement of a control rod), and magnetic-tape recordings of oscillations in reactor power during periods in which the reactor was operated without manual control adjustments. In Run 29A, 17-min oscillations with a peak-to-peak amplitude of 2 MW were noted at 41.5 MW with full flow. At 22.5 MWt with 58% flow, the oscillations were characterized by a 17-min period and a peak-to-peak amplitude of 1 MW. Finally, at 23 MW with full flow, the period increased to 23 min and the peak-to-peak amplitude to 4 MW. After the replacement of the stainless steel reflector (for Run 29C), essentially no power oscillation was observed.

(c) Reactivity-monitoring Meter. Refinements to the reactivity-monitoring meter were made during Runs 29A through 29C. Various electronic multipliers were tested. The most suitable were installed, and the previously observed noise level was decreased significantly. Various amplifiers were also tested.

(d) Mark-IIA Safety Analysis. In a continuing effort to identify the most hazardous of the 19 hypothetical, but credible, Mark-IIA core loadings (defined in Table I.E.2), the following properties were calculated for each core:

(1) Peak Power Density. Power-density distributions were calculated for each core. Figure I.E.1 gives the peak power density as a function of core loading (the peak power density of each core occurred at core center except where indicated otherwise). Cores containing oxide experiments loaded at the core edge had the highest peak power densities. All peak power densities were considerably less than that calculated for a 60-subassembly core of the configuration used for the EBR-II wet-critical experiments. The peak power density calculated for that core at an assumed power level of 62.5 MWt was 1.12 MW/liter.

TABLE I.E.2. Core Loadings for EBR-II Mark-IIA Safety Analysis

Core No.	Composition of Central Core Region by Volume	Volume Fraction of Core in Central Region (%)	Composition of Outer Core Region by Volume	Volume Fraction of Core in Outer Region (%)	Comments
1	100% Mark-IIA	100	-	0	Base Case-- homogeneous core
2	90% Mark-IIA 10% oxide exp.	100	-	0	homogeneous core
3	70% Mark-IIA 30% oxide exp.	100	-	0	homogeneous core
4	50% Mark-IIA 50% oxide exp.	100	-	0	homogeneous core
5	100% oxide exp.	10	100% Mark-IIA	90	
6	100% oxide exp.	30	100% Mark-IIA	70	
7	100% oxide exp.	50	100% Mark-IIA	50	
8	100% Mark-IIA	90	100% oxide exp.	10	
9	100% Mark-IIA	70	100% oxide exp.	30	
10	100% Mark-IIA	50	100% oxide exp.	50	
11	90% Mark-IIA 10% carbide exp.	100	-	0	homogeneous core
12	70% Mark-IIA 30% carbide exp.	100	-	0	homogeneous core
13	50% Mark-IIA 50% carbide exp.	100	-	0	homogeneous core
14	100% carbide exp.	10	100% Mark-IIA	90	
15	100% carbide exp.	30	100% Mark-IIA	70	
16	100% carbide exp.	50	100% Mark-IIA	50	
17	100% Mark-IIA	90	100% carbide exp.	10	
18	100% Mark-IIA	70	100% carbide exp.	30	
19	100% Mark-IIA	50	100% carbide exp.	50

Fig. I.E.1

Peak Power Density as a Function of Core Loading: 91 Subassembly Cores; 62.5 MWt; Variable Driver Enrichment

(2) Critical Radius. Calculations reported in the Progress Report for April 1968, ANL-7445, p. 45, indicated that all cores can be made critical with a k_{ex} of at least 1.5% if the Mark-IIA driver-fuel enrichment is 61.5 a/o. Since the maximum permissible k_{ex} is 1.5%, a computation was made for each core to determine the number of subassemblies necessary to make each core critical with $k = 1.015$ (control rods fully inserted). The results are shown in Fig. I.E.2. These data indicate that all 19 cores can be made critical with $k = 1.015$ with between 63 and 92 subassemblies (including experiments). The number of experimental subassemblies accommodated in each core may be estimated from the data given in Fig. I.E.2 by multiplying the abscissa times the ordinate. The maximum number of experimental subassemblies thus far considered is therefore 46.

Fig. I.E.2. Total Subassemblies Required for $k = 1.015$ When the Driver-fuel Enrichment Is 61.5 a/o

(3) Peak Power Density for Condition (b). The peak power density was computed for each core by assuming a driver enrichment of 61.5 a/o and a core radius such that $k = 1.015$. The results are

presented in Fig. I.E.3. As expected, the smallest core (No. 1) had the highest power density (1.17 MW/liter). Cores with edge-loaded experiments had higher power densities than the others. For the same power outputs, Core No. 2 had a power density 4% higher than the wet-critical core referred to above.

Fig. I.E.3. Peak Power Density as a Function of Core Loading at 62.5 MWt and 61.5% Enrichment.

c. Nuclear Analysis Methods Development (P. J. Persiani)

Last Reported: ANL-7460, pp. 47-50 (June 1968)

(i) Physics Analysis

(a) Core-subassembly Worths. In a multifueled, nonuniform system such as EBR-II, it is necessary to structure calculations in such a way that each subassembly in the core can be dealt with independently of all others. The full representation of the EBR-II system in SNARG-2D (see Progress Report for March 1968, ANL-7438, pp. 43-46) was devised for this reason. A series of calculations have been run with this meshing to compare with the experimental values obtained during the attempts to bring Run 27-1 to criticality.

These attempts were made in four cumulative steps, beginning with the initial Run 27 configuration. Step 1 involved the replacement of half-loaded fuel subassemblies in locations 3E2 and 6E1; in step 2, the loading was returned to the configuration of Run 26C, except that the new fuel subassemblies used for Run 27 were retained. In step 3, experimental Subassemblies XA08 and XA05 were replaced by half-loaded fuel, the half-

TABLE I.E.3. Experimental and Calculated Values for Reactivity Adjustments during EBR-II Run 27-1

Step No.	Reactivity Change between Steps (1h)	
	Calculated	Experimental
1	-321.6	-283
2	149.6	175
3	109.6	66
4	-251.1	-271

loaded fuel in 4E3 was replaced by experimental Subassembly XO29, and the fuel in 5E2 was replaced by experimental Subassembly XO33. Finally, step 4 consisted of moving experimental Subassemblies XO15 and XO16 to 4A2 and 4D2, respectively, with half-loaded fuel subassemblies inserted in 3E2 and 6A1. The corresponding measured reactivity changes are compared with the computed values in Table I.E.3. Three of the four values are within 15% of experi-

mental values, whereas the fourth has an error of over 60%. These discrepancies are not fully understood at this time.

(b) Effect of Reflector and Blanket Systems on Core Loadings. The SNARG-2D twofold symmetric xy-geometry configuration previously described has been used for a typical EBR-II loading in order to determine the properties of a number of reflector-blanket systems. Figure I.E.4 contains a comparison of stainless steel and nickel reflectors with the blanket, for replacements ranging from Row 7 alone to the whole blanket area, Rows 7-14. These results show that both types of reflectors provide an increase of reactivity over the blanket in all thicknesses, with the nickel reflector being approximately twice as effective as that of stainless steel. The optimum balance between reactivity increase and the minimum reflector thickness is attained with approximately 4 rows of reflector material.

Table I.E.4 presents a measure of the fission rates in Row 11 of the blanket for several of these reflector-blanket systems. The heating rates show that the absorption and transport properties of the nickel reflector are similar to those of the blanket, whereas the stainless steel reflector permits larger rates, by amounts ranging from 25 to 50% with increasing thickness.

Fig. I.E.4. Reactivity Worths: Uranium Blanket vs. Nickel or Stainless Steel Reflector.

TABLE I.E.4. EBR-II Neutron Heating for
Various Reflectors at 45 MW

Reflector Type	Reflector Location	U-blanket Location	Neutron Heating for Row 11, W/cc
None	-	7-14	1.62
Ni	7-9	10-14	1.59
NI	7-10	11-14	1.60
SS	7-8	9-14	2.03
SS	7-10	11-14	2.37

As part of the investigation to increase the test space of EBR-II, a comparison was made between a fully uranium-blanketed system and a variety of systems containing nickel reflectors with the partial loading of fuel subassemblies in Row 7 replacing the blanket or reflector. The results of the computations, presented in Table I.E.5, indicate that the largest reactivity change, of about 10% $\Delta k/k$, is produced by a fully fueled Row 7 and a nickel reflector in Rows 8-10.

TABLE I.E.5. Reactivity Changes due to Variations in
the Thickness of the Nickel Reflector and
Row 7 Fuel Content, Δk

Locations of Fuel Insertion	Blanket and Reflector Location, by Row Number			
	Bl: 7-14 Ref: -	Bl: 9-14 Ref: 7-8	Bl: 10-14 Ref: 7-9	Bl: 11-14 Ref: 7-10
None (Base)	-	0.02331	0.03190	0.03677
7E1, 7B6	0.00260	0.02523	0.03399	0.03903
7E4, 7B4	0.00546	0.02791	0.03673	0.04180
7BN, 7EN ^a	0.02894	0.04782	0.05763	0.06335
7NN ^b	0.07137	0.08306	0.09415	0.10136

^a7BN means all of 7B1, ..., 7B6.

^b7NN means all of Row 7.

(c) Dynamics Study. Further modifications have been made in the static bowing-model code BOW-II. A damping factor has been inserted in the iteration scheme, eliminating the oscillations which previously occurred. The discrepancy between the allowable motions of assembly top and nozzle within the mechanical constraints, and the superposition of the two nonlinear distortions needed to bring both locations into contact has been resolved. This now allows a more consistent representation of the spatial modes with respect to constraint positions and loadings.

An additional modification of the program involves a beam-type deformation mode based on a perfectly rigid nozzle, which has been found to be overly restrictive for the critical Rows 7 and 8.

d. Fuel Performance Studies--Mark I Series (F. G. Foote)

Last Reported: ANL-7460, p. 52 (June 1968).

(i) Effect of Trace-element Impurities upon Fuel Swelling. As mentioned in ANL-7460, many fuel pins containing silicon in the range from 570 to 780 ppm (above the presently specified maximum of 520 ppm) showed unusually high axial growth even though the total swelling was low. Five additional fuel pins from casting batch 0456, irradiated in Subassembly C-2133 (a Mark-IB type) to a burnup of ~0.8 a/o, have shown the same effect.

Unusually high axial growth at low total swelling has been observed in only two melts (R410 and 0456). However, Subassemblies C-2154, C-2155, C-2156, C-2162, C-2163, C-2164, and B-3030, which were inserted into the core at the beginning of Reactor Run 29, all contain some fuel alloy with more than 520 ppm silicon. Casting batches 4257, 4258, 4261, 4263, and 4264 with silicon contents ranging from 660 to 920 ppm are represented in these subassemblies. The fuel-surveillance program has been modified to allow examination at lower burnups and at more frequent intervals. None of this fuel will be carried beyond a burnup of 0.9 a/o until swelling results at lower burnups are available.

(ii) Fuel Surveillance: Cold-line Fuel and Mark IB. Fuel elements from Subassembly C-2124, the first subassembly containing cold-line fuel, have been examined at a burnup of 0.65 a/o. Swelling values were quite normal and fit the previously established curve of swelling burnup.

Fuel elements from Subassembly C-2133, the second subassembly containing Mark-IB fuel elements, have been surveyed. Swelling values were quite normal. The only unusual effect was the high axial growths of pins from casting batch 0456 (discussed in the preceding section).

e. Mark II Driver Fuel Element Development (J. H. Kittel)

(i) Element Irradiation Tests

(a) Encapsulated Irradiations in EBR-II (M. A. Pugacz)

Last Reported: ANL-7460, pp. 56-57 (June 1968).

As of July 11, 1968, the experimental fuel capsules in Subassembly XO29 had been exposed to 2350 MWd of operation in EBR-II. The maximum calculated burnup was 1.36 a/o. Fifteen of the 22 replacement

capsules for Subassembly XO29 that were rejected for thin welds at the closures (see Progress Report for April 1968, ANL-7445, pp. 69-70) have been repaired and/or reprocessed, inspected, and shipped to Idaho.

(b) Encapsulated Irradiations in CP-5 (W. N. Beck)

Last Reported: ANL-7445, pp. 69-70 (April 1968).

The three prototype Mark-II elements being irradiated in CP-5 at maximum fuel temperatures of 710°C have reached a calculated burnup of 1.5 a/o. The irradiation temperatures, measured at the center of the fuel, are representative of EBR-II Mark-II fuel when the reactor is operating at a power level of 62.5 MW. The fuel pins will be examined periodically (nondestructively by neutron radiography) to observe dimension changes. Postirradiation studies will include a detailed examination of the fuel-clad interfaces for evidence of interaction.

Two of the elements had been instrumented with thermocouples to measure temperature at the center of the fuel. Although the life expectancy of small-diameter sheathed thermocouples (0.025-in. OD) is usually only a few hours, these thermocouples lasted five days. The central temperatures of both specimens agreed within 5°C, and the maximum temperature cycling was $\pm 7^\circ\text{C}$. The temperature information confirmed the preirradiation calculations of expected maximum fuel temperatures as well as the temperature drop from the center of the fuel to the outer diameter of the cladding.

A second capsule is being assembled to repeat this experiment. An attempt will be made to obtain additional information on the change of central-fuel temperatures due to fuel-pin swelling. The gas pressure in the plenum will be measured with a null-type pressure transducer.

(ii) Fuel/Cladding Compatibility Studies

(a) In-reactor Compatibility Studies (K. F. Smith)

Not reported previously.

Preliminary planning is in progress to assess the possible effect of irradiation on the compatibility of EBR-II fuels with stainless steel cladding materials. Present plans are to construct a column by alternating thin wafers of U-2 w/o Fs alloy, both depleted and enriched, with cylindrical pellets of Types 304 and 316 stainless steels. The column will be exposed in the MTR (or ETR) at temperatures of 500, 550, and 600°C, and the results compared with out-of-reactor tests of a similar column. The test temperatures were selected to bracket fuel-cladding

interface temperatures in EBR-II at power levels of 45 and 62.5 MW. Hopefully, 1000-hr tests will be sufficient to compare the effect of recoil flux as a function of enrichment in contact with the two types of steel.

The scheduled shutdown of CP-5 in November 1968 necessitated the use of another facility. The use of MTR for the test appears feasible.

(iii) Effects of High EBR-II Fluence on Properties of Stainless Steel

(a) Tensile Tests with Irradiated Type 304 Stainless Steel
(R. Carlander)

Not reported previously.

A program was initiated to tensile test Type 304 stainless steel from the guide thimble of EBR-II control rod 12 that was irradiated to a nominal fluence of 5×10^{22} n/cm² at estimated temperatures in the range from 371 to 480°C. The objective of the program is to determine the effects of fluence, test temperature, and irradiation temperature on the strength and ductility of Type 304 stainless steel exposed to flowing sodium in a fast-neutron environment.

Sheet tensile specimens were prepared from the irradiated guide thimble with a Tensilkut machine, using a solid carbide cutter rotating at 20,000 rpm. The specimens were $2\frac{1}{2}$ in. long and 0.040 in. thick, and had a gauge width of approximately 0.210 in. The specimens were tested at a constant crosshead speed and at a strain rate of 0.02 min^{-1} at temperatures of 450, 550, 650, and 750°C. Total elongations were measured between gauge marks spaced $1\frac{1}{2}$ in. apart. The preliminary results are given in Table I.E.6 in terms of true stress-true strain.

TABLE I.E.6. True Stress-True Strain Properties of Irradiated Type 304 Stainless Steel

Test Temp (°C)	Fluence (n/cm ² x 10 ⁻²²)	Irradiation Temp (°C)	Yield Stress (ksi)	Ultimate Stress (ksi)	Uniform Strain (in./in.)	Total Strain (in./in.)
450	0.4	460	36.0	75.0	0.169	0.176
450	2.0	480	64.6	77.9	0.056	0.072
450	2.0	480	64.6	80.2	0.072	0.080
450	2.0	371	75.0	78.3	0.025	0.036
450	2.0	371	85.1	91.5	0.027	0.039
450	4.8	420	95.6	100.3	0.020	0.027
550	<0.1	460	21.0	57.7	0.182	0.190
550	<0.1	460	25.7	66.0	0.180	0.189
550	0.6	480	41.8	62.8	0.122	0.132
550	0.6	480	41.9	64.8	0.125	0.139
550	2.0	371	69.6	72.3	0.020	0.031
550	3.6	460	70.5	75.6	0.021	0.025
650	0.4	460	25.4	40.4	0.086	0.095
650	1.4	480	33.0	39.3	0.048	0.065
650	1.4	480	34.9	42.7	0.045	0.069
650	2.6	480	37.7	42.6	0.029	0.034
650	2.6	480	35.9	40.4	0.025	0.038
750	<0.3	460	15.1	19.0	0.055	0.092
750	1.0	480	15.5	19.4	0.054	0.092
750	1.0	480	15.5	21.1	0.065	0.088
750	2.6	480	20.2	21.5	0.018	0.038
750	2.6	480	20.6	23.0	0.020	0.033

The results of tests at 450°C show the effect of fluence; both yield and ultimate strengths increased with increasing fluence. At a fluence of 4×10^{21} n/cm² and an irradiation temperature of 460°C, the irradiated yield stress was 36,000 psi, compared with 95,600 psi at a fluence of 4.8×10^{22} n/cm² and an irradiation temperature of 420°C. An increase in fluence of the same magnitude resulted in a decrease in uniform strain from 0.169 to 0.020 and a decrease in total strain from 0.176 to 0.027.

There was a marked dependence of irradiated strength and ductility of the material on irradiation temperature at a test temperature of 450°C (see Table I.E.6). Exposure of the Type 304 stainless steel to a fluence of 2×10^{22} n/cm² at 371°C resulted in irradiated yield strengths of approximately 80,000 psi, compared with approximately 65,000 psi at an irradiation temperature of 480°C. The ductility at 371°C was about half that observed at 480°C.

The irradiated yield strength and ductility of Type 304 stainless steel decreased as the test temperature was increased from 450 to 750°C. The yield strength of material irradiated to a fluence of 2×10^{22} n/cm² at 480°C and tested at 450°C was approximately 65,000 psi, compared with approximately 37,000 psi and approximately 20,000 psi for material irradiated to a fluence of 2.6×10^{22} n/cm² at 480°C and tested at 650 and 750°C, respectively. The irradiated uniform and total strains at the above irradiation conditions were approximately 0.064 and 0.076 at 450°C, compared with approximately 0.027 and 0.036 at 650°C and approximately 0.019 and 0.036 at 750°C.

f. Equipment--Fuel Related

(i) New Control Rod Subassemblies (E. Hutter)

Last Reported: ANL-7427, p. 60 (Feb 1968).

(a) Greater-worth Control Rod. So that up to four of the present 12 control-rod locations in EBR-II can be made available for experimental devices, a new concept for an absorber-follower control rod that has more reactivity worth than the present fuel-type control rod has been designed (see Progress Report for January 1968, ANL-7419, p. 60). The absorber-follower concept was chosen to optimize control-rod worth while minimizing the neutron-absorbing effect of the control rod on the reactor core during power operation; as the Mark-I-type 61-fuel-element bundle fuel section is lowered out of the core region, the upper boron-carbide absorber section moves in to reduce the reactivity further.

Fig. I.E.5. Absorber-follower Control-rod Design Concept

This new control rod consists of a standard subassembly top adapter (not shown in Fig. I.E.5), a cluster of seven tubes (~36.5 in. long, containing B_4C), a 3/8-in. flow-plenum space, the Mark-I-type 61-fuel-element cluster (~18 in. long), and the standard lower preassembly (not shown). Each absorber rod contains 732 gm of natural B_4C vibrationally compacted into a 14-in.-high space at the lower part of the rod (the B_4C density will be 75 to 85% of the theoretical 2.52-gm/cm^3 density; the boron-10 content in the natural B_4C is 19.8%), an 8-in.-high stainless steel shield section to protect the upper part of the reactor structure from radiation, and a 12-in.-high gas volume to accommodate the helium given off by neutron-induced transmutation of the boron. The ~36.5-in.-high stainless steel container tube has a 0.625-in. OD and a 0.555-in. ID; with a 0.030-in.-dia spacer wire wrapped helically around each tube, only a single top grid is needed to fasten the seven tubes into a cluster. When the reactor is operating at 62.5 MWt, heat generation in the B_4C will be 14.3 W/cm^3 , 5.5 W/cm^3 in stainless steel, and 7.6 kW/ft in the fuel (for a total of 519 kW in the 61 fuel pins); when sodium flows at a rate of 60 gal/min, the temperature will rise 4°F by the heat generated in the B_4C and in the stainless steel shield and containment tubes.

The maximum burnup that an absorber element is expected to be able to attain has been calculated to be 4.2 a/o. The calculation is based on conservative assumptions; with the absorber rods containing 80% dense B_4C inserted fully in the core, it is assumed that 100% of the transmutation-product helium would be released from the B_4C to the gas space and that 9400 psi is the maximum allowable

stress in the container tube (although the ASME code for pressure vessels allows the stress in stainless steel at 900°F to be 14,000 psi, 9400 psi was selected as the limit for this design to allow for uncertainties and acceptable deformation). The tube stress will not reach 9400 psi until the gas pressure inside the tube rises to 1138 psi, which would occur when the burnup of B_4C reaches 4.2 a/o, even if 100% of the helium formed is released to the gas space.

Consideration was given to using B_4C enriched in boron-10. However, although the absorber elements could withstand 4.2% burnup, they would sustain only 0.95% burnup during the time while the fuel section sustained its maximum allowable burnup of 1.2%. Because economics would require that the very expensive enriched- B_4C rods be reused, which would complicate control-rod design and handling due to the need for remote separation and reassembly, it was decided to employ natural B_4C , whose cost would not warrant reuse.

Detail drawings for a mockup of the B_4C cluster have been completed. When fabricated, this mockup will be flowtested in water to verify the pressure-drop calculations and to determine orifice sizes.

g. New Subassemblies Design and Experimental Support (E. Hutter)

Last Reported: ANL-7460, pp. 58-59 (June 1968).

(i) Mark-II Core Development. The Model-SCPS upper and lower shield pieces are being assembled for in-reactor tests of the prototype Mark-II driver fuel. After the irradiation results are available, further improvements or modifications will be considered.

(ii) Radial Blanket. Preliminary assembly and detail drawings were completed for the proposed nickel reflector subassembly. Heat generation and coolant flow requirements are being calculated. The resulting temperature relationship that affects subassembly bowing will be investigated for various reactor loading configurations.

Dummy hardware has been ordered for use in hydraulic flowtests.

(iii) Irradiation Subassemblies

(a) Mark-A. Mixing tests were performed with the flat-plate-type flow deflector. The results were not encouraging; the mixing was no better than with button-type deflectors. The objective is to achieve complete mixing of 1200 and 700°F sodium in a 4-in. space.

Studies are being made to determine the coolant flow necessary to provide an acceptable subassembly outlet temperature with no bypass coolant flow and a lower fuel temperature. The experimenter may compromise his high-temperature requirements to accomplish an irradiation in the near future.

(b) Mark-F. The special-size hexagonal tubes for use as the inner subassembly tubes in the Mark-F37 subassembly were shipped to the reactor site. Fabrication of the special grid assemblies has been nearly completed.

(c) Mark-H. Flowtests were made on a dummy Mark-H37 subassembly having 0.040-in.-dia spacer wires with a special 12-in. pitch. Dummy elements are being fabricated with a standard 6-in. pitch for flowtests to compare the pressure drops.

h. Process Chemistry (D. W. Cissel)

Last Reported: ANL-7460, pp. 61-62 (June 1968).

(i) Sodium Coolant Quality Monitoring and Control

(a) Primary Sodium

(1) Radionuclides. Duplicate samples in quartz beakers were taken from the FERD loop and the purification loop for comparative analysis. The results of analysis for ^{137}Cs and ^{131}I are listed in Table I.E.7.

TABLE I.E.7. Radionuclides in Primary Sodium

Sample Date--June 3, 1968

Sampling Point	^{137}Cs ($\mu\text{Ci/g} \times 10^2$)	^{131}I ($\mu\text{Ci/g} \times 10^4$)
Purification Loop	1.52	3.88
FERD Loop	1.51	4.20

(2) Trace Metals. A sample of primary sodium has been analyzed for trace-metal impurities by atomic absorption. The results are listed in Table I.E.8.

TABLE I.E.8. Trace-metal Impurities in Primary Sodium

Sample Date--June 26, 1968

Metal	Concentration (ppm)	Metal	Concentration (ppm)
Bi	1.3	Mg	0.14
Ca	<0.2	Mn	<0.13
Cd	<0.2	Ni	<0.6
Co	<0.5	Pb	11.5
Cu	<0.3	Sn	19.3
Fe	<0.6		

(3) Oxygen. A sample of primary sodium taken June 3, 1968, was analyzed for oxygen content by the mercury amalgamation method. The oxygen content was 10 ppm. Contamination of the sample is suspected, since the plugging temperature of the primary sodium was <225°F.

(4) Hydrogen. A sample of primary sodium taken June 3, 1968, was analyzed for hydrogen by the isotope dilution method. Duplicate analyses indicated 2.0 and 1.3 ppm.

(5) Carbon. A sample of primary sodium taken June 26, 1968, was analyzed for total carbon content by the oxyacidic-flux method. The average of five aliquots was 4.1 ± 0.4 ppm.

(6) Sulfide. An ion-specific electrode was used to analyze for sulfide in a sample of primary sodium taken May 22, 1968. Sulfide content was <0.1 ppb.

(b) Secondary Sodium

(1) Oxygen. A sample of secondary sodium taken June 4, 1968, was analyzed for oxygen by the mercury amalgamation method. The oxygen content was 9 ppm. The secondary plugging temperature at the time of sampling was 250°F.

(2) Hydrogen. A sample of secondary sodium taken June 4, 1968, was analyzed for hydrogen by the isotope dilution method. Duplicate analyses indicated 2.9 and 1.5 ppm.

(3) Sulfide. An ion-specific electrode was used to analyze for sulfide in a sample of secondary sodium taken June 14, 1968. Sulfide content was <0.1 ppb.

i. Experimental Irradiation and Testing (D. W. Cissel)

Last Reported: ANL-7460, pp. 63-65 (June 1968).

(i) Experimental Irradiations

(a) Status of Experiments in EBR-II. The status of the experimental irradiations in EBR-II as of July 31, 1968, is shown in Table I.E.9.

Temporary changes in location of several inner-blanket irradiations were made to facilitate reactor physics measurements during the exchange of the stainless steel blanket for depleted uranium in Runs 29A, B, and C. The changes were as follows:

<u>Subassembly</u>	<u>From</u>	<u>To</u>
XO10	7F3	7F1
XO35	7B4	Basket
XO37	7C3	Basket
XO38	7C5	Basket
X900	7A4	7E1

These irradiations were returned to their original positions before the start of Run 29D.

The NRL structural-materials capsules which were recently discharged were disassembled from experimental Subassembly XO26 and shipped to the TAN Hot Shop for postirradiation processing for the experimenter.

(b) Status of Irradiation Experiment Proposals Given Approval-in-Principle (AIP). Table I.E.10 summarizes the present status of all irradiation experiments which have been given AIP but have not been inserted in the reactor.

(c) New Experimental Subassembly Design. Flow tests in connection with modification of the Mark-A subassembly design to permit adequate mixing of high- and low-temperature shroud-tube effluent streams for the GE Group-5-program capsules have shown that mixing in the present 4-in. plenum length is apparently inadequate. Some additional flow tests will be made before final recommendation on a new or modified experimental vehicle is made.

(d) Other Work. Metallographic examination has been completed of a Type 304 stainless steel capsule containing seven melt-wire alloys which was held at 1100°F for about 18 days. The relatively small

TABLE I.E.9. Status of EBR-II Experimental Irradiations

Subassembly (Position)	Date Charged	Capsule Content and Number of Capsules (.)	Experimenter	Accumulated Exposure (MWd) 7/31/68 ^a	Estimated Goal Exposure (MWd)	Approx Accumulated Burnup (fuels--a/o) (nonfuels--nvt x 10 ⁻²²)
XG02 (7A1)	7/16/65	UO ₂ -20 w/o PuO ₂ Stainless Dummies (1) (18)	GE	15,582	16,700	3.3
XG03 (7D1)	7/16/65	UO ₂ -20 w/o PuO ₂ Stainless Dummies (2) (17)	GE	15,582	22,500	3.3
XG04 (7B1)	7/16/65	UO ₂ -20 w/o PuO ₂ Stainless Dummies (2) (17)	GE	15,582	42,000	3.3
XA08 (4F2)	12/13/65	UC-20 w/o PuC Structural (8) (11)	ANL ANL	11,976	19,800	7.4 4.4
XO10 ^b (7F3)	3/24/66	UO ₂ -20 w/o PuO ₂ Structural (4) (11) Structural (4)	GE ANL PNL	12,413	19,600	4.0 2.5 2.5
XO12 (4B2)	8/10/66	UO ₂ -20 w/o PuO ₂ (19)	NUMEC	8,352	20,100	5.1
XO15 (4A2)	11/15/66	UO ₂ -20 w/o PuO ₂ UO ₂ -20 w/o PuO ₂ UC-20 w/o PuC U-Fs (Mk-1A) (2)	NUMEC GE ANL ANL	6,697	11,000	4.0 3.7 4.1 2.1
XO16 (4D2)	1/13/67	Structural (9) Structural (10)	ANL GE	6,343	7,400	2.3 2.3
XO17 (4C3)	11/15/66	UO ₂ -20 w/o PuO ₂ UC-20 w/o PuC U-Fs (Mk-1A) (5)	NUMEC UNC ANL	6,697	6,500	4.0 4.1 2.3
XO18 (2B1)	12/6/66	Structural (3) Structural (1) Structural (2) Structural and Heavy Metal Fission Yield Sample (1)	GE PNL ANL ANL	6,983	21,300	2.9 2.9 2.9 2.9
XO19 (6D2)	1/13/67	UO ₂ -20 w/o PuO ₂ UC-20 w/o PuC Structural (8) Graphite (1)	GE UNC PNL PNL	5,769	7,500	2.1 2.4 1.6 1.6
XO20 (6B5)	1/13/67	UO ₂ -PuO ₂ UC-20 w/o PuC Structural (4) Structural (2) Graphite (1)	GE UNC PNL ANL PNL	5,769	7,500	2.1 2.4 1.6 1.6 1.6
XO21 (2D1)	2/25/67	Structural (7)	PNL	6,343	21,500	2.6
XO25 (4E2)	10/10/67	Structural (19)	GE	4,695	7,400	1.7
XO27 (4B3)	11/21/67	UO ₂ -25 w/o PuO ₂ Structural (1)	GE PNL	2,789	7,200	1.7 1.0
XO29 (4E3)	12/22/67	U-Fs (Mk-II) (37)	ANL	2,745	5,100	1.6
XO32 (6F1)	11/22/67	UO ₂ -25 w/o PuO ₂ (19)	PNL	3,092	11,000	1.3
XO33 (5E2)	12/22/67	UC-20 w/o PuC (19)	UNC	2,239	10,000	1.2
XO34 (2F1)	4/13/68	Structural (7)	ORNL	2,239	13,500	0.9
XO35 ^c (7B4)	4/13/68	Structural (7)	ORNL	1,822	40,500	0.4
XO36 (7E1)	7/25/68	UO ₂ -25 w/o PuO ₂ (19)	GE	187	30,000	0
XO37 ^c (7C3)	5/8/68	Structural (7)	INC	1,313	3,400	0.3
XO38 ^c (7C5)	5/7/68	Structural (7)	INC	1,313	16,000	0.3
XO41 (7A3)	7/24/68	Structural (7)	PNL	187	15,000	0
XO42 (7D5)	7/24/68	Boron Carbide Tantalum (4) (3)	PNL PNL	187	25,000	0
X900 ^d (7A4)	3/20/68	Structural (18)	ANL	2,525	2,700	0.5

^aAccumulated exposure during July was 543 MWd.^bRelocated to 7F1 for Runs 29A, B, and C.^cStored in basket for Runs 29A, B, and C.^dRelocated to 7E1 for Runs 29A, B, and C.

TABLE I.E.10. Status of Experiments with AIP

Sponsor	Material of Test	Number of Capsules	Suggested Reactor Row ^a	Date of AIP	Estimated Delivery
ANL	Oxide fuel, Group 0-3	19	3	9/22/67	12/68
	Carbide fuel, Group C-5	19	2	9/22/67	1/69
	Oxide fuel (unencap)				
	Safety Program	19	4	9/6/66	6/67 ^b
	Mark-II fuel (unencap)	455	Core	9/22/67	10/68
	Mark-II fuel (encap) ^c	25	4	6/3/67	6/68 ^b
	Metal fuel, M-4	19	3	2/13/68	10/68
	Structural, Group S-9	19	4	9/22/67	8/68
	Magnetic materials	7	7	6/8/67	8/68
	Instrumented subassembly	1	5	5/9/68	2/69
BMI	Nitride fuels	8	7	2/13/68	9/68
GE	Oxide fuels (unencap), Group 3	16	4	9/6/66	6/67 ^b
	Oxide fuels, Group 7	4	4	9/2/66	2/68 ^b
	FCR Structural Group L-15	4	2	1/4/68	11/67 ^b
	Oxide fuel, Group 5	16	4	1/20/67	10/68
	Oxide fuel (unencap), Group 9A	37	4	1/20/67	7/68
	Oxide fuel (unencap), Group 9B	37	6	5/9/68	9/68
	Oxide fuel (unencap), Group 9C	37	5	5/9/68	7/68
	Oxide fuel (unencap), Group 9D	37	7	5/9/68	8/68
LASL	Oxide insulators	1	8	6/6/67	9/68
	Carbide fuels	1	3	2/27/67	3/68 ^b
		6	3		9/68
NRL	Structural materials	7	7	9/6/67	10/68
ORNL	Oxide insulators	7	8	9/21/67	11/68
	Sol-gel oxide fuel	5	4	5/9/68	9/68
PNL	Oxide fuel (unencap low power)	37	3	9/21/67	8/68
	Oxide fuel (unencap normal power)	37	4	9/21/67	9/68
	Oxide fuel (unencap high power)	37	3	9/21/67	10/68
	Graphite materials	7	7	7/31/67	Indefinite
	Thermocouple materials	5	4	7/31/67	1/69
		2	4	7/31/67	2/68 ^b
	Oxide fuel, Group X	5	3	3/4/68	3/68 ^b
UNC	Carbide fuel (129-147)	19	7	6/6/67	9/68
	Carbide fuel (high power)	6	3	1/20/67	6/68 ^b
W	Carbide fuel (+ V-alloy)	2	3	9/21/67	10/68
	Carbide fuel (+ V-alloy)	9	4	9/21/67	Indefinite

^aBased upon fission rates at a reactor power of 45 MWt.^bDate received.^cReplacements for capsules of same type whose irradiation is nearing completion (e.g., Subassembly XO29).

increase in penetration compared to a similar capsule held at 1100°F for five days indicates that the rate of interaction beyond the initial five days is very slow. The examination established that these melt-wire alloys may be safely used with containment capsules of this type in reactor applications.

(ii) Procurement and Nondestructive Testing

(a) Preirradiation Inspections. The following capsules were X-rayed and recommended for approval for the intended irradiations: 15 experimental capsules containing metallic uranium fuel from the ANL M-2 series; 6 experimental capsules containing mixed carbides from United Nuclear Corporation; 7 structural capsules sponsored by PNL, containing boron carbide and tantalum samples.

(b) Neutron Radiography. A total of 32 irradiated driver-fuel elements which are involved in the Mark-IB and silicon surveillance programs were neutron radiographed at TREAT (see Sect. I.E.2.b).

The EBR-II oscillator rod was radiographed to determine the exact orientation of the boron carbide rod to facilitate its destructive examination. No physical changes in the B_4C were visible in the radiographs.

j. FCF Process Analysis and Testing (M. J. Feldman)

Last Reported: ANL-7460, pp. 65-68 (June 1968).

(i) Test and Analytical Methods

(a) Interim Examination Program. Out-of-cell testing of equipment for neutron radiography, bonding, leak testing, and profilometry continued. The foil- and specimen-transport mechanism for neutron radiography is ready for in-cell installation. Equipment was received, and out-of-cell tests were initiated for the prototype bond-test equipment for interim and postirradiation examination of experimental capsules. Preliminary efforts are directed toward measuring the quality of the sodium bond and height in the external bond of doubly encapsulated capsules. Later investigations will be conducted to determine the feasibility of measuring the quality and height of the sodium bond in the internal capsule.

k. FCF Equipment Improvement (M. J. Feldman)

Last Reported: ANL-7460, p. 66 (June 1968).

(i) Auxiliary Equipment. The pneumatic-tube transfer system between the FCF and the analytical laboratory was checked and worked very well. Travel time was 6 sec each way for the rabbits. The FCF terminal will be installed in coordination with the final installation of the transfer plug and master-slave manipulators in the transfer subcell.

The additional electromechanical manipulator carriage has been checked out in the mockup area. After several minor changes are completed, it will be ready for cell service.

(ii) Repair and Decontamination. Detailed designing of the equipment-decontamination facility continues. The manipulator-decontamination facility is progressing satisfactorily, and most of the steel has been erected. Provisions are being made to provide temporary crane service to the air cell if required before completion of the job.

1. Hot Fuel Examination Facility (N. J. Swanson)

Last Reported: ANL-7460, pp. 66-67 (June 1968).

(i) Sodium Lamps. We are continuing to investigate high-efficiency sodium vapor lamps for use in HFEF. The following table lists those currently of interest:

<u>Manufacturer</u>	<u>Manufacturer's Catalog Number</u>	<u>Type</u>	<u>Watts</u>	<u>Lumens per Watt</u>	<u>Status</u>
Osram (W. Germany)	Na200-W2	Tin coated	200	150	On hand
Osram (W. Germany)	Na60-W2	Tin coated	60	153	On hand
Osram (W. Germany)	Na180-W4	Indium coated	180	172	Production not yet started
Osram (W. Germany)	Na55-W4	Indium coated	55	134	Production not yet started
Philips (Netherlands)	126556	Indium coated	180	172	On hand
Philips (Netherlands)	126582	Indium coated	55	140	Ordered--not received

The starting voltage for 55- and 60-W lamps is about 420 V; for the 180- and 200-W lamps it is 660 V. The smaller lamps have a starting time of 6 min, the larger of 12 min (the time it takes for the lamp to reach 80% of full output after power is applied to a cold lamp). Restart times (warm lamp) were tested and found to be approximately 5 min for the Na60-W2 and about 3 min for the 126556 lamps.

Voltage breakdown tests have been completed of the sodium lamp bases and sockets received from Osram and Philips. Since the Osram and Philips bases appear identical, only the Osram bases were tested. The Philips sockets tested can be used with all of the listed Philips and Osram lamps. Tested breakdown voltages for air and argon at STP were as follows:

<u>Sample</u>	<u>Breakdown--Volts AC</u>	
	<u>Air</u>	<u>Argon</u>
Osram Socket 1	2600	1100
Osram Socket 2	3800	1100
Osram Socket 3	3200	1120
Osram Base 1	6600	1500
Osram Base 2	6900	1500
Osram Base 1 and Socket 2 Mated	4000	not tested
Philips Socket P1	6100	1900
Philips Socket P2	6100	1800
Philips Socket P3	7200	1850

The argon atmosphere during the tests contained 11 ppm H_2O and 1% N_2 . The oxygen content, measured with three different analyzers, gave readings of 16, 26, and 60 ppm; the exact concentration is unknown.

The ratio of the breakdown voltage of the bases (1500 V) to the maximum lamp-starting voltage (660 V) is rather low. A bigger ratio is normally provided. Also, because of the lower barometric pressure at the HFEF site, breakdown voltages in argon there would be approximately 10% lower than those listed above. However, since the environmental parameters are fairly well-known in this case, and are relatively constant, the lamp bases and sockets are probably acceptable. The ballasts normally used with these lamps are capable of starting them at temperatures as low as $-15^{\circ}C$. This is unnecessary for HFEF. Special ballasts could be made which would reduce the starting voltage of the 180-W lamps to 575 V or less. Such ballasts would be capable of starting the lamps at temperatures down to $5^{\circ}C$. Further evaluation of these parameters will be carried out. Special care in the selection of other connectors and in wiring practice also will be required.

A work plan for the radiation and life testing of the lamps is in progress. Radiation testing will start in August.

(ii) Overhead General-purpose Handling System. Studies have continued of the main cell overhead support system concepts mentioned in ANL-7460. Tentative design requirements of the system are being drawn up. These include such parameters as needs for backup capabilities, repair and maintenance needs, crane and hoist capacities, bridge, carriage, and boom speeds, crane hook coverage, manipulator coverage, types of motors and control, and maximum allowable deflections. A preliminary physical sizing was made of the crane carriage, hoist, and bridge components.

(iii) Revised Reference Configuration. The revised reference configuration has been reviewed, and a draft of comments prepared. Particular emphasis was put on evaluating the material flow paths and identifying handling problems.

(iv) Manpower Status. The manpower in the HFEF-Illinois Design Group is being increased to resume full activity on the design of cell equipment. When fully manned, there will be 10 Staff, 2 Salary, and 6 Hourly people in this group (some part time).

m. Superheater and EM Pump Study and Test

(i) Superheater Vibration Study (G. S. Rosenberg)

Last Reported: ANL-7457, pp. 82-84 (May 1968).

An accurate prediction of the natural frequencies of tube vibrations is important in assessing the potential for exciting resonant

vibration. It has been assumed (see Progress Report for October 1967, ANL-7391, pp. 104-105) that the baffles act as internal supports and prevent tube motion at baffle locations; actually, baffle-tube clearances allow the tubes to impact with the baffles. This impaction is being studied in relation to (a) the assumptions used in computing the natural frequencies, (b) its effect in altering other vibrational characteristics of the tubes, and (c) their response.

n. Nuclear Instrument Test Facility Study (B. C. Cerutti)

Last Reported: ANL-7460, p. 67 (June 1968).

(i) Out-of-core Thimble Modifications to Provide a Facility for Tests to 650/700°F. A GE compensated ion chamber (Type NA09) was installed in the test facility and data were taken for a neutron-flux plot of the thimble with the reactor at 45 MWt. The thimble was cooled to 125°F during the measurements. Table I.E.11 summarizes the data obtained.

TABLE I.E.11. Neutron Flux in Nuclear Instrument Test Facility

Distance above (+) or below (-) Centerline of Reactor Core (in.)	Neutron Flux (nv)
-4	4.5×10^8
0	4.5×10^8
10	4.5×10^8
20	4.2×10^8
30	3.3×10^8
40	1.8×10^8
50	8.0×10^7
60	4.0×10^7
100	7.5×10^6
140	1.7×10^6

o. Feasibility Study of Fuel Failure Detection--Chemical and Mechanical Methods

(i) Trace Elements Analytical Techniques (C. E. Crouthamel)

Last Reported: ANL-7460, p. 67 (June 1968).

In choosing appropriate tagged elements for the identification of failed fuel elements in EBR-II, the type and concentration of impurities already present in the sodium coolant must be known. As a part of this evaluation study, two samples of EBR-II primary sodium are being

examined for impurity content. Initial examinations of these samples by gamma spectrometry and mass spectrometry were reported previously (see ANL-7460).

The application of ion-exchange techniques to the separation of trace elements from the bulk sodium is continuing. This separation is necessary for gamma-spectrometric analysis because the large Compton scattering of ^{22}Na considerably reduces the detection sensitivity for impurities with low gamma energies. In recent tests, adsorption of impurities from a solution of an EBR-II sodium sample was attempted by contacting the solution first with Dowex-1 resin and then with Chelex-100 resin. Although analysis of the data is not yet complete, it appears that this treatment is effective in reducing the ^{22}Na content to a low enough level that radioactive impurities adsorbed on the resins can be determined. Additional samples of EBR-II sodium have been requested for further testing of this separation technique.

(ii) Sodium Loop--Tag Confirmation Study (F. A. Cafasso)

Last Reported: ANL-7460, p. 68 (June 1968).

Design, procurement, and assembly of components of a loop for evaluating the chemical stability of isotopic tags in circulating sodium is continuing. Fabrication of the sampling reservoir and cold trap is expected to be completed by September 1968. Design of support members for these components, and construction of an instrument console for controlling and recording loop operation have been started.

(iii) Mechanical Failed-fuel Locator (E. Hutter)

Last Reported: ANL-7460, p. 69 (June 1968).

(a) Design Test Loop. Assembly and detail drawings of the prototype mechanical failed-fuel locator are 95% complete. Design and drafting of the sodium test loop components are approximately 50% complete.

Safety analysis of the sodium test loop and its related components is underway; when completed, the analysis will be released for a review by a safety committee.

p. Study of Operation with Failed Fuel (R. R. Smith)

(i) Studies of Fuel Failure

(a) TREAT Irradiation of a Gas-bonded EBR-II Driver Element. Postirradiation analyses continue of the second TREAT experiment aimed at establishing the consequences of a sodium bond loss from

an EBR-II driver-fuel element (see Progress Report for June 1968, ANL-7460, pp. 42-43). The equivalent EBR-II power levels that the test pin would have experienced had it been at the core center in EBR-II are plotted in Fig. I.E.6. It may be seen that the power first reached a peak of 77.2 MWt, then decreased to 66.0 MWt, before increasing to a final maximum of 99.5 MWt. Duration of the run was 8.3 sec. Average power between the initial power peak and the time at which the TREAT reactor was scrammed was 80.5 MWt for a duration of 6.2 sec.

Fig. I.E.6. Power-Time Relationship for TREAT Experiment with Gas-bonded EBR-II Driver Fuel Element, ID-RP-2

Temperature profiles of the flowing sodium during the transient are plotted in Fig. I.E.7. The outlet thermocouple in the test specimen recorded an initial sodium temperature of 815°F (435°C), a peak of 1085°F (583°C) at the time of the TREAT scram, and a final temperature of 955°F (513°C). Sodium flowrate throughout the transient was approximately 4.3 m/sec.

Because of the high activity level of the tantalum sheath around the pin used to provide a more representative flux profile, the test assembly was allowed to cool for 28 days before the irradiated element was removed from the assembly. At the time of removal, the test specimen had a contact reading of 17 R/hr, and the element 1 R/hr. Though initial radiographs indicate that the fuel had melted, no effects in the exterior cladding were evident from visual inspection. The element appeared

to be straight and in its original condition, with no obvious diametral change. Micrometer readings taken at the top of the element, near the top, at the center, and near the bottom indicated outside diameters of 0.1742, 0.1745, 0.1750, and 0.1750 in., respectively. Readings taken with the element rotated 90° were identical. Tubing is supplied from the vendor with a 0.174 ± 0.0015 -in. outside diameter.

Fig. I.E.7. Temperature of Flowing Sodium in Test Specimen

q. Driver Fuel Transient Performance Studies--
TREAT Experiments

(i) Mark-IA Fuel TREAT Experiments (P. J. Persiani)

Last Reported: ANL-7460, p. 53 (June 1968).

Two unirradiated Mark-IA fuel pins were tested in TREAT to provide metallurgical control samples for this series of tests as well as

an in-pile check on the instrumentation of axial fuel motion in clad Mark-IA fuel using eddy-current motion transducers.

One pin, SL-33-3, was irradiated to integrated doses of 94, 143, and 172 Joules/gm fuel in three successive transients in the TREAT facility. Peak cladding temperatures for the center thermocouple, which were reached at 3.1, 3.3, and 3.9 sec, respectively, were 503, 630, and 657°C. In each case the pin was preheated electrically to approximately 200°C to ensure that the sodium bond was molten prior to the transient. Two eddy-current motion detectors were provided in the dry, transparent test capsule. The center planes of these transducers were positioned at the top and bottom fuel surfaces of the pin, with the sodium bond molten, preliminary to insertion into the test capsule.

After a one-day cooling period this pin was withdrawn from the test capsule. Thermocouples were welded to the second pin, SL-33-6, which was then equipped with the original motion transducers and heater, and inserted in the transparent meltdown facility. The second pin was irradiated in another series of transients to successive integrated doses of 172, 176, and 225 Joules/gm fuel. Peak cladding temperatures for these transients were 668, 714, and 918°C, respectively.

The motion detectors were operative for all transients. Extreme bowing and separation of the fuel-spacer wire were observed at the termination of the third transient on the first pin, due to mechanical constraint of the pin by the upper coil support from Lavite insulator. This was corrected after removal and replacement of that pin. Bowing was observed after the third transient on the second pin. Since there was no detectable separation of the spacer wire, it appears that this was from normal sample behavior at elevated temperatures rather than from mechanical constraint.

r. Systems Engineering

(i) Surveillance, Evaluation, and Study of Systems (R. A. Jaross and G. S. Rosenberg)

Last Reported: ANL-7457, pp. 67-68 (May 1968).

(a) Study of Sodium-Water Reaction Effects. This study is continuing on schedule, and will soon provide conclusions concerning the pressure response coupling to the superheater shell and the relief system.

To estimate the early-time loading on adjacent tubes, energy and mass balances have been written for growth of spherical reaction-product (assumed to be sodium monoxide and hydrogen) bubbles

in a field of unreacted sodium. The compliance of the unreacted sodium, represented by the relation between bubble-boundary pressure and the radial velocity of the surrounding fluid, can be obtained either from incompressible-flow theory or from a theory including compressibility (acoustic propagation) effects at early times. In either case, the determination of bubble growth requires simultaneous integration of the equations of motion of the bubble and the surrounding fluid.

A method has been developed for analyzing the propagation of disturbances through a pipe network representing a liquid-sodium circuit. The analysis is based on linear fluid-hammer theory. Elastic pipe dilation is assumed; fluid velocities are assumed to be small compared to the propagation speed, except possibly in constricted outflows through pressure-relief ports. A computer code was written in FORTRAN IV to perform the calculations for a network having up to ten nodes connected by pipe branches and with up to three branches meeting at a node. Structurally, the nodes are assumed to be rigidly supported externally. The program is a simple type of discrete-element code, in which each branch represents an element. This network analysis will make possible evaluation of the performance of pressure-relief provisions in the secondary sodium circuit of EBR-II.

The one-dimensional energy and mass balances for reaction-product zone growth have been generalized to apply to steam injection at a node having up to three branches. Simplifying assumptions compatible with those of the network response analysis were made. Integration of the product-zone equations will be performed by a subroutine to be coded for use with the network analysis.

(ii) Reactor and Primary System Improvements (B. C. Cerutti)

Last Reported: ANL-7460, pp. 57-58 (June 1968).

(a) Bus-bar Tips for Auxiliary Primary Coolant Pump.

The tips of the bus bars for the auxiliary primary coolant pump were reworked in March 1967. This rework consisted of jacketing the exposed copper tips, which were being severely corroded by the primary tank sodium, with a stainless steel sleeve. The success of this rework has been evaluated by removing one of the bus bars for inspection. The entire bus bar was thoroughly cleaned and visually examined for evidence of corrosion and/or cracking. The reworked tip and lower 40 in. of the bar was dye-penetrant inspected. The dimensions of the reworked tip were checked against the as-built dimensions. No corrosion, cracking, dimensional changes, or other damage was apparent following the year of operation. The inspection indicates that the revised design is a satisfactory solution to the problem of corrosion of the tip.

(b) Fuel-handling Operations. The control-rod gripper-jaw mechanical-interlock system for the control-rod-lifting platform was inadvertently damaged during fuel-handling operations, immediately after the removal of an oscillator rod. A detailed investigation was conducted to find the cause of the damage. The conditions present when the damage occurred were simulated in an attempt to duplicate the damage mechanism, but without success. The system has been restored to normal, and two new platform-interlock block driveshafts were fabricated and installed to replace the damaged units. The system operates normally, and no problems have been encountered since the incident.

(c) Rotating-plug Freeze-seal System. Seal alloy and dross from the rotating plug seals were sampled on July 24. The sodium concentrations in the seal alloy of the large and small troughs were 250 and 540 ppm, respectively, and the values in the dross were 5200 and 6500 ppm. Both troughs had been thoroughly cleaned and refilled several times during the past year, most recently on June 16 and 17. The sodium content of the seal-alloy samples from the large trough was about 10 times less than that of June 1967 samples, while the sodium content of the seal-alloy samples from the small trough was about 8 times greater than that of the June 1967 samples. The change in sodium content in the dross is comparable (see Progress Report for August 1967, ANL-7371, pp. 41-42). There is no ready explanation for the difference between the two seal troughs.

2. Operations

a. Reactor Plant (G. E. Deegan)

Last Reported: ANL-7460, pp. 75-76 (June 1968).

(i) Reactor Operations. The reactor was operated for 543 MWdt in Runs 29A, B, C, and D in July. The primary purpose was to conduct physics experiments related to the exchange of depleted uranium for stainless steel in Rows 7 and 8. Run 29A was conducted with the stainless steel still in place; 29B was conducted with nearly all the stainless steel subassemblies of Row 7 replaced by depleted uranium; and 29C was conducted with all the stainless steel subassemblies of Rows 7 and 8 replaced by depleted uranium. Run 29D is a continuation of power operation with the depleted-uranium blanket. The cumulated total of EBR-II operation is 18,588 MWdt.

The physics experiments of Run 29A began with a series of measurements of the power-reactivity decrement (PRD) while ascending to and descending from various power levels. Stainless steel rod-drop experiments were conducted at power levels between 0.5 and 45 MWt with 100% primary-system flow, and at 22.5 MWt with 58% flow. Criticals were measured with reduced power and flow, and trapezoidal reactivity-change experiments were performed at 22.5 MWt with 58% flow. Tests of power-level drift were carried out at various flowrates and nominal power levels while selected system conditions were recorded. Following the completion of these physics tests, the reactor was operated at 45 MWt until Run 29A was terminated at 185 MWd.

In the loading changes for Run 29B, Row-7 stainless steel subassemblies were replaced by depleted uranium except for two positions in which the steel subassemblies were retained. The physics experiments with this configuration included PRD measurements, critical measurements at 41.5, 30, and 22.5 MWt with 100% primary flow, and critical measurements at 22.5 MWt with 58% flow. The run was terminated at 24 MWd, and the stainless steel subassemblies of Row 8 plus the remaining two of Row 7 were exchanged for Row-14 depleted-uranium subassemblies. During fuel handling for Run 29C, the negative bus bar for the auxiliary primary coolant pump was removed, inspected, and reinstalled; no evidence was found of deterioration or cracks in the stainless steel sheathing.

With both Row 7 and 8 changed to depleted uranium, the physics experiments of Run 29A were repeated in 29C. The run was terminated after 205 MWd of operation.

Loading changes for Run 29D included the installation of several special flux-wire subassemblies to establish the neutron-flux

distribution in the core and inner blanket. After irradiation for 1 hr at 50 kW, these subassemblies were removed and replaced by standard subassemblies for the continuation of the run.

The loading changes during July are summarized in Table I.E.12.

TABLE I.E.12. Loading Changes in EBR-II during July 1968

Date	Grid Position	Removed ^a	Installed ^a	Comments
7/6/68	7A3	A-806	A-799	Exchange of depleted uranium for stainless steel in Row 7
	7A4	A-805	A-898	
	7A5	A-818	A-787	
	7B3	A-801	A-795	
	7B4	A-831	A-748	
	3B1	C-2136	C-2071	
	7B5	A-819	A-793	
7/7/68	7C3	A-814	A-766	
	7C4	A-821	A-798	
	4E1	C-2135	C-2052	
	7C5	A-837	A-896	
	7D3	A-809	A-789	
	7D4	A-815	A-735	
	7D5	A-813	A-773	
	7E3	A-816	A-899	
	7E4	A-823	A-796	
	7E5	A-824	A-794	
	7F3	A-804	A-792	
	7F4	A-811	A-786	
7/8/68	7F5	A-817	A-737	
	7A2	A-800	A-777	Reactivity adjustment
	7A6	A-826	A-782	
	7B2	A-807	A-783	
	7B6	A-827	A-784	
	7C2	A-834	A-778	
	7C6	A-828	A-790	
	7D2	A-803	A-779	
	7D6	A-829	A-791	
	7E2	A-810	A-780	
	7F2	A-825	A-781	
	7F6	A-833	A-897	
7/9/68	6A1	A-739	B-373	
	6E1	A-788	B-371	
	7E3	A-899	A-739	
	7A4	A-898	A-788	
7/11/68	8D5	U-1611	U-1428	Exchange of depleted uranium for stainless steel in Row 8 and remainder of Row 7
	14C11	U-1128	U-1611	
	8D4	U-1606	U-1128	
	14C10	U-1098	U-1606	
	8C5	U-1609	U-1098	
	14B12	U-1319	U-1609	
	8C4	U-1604	U-1319	
	14B11	U-1521	U-1604	

TABLE I.E.12 (Contd.)

Date	Grid Position	Removed ^a	Installed ^a	Comments
7/11/68 (Contd.)	8B5	U-1607	U-1521	Exchange of depleted uranium for stainless steel in Row 8 and remainder of Row 7
	14B10	U-1084	U-1607	
	8B4	U-1602	U-1084	
	14A10	U-1447	U-1602	
	8A5	U-1610	U-1447	
7/12/68	14F11	U-1303	U-1621	
	8F5	U-1608	U-1303	
	14F10	U-1014	U-1608	
	8F4	U-1603	U-1014	
	14E10	U-1412	U-1603	
	8E4	U-1600	U-1412	
	14D10	U-1428	U-1600	
	8A6	U-1615	U-1087	
	14A12	U-1056	U-1615	
	8A3	U-1621	U-1056	
	14B4	U-1449	U-1620	
	8B6	U-1612	U-1449	
	14B3	U-1032	U-1612	
	8B3	U-1618	U-1032	
	14A3	U-1087	U-1618	
7/13/68	8F6	U-1613	U-1221	
	14F12	U-1532	U-1613	
	8F3	U-1619	U-1532	
	14E12	U-1384	U-1619	
	8E6	U-1601	U-1384	
	14E11	U-1234	U-1601	
	8E3	U-1617	U-1234	
	14D12	U-1440	U-1617	
	8D6	U-1616	U-1440	
	14D11	U-1089	U-1616	
	8D3	U-1622	U-1089	
	14D3	U-1388	U-1622	
	8C6	U-1614	U-1388	
	14C12	U-1106	U-1614	
	8C3	U-1620	U-1106	
	8B2	U-1624	U-1514	
	14A5	U-1491	U-1624	
	8A7	U-1633	U-1491	
	14A4	U-1015	U-1633	
	8A2	U-1627	U-1015	
	14F3	U-1221	U-1627	
	14F1	U-1213	U-1639	
	8F7	U-1631	U-1213	
	14F4	U-1547	U-1631	
	8F2	U-1625	U-1547	
	14E4	U-1382	U-1625	
	8E7	U-1629	U-1382	
	14E3	U-1380	U-1629	
	8E2	U-1623	U-1380	
	14D4	U-1292	U-1623	

TABLE I.E.12 (Contd.)

Date	Grid Position	Removed ^a	Installed ^a	Comments
7/13/68 (Contd.)	8D7	U-1634	U-1292	Exchange of depleted uranium for stainless steel in Row 8 and remainder of Row 7
	14D3	U-1102	U-1634	
	8D2	U-1628	U-1102	
	14C5	U-1137	U-1628	
	8C7	U-1632	U-1137	
	14C4	U-1136	U-1632	
	8C2	U-1626	U-1136	
	14B2	U-1448	U-1626	
	8B5	U-1630	U-1448	
	14B5	U-1514	U-1630	
	8D1	U-1640	U-1178	
	14D5	U-1308	U-1640	
	8C1	U-1638	U-1308	
	14C6	U-1212	U-1638	
	8B1	U-1636	U-1212	
	14B7	U-1377	U-1636	
	8A1	U-1639	U-1377	
7/14/68	14F6	U-1325	U-1610	Installation of subassemblies containing melt wires
	8F1	U-1637	U-1325	
	14E5	U-1170	U-1637	
	8E1	U-1635	U-1170	
	14D6	U-1178	U-1635	
	7D4	A-735	A-899	
	8E3	U-1234	U-1212	
	9E4	U-1311	U-1195	
	12E6	U-1029	U-1194	
	16E9	U-1497	U-1398	
	7A3	A-799	A-898	
	7C1	A-802	A-735	
	7E6	A-830	A-799	
	6C1	A-723	B-3025	
	6D1	A-785	B-3027	
7/22/68	4F1	C-2165	C-2104	Removal of subassembly containing 70% enriched pins
7/23/68	7D4	A-899	W-6	New experimental subassembly
	7E4	A-796	XO36	
	7C5	A-896	XO38	
	7E2	C-2066	W-14	Installation of flux-wire subassembly
	7C3	A-766	XO37	Installation of flux-wire subassembly
	8D4	U-1128	W-7	
	5F4	C-2062	C-2130	Installation of flux-wire subassembly
	4A1	C-2134	W-15	
	3A2	C-2061	C-2120	
	7B4	A-748	XO35	
	7E1	X900	A-748	
	7A4	A-788	X900	
	7F1	XO10	A-788	
	7F3	A-792	XO10	

TABLE I.E.12 (Contd.)

Date	Grid Position	Removed ^a	Installed ^a	Comments
7/24/68	4B1	C-2166	C-2080	Removal of subassembly containing 70% enriched pins
	3C2	C-291	C-2106	
	8E3	U-1212	U-1234	
	9E4	U-1195	U-1311	
	2E1	C-2038	C-2107	
	12E6	U-1194	U-1029	Installation of flux-wire subassembly
	6D4	B-377	W-12	
	7A3	A-898	A-723	
	16E9	U-1398	U-1497	
	1A1	C-2162	W-13	
	3F1	C-2121	C-2103	Installation of flux-wire subassembly
	5B4	C-2070	W-16	
	7A3	A-723	XO41	
	7D5	A-773	XO42	
7/25/68	4A3	C-2006	C-2048	Removal of flux-wire subassemblies
	8D4	W-7	U-1128	
	4A1	W-15	C-2134	
	6D4	W-12	B-377	
	1A1	W-13	C-2162	
7/26/68	5B4	W-16	C-2070	Source rod
	7D4	W-6	A-723	
	3E2	W-14	C-2066	
	7E4	XO36	A-766	
	7E1	A-748	XO36	
	13E6	SO-1913		
	8A4		SO-1913	

^aA--Depleted-uranium inner-blanket subassembly.

S--Safety rod subassembly.

B--Row-6-type driver fuel subassembly.

X--Experimental subassembly.

C--Driver fuel subassembly.

OSR--Oscillator rod.

L--Control rod subassembly.

b. Fuel Cycle Facility (M. J. Feldman)

Last Reported: ANL-7460, pp. 76-84 (June 1961).

(i) Fuel Production

(a) Hot Line Operation. Table I.E.13 summarizes the production activities for July. Production has been limited because of high-silicon analyses in a few melt-refining runs and low yields in injection-casting runs. Analytical results from three recent melt-refining or melt-consolidation runs have indicated silicon concentrations of 1200 to 1400 ppm, well above the 350 ppm that could be expected on the basis of charge data. The cause of these results is being investigated by various means, including reanalysis of the product and charge material. The high-silicon material has been transferred to an off-site processing plant for recovery of the uranium.

TABLE I.E.13. Production Summary for Hot Line

	7/1/68 through 7/31/68	Total This Year
1. Subassemblies Received:		
Core, Control, Safety	6	66
Other	26	36
2. Subassemblies Dismantled (for processing)	2	64
3. Subassemblies Dismantled (for examination, etc.)	2	13
4. Subassemblies Fabricated	1	23
5. Subassemblies Transferred to Reactor:	6	41
Subassemblies Stored in L&O Vault and Interbuilding Corridor	0	0
Subassemblies Sent to ICPP for Storage	0	2
6. Elements Decanned:		
From Irradiated Subassemblies	731	4,519
Rejects	98	303
Other	0	161
Total Decanned	829	4,983
<u>Melt Refining</u>		
	<u>Irradiated Fuel</u>	<u>Recycle Material</u>
7. Number of Runs	6	3
8. Average Pour Yield, %	90.0	87.8
Total 1968 to Date	33	15
<u>Processing</u>		
	7/1/68 through 7/31/68	Total This Year
9. Injection-casting Runs (total number)	11	48
10. Elements Processed:		
Accepted	249	2,198
Rejected	130	476
11. Elements Welded:	153	2,124
Rewelded	0	0
12. Elements Leak-tested:		
Accepted	146	2,357
Rejected	7	57
13. Elements Bonded (including recycle)	213	2,977
14. Elements Bond-tested:		
Accepted	354	2,237
Rejected	46	214
15. Elements to Surveillance	9	1,532
Number of Subassemblies	1	39
<u>Waste Shipments</u>		
16. Cans to Burial Ground	2	40
17. Oxide and Glass Scrap to ICPP	1	22

The acceptable pin yields from the injection-casting and pin-processing equipment have been erratic, varying from low (less than 50%) to the normal values of 90 to 100%. The principal causes for rejection have been incomplete filling of molds, short castings, and brittle castings that broke during demolding or shearing. A systematic examination of all casting parameters of both acceptable and low-yield runs is being undertaken to correlate variations in these parameters with results.

The parameters that are being evaluated are casting temperature, quench-gas flowrate, casting gas pressure and rate, and analytical variations. Replacement of one injection-casting furnace that had malfunctioned is underway.

Stainless steel blanket subassemblies are being received from the reactor, washed, and transferred to the burial ground. External measurements on one of these subassemblies indicate no out-of-specification values. Additional subassemblies will be measured.

A report on the status, conclusions, and recommendations for the element tube-to-tip welding problem has been issued. The areas being investigated were detailed in ANL-7460. The recommendations are that the present method of leak testing this area of the element tubing with 60-psi helium is adequate for elements that undergo impact bonding, since the impact bonding introduces a transient high-pressure, high-temperature sodium pulse to this area of the tube as a result of deceleration during bonding. High-pressure (1200-psi) helium leak testing is recommended if a type of bonding operation (i.e., centrifugal) which does not produce high sodium pressure is employed. Another recommendation is that development of eddy-current inspection techniques continues.

(b) Cold Line Operation. Normal production is continuing on the cold line. A summary of the production data is given in Table I.E.14.

TABLE I.E.14. Production Summary for Cold Line

	7/1/68 through 7/31/68	Total This Year
1. Alloy Preparation Run:		
New Fuel	0	6
Remelts	7	38
Total	7	44
2. Injection-casting Runs	6	69
3. Pins Processed:		
Accepted	443	5,372
Rejected	18	290
4. Elements Welded	658	4,435
Elements Rewelded	0	14
5. Elements Leak-tested:		
Accepted	656	4,539
Rejected	2	57
6. Elements Bond-tested:		
Accepted	737	4,481
Rejected	15	288
7. Subassemblies Fabricated	3	39
8. Subassemblies Sent to Reactor	2	12

(Worked on Aerojet Qualification Elements during month.)

In addition to the normal production and fabrication activities involved in the cold line, acceptance inspections were continued for fuel pins and elements received from Aerojet.

(ii) Analyses(a) Product Analysis

(1) Chemical Analyses. The numbers of fuel-product analyses on hot line, cold line, and vendor fuel samples, together with the average values and ranges, were as follows:

<u>Analyzed for</u>	<u>Number</u>	<u>Average Value</u>	<u>Range</u>
U (total and isotopic)	19		
U (total)		94.36 w/o	93.94-95.31
²³⁵ U (% of total U)	19	52.07 w/o	51.94-52.26
²³⁹ Pu	7	1250 ppm	1014-1430
Mo	19	2.51 w/o	2.41-2.61
Ru	19	1.88 w/o	1.78-1.94
Rh	12	0.270 w/o	0.242-0.286
Pd	20	0.185 w/o	0.160-0.196
Zr	23	0.075 w/o	0.020-0.123
Nb	13	0.0115 w/o	0.0089-0.015
Si	70	532 ppm	136-1480
Al	15	129 ppm	56-283
C	11	227 ppm	138-346
Cr	11	54 ppm	22-163
Fe	25	455 ppm	81-1128
Ni	11	144 ppm	70-195
Total Analyses: 294			

Atomic absorption techniques are now being used in the chemistry analytical caves for Mo, Ru, Rh, Pd, Cr, and Ni analyses on hot-line fuel samples. As no chemical separations are required prior to these analyses, considerable time is saved with no sacrifice in precision and accuracy.

To perform these analyses remotely, a modified optical bar carrying hollow-cathode lamps, a burner system, and focusing lenses was installed in one of the caves. Light from the hollow-cathode lamps and burner system is led through the $2\frac{1}{2}$ -ft concrete cave wall by means of a lens system to the 0.5-m spectrometer outside the cave for measurement. All electronics and flow controls are located outside the cave for easy access. This system has been in operation for about three weeks with very satisfactory performance.

(b) Postirradiation Analysis of EBR-II Fuel

(1) Mark-IB Fuel Surveillance. The second EBR-II driver subassembly of the Mark-IB fuel-surveillance program reached target burnup in the reactor. This subassembly, C-2133, was irradiated during Runs 27 and 28 for a total of 2443 MWd in grid position 3C1 to a calculated maximum fuel burnup of 0.83 a/o (0.73 a/o average). Postirradiation tests were conducted in the FCF of all elements from the subassembly in order to appraise their metallurgical performance, especially irradiation swelling of the U-5 w/o Fs fuel pins.

Table I.E.15 presents pertinent fabrication data for the elements from C-2133 and summarizes results of postirradiation examinations. The swelling data, along with results obtained earlier for the first Mark-IB surveillance subassembly, C-2132 (reported in ANL-7460), indicate that swelling of Mark-IB elements following irradiation to approximately 0.85 a/o is similar to that indicated for large numbers of irradiated Mark-IA elements tested in the past.

TABLE I.E.15. Summary of Fabrication and Irradiation Data for Mark-IB Driver Subassembly C-2133
(Irradiated in Reactor Grid Position 3C1)

Injection-casting Batch Number	Silicon Content (ppm)	Average of Element Maximum Burnup (a/o)	Number of Elements Examined	Fuel Total Volume Swelling $\left(\frac{\Delta V}{V}, \%\right)$			Fuel-pin Length Increase (in.)			Volume Swelling Resulting from Length Increase (%)	
				Minimum	Maximum	Average	Minimum	Maximum	Average	Maximum	Average
0456	560	0.807	15	3.5	5.9	4.8	0.2	0.4	0.32	3.0	2.4
0457	230	0.804	15	3.7	4.9	4.1	0	0	0	0	0
4243	320	0.805	13	3.7	4.9	4.1	0	0.1	0.05	0.7	0.4
4245	150	0.801	20	3.6	5.5	4.5	0	0.1	0.04	0.7	0.3
4246	290	0.803	11	3.8	4.5	4.2	0	0.2	0.09	1.5	0.7
4247	130	0.808	15	3.4	4.8	4.3	0	0.1	0.01	0.7	0.1

As described in ANL-7460, a single U-5 w/o Fs injection-casting batch, No. 0456, displayed unexpected length increases following irradiation to approximately 0.5 a/o in the first Mark-IB surveillance subassembly, No. C-2132. Fuel pins manufactured from Batch 0456 were also present in the second surveillance subassembly, C-2133, just examined. As shown in Table I.E.15, significant length increases in fuel pins were again indicated (as determined by eddy-current bond tests) for fuel elements manufactured from this high-silicon (560-ppm) fuel batch. Five of the 15 fuel pins in C-2133 which were manufactured from Batch 0456 indicated length increases of approximately 0.4 in.; eight showed increases of approximately 0.3 in.; and two displayed increases of 0.2 in. The five other batches represented in C-2133 showed much smaller length increases, although total volume swelling of fuel pins for all batches was similar. Silicon content in the alloy of these other batches ranged from 130 to 320 ppm.

The variations of length increase in the irradiated fuel pins, which may be related to the increased silicon content in the U-5 w/o Fs alloy, will be studied further in a silicon surveillance program

in which the irradiation performance of the alloy with widely varying silicon contents will be investigated. The further investigation of Batch 0456 will include thermal cycling and dilatometer measurements.

Neutron radiographs of selected elements from Subassemblies C-2132 and C-2133 were made with TREAT as the neutron source. These radiographs, which confirmed the eddy-current indications, are presented in Figs. I.E.8 and I.E.9, and illustrate the upper region of the irradiated fuel elements. The radiographs show the top ends of the fuel pins and their positions relative to the bottom ends of the restrainers in the gas-plenum regions of the elements. It can be observed in the radiographs of elements from both C-2132 and C-2133 that the original, as-fabricated gap of approximately 0.4 in. between the fuel-pin top end and the lower region of the element restrainer has diminished to varying degrees

Fig. I.E.8. Neutron Radiographs of Mark-IB Driver Elements from Subassembly C-2132

Fig. I.E.9. Neutron Radiographs of Mark-IB Driver Elements from Subassembly C-2133

in elements containing pins from Batch 0456. In Element 53 from C-2133, intimate contact between the fuel-pin top end and the restrainer has apparently taken place, and some bowing of the restrainer stem has resulted, presumably from the compressive action on the stem by the elongated fuel pin. In contrast, little if any length increases are noted in Element 57 from C-2132 and Element 85 from C-2133. These elements were made with fuel from injection-casting Batch 0457 (230 ppm Si) and are typical of the lower-silicon batches (30-320 ppm) in the two subassemblies.

Careful inspection of the neutron radiographs indicates that some degree of waviness may be present in the longitudinal profiles of the fuel pins. However, no deformation of the element cladding has been discerned. Similar observations have been made recently with irradiated Mark-IA elements. With the advent of neutron-radiography capabilities in the FCF, additional observations are planned.

The third subassembly of the Mark-IB fuel surveillance program, C-2134, will be examined following irradiation to approximately 1 a/o. Irradiation is scheduled for completion at the end of Run 29. This subassembly also contains some elements manufactured from the same high-silicon batch (0456) as in C-2132 and C-2133.

(2) Cold-line-fuel Surveillance. C-2124, the first subassembly of the FCF cold-line fuel-surveillance program, has been examined following a calculated maximum fuel burnup of 0.68 a/o (0.59 a/o average). Preliminary data indicate that the irradiation swelling averaged about 2.8%. This value is in close agreement with that observed in the past for Mark-IA elements manufactured in the FCF hot line. Additional analysis of these data is in progress.

The second surveillance subassembly of this program, C-2121, is scheduled to complete its target burnup of 0.95 a/o at the end of Run 29C.

Optical metallography of as-cast fuel pins from FCF injection-casting Batch 4137 has been conducted to evaluate the microstructural effects caused by variations in cooling rates, position, and orientation of pins in the casting bundle. In all pins examined, the microstructural variations were more pronounced in the axial than in the radial direction (relative to the casting bundle). This results from solidification of the bottom ends of the pins at a slower rate than for the top ends, thereby producing a larger parent grain size. Following solidification, cooling at the bottom ends is also slower, resulting in greater precipitation and coalescence of secondary phases. There was little variation in gamma-grain size relative to radial position in the bundle in either the top ends or the bottom ends of the pins; hence, the radial cooling rate of the bundle

appears to be relatively uniform. Measured average grain diameters varied from 0.012 to 0.022 mm at the top ends and from 0.05 to 0.06 mm at the bottom ends.

Fuel pins from this batch were later irradiated in Subassembly C-2035 as part of Task 10, Mark-IA Anomalous Fuel Swelling Investigation.

PUBLICATIONS

How to Control Gas Quantities Being Fed to a System

W. F. Holcomb

Chem. Eng. 75, 216 (March 11, 1968)

High-Pressure Gas Bottles Made Safe by Orifice

W. F. Holcomb and T. W. Asbury

Chem. Eng. 75, 220 (May 6, 1968)

The Experimental Breeder Reactor II Shield Design and Performance

A. E. McCarthy and F. S. Kirn

AERE Intern. Conf. on the Physics Problems of Reactor Shielding, Harwell, September 26-29, 1967. AERE-R5773 (1968), Vol. 5, pp. 1373-1399

II. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Theoretical Reactor Physics--Research and Development

a. Cross Section Data Evaluation (C. N. Kelber)

Last Reported: ANL-7460, pp. 88-89 (June 1968).

(i) ENDF/B Processing. ENDF/B data tapes 114, 115, and 116 containing revised Category 1 data have been received from Brookhaven. The programs CHECKER and DAMMET were revised slightly to accommodate the revised format, which includes a table of contents. The data have been processed through CHECKER, and a start has been made at processing the data through DAMMET. The data will be processed through ETØE and MERMC2 to provide MC² library tapes to use for calculations of the ENDF/B fast reactor benchmark assemblies, such as ZPR-3 Assemblies 48 and 11.

(ii) ²³⁹Pu Evaluation. The analysis of the Petrel ²³⁹Pu fission data has been extended to 150 eV. A preliminary fit of the energy range from 40 to 150 eV has been obtained. The results of the final fit will be reported separately.

The code SIGMA, written by F. T. Adler and D. B. Adler, has been adapted for use with the system 360. This code computes any one of fission, total, or capture cross sections from the parameters of the other two. The code has been checked with theoretical data.

b. Reactor Computations and Code Development (B. J. Toppel)

Last Reported: ANL-7419, pp. 90-95 (Jan 1968).

A technique for variational flux synthesis has been formulated, using the multichannel approach and allowing in each channel or node for the existence of different synthesis regions, i.e., regions where different sets of trial functions are used as a basis for the expansion of the flux.

In the following example the essential features of the method are explained. The object of the calculation is to obtain an approximate expression for the three-dimensional time-dependent multigroup fluxes in the reactor schematically represented in Fig. II.A.1. Delayed neutrons have not been included for simplicity. The reactor is in this case divided into two concentric cylindrical volumes, which are the nodes or channels of the present example. A horizontal plane divides each of the cylinders into two synthesis regions. The geometry of the reactor is described by

cylindrical coordinates (r, θ, z) . A set of I mesh points z_i is defined along the z coordinate (see Fig. II.A.2); z_0 is the z coordinate of the bottom of the reactor, and z_I corresponds to the top plane.

Fig. II.A.1. Schematic Three-dimensional Representation of Reactor Volume

Fig. II.A.2. The Set of I Mesh Points along the Axis of the Cylindrical Reactor

The following synthesis assumptions are made for each channel, synthesis region, and energy group:

$$\phi_{ns}^g(r, \theta, z, t) \cong \sum_{m=1}^{M_{gns}} \sum_{i_s} \Delta_i H_{mns}^g(r, \theta) \psi_{mins}^g(t); \quad (1)$$

$$j_{ns}^{gr}(r, \theta, z, t) \cong \sum_{m=1}^{M_{gns}} \sum_{i_s} \Delta_i \left\{ \left[\Delta_{(i-1)}' D_{(i-1)}^g + \Delta_i' D_i \right] \frac{\partial H_{mns}^g}{\partial r} \alpha_{mins}^g(t) + H_{mns}^g(r, \theta) \beta_{mins}^g(t) \right\}; \quad (2)$$

$$j_{ns}^{g\theta}(r, \theta, z, t) \cong \sum_{m=1}^{M_{gns}} \sum_{i_s} \Delta_i \left\{ \left[\Delta_{(i-1)}' D_{(i-1)}^g + \Delta_i' D_i^g \right] \frac{1}{r} \frac{\partial H_{mns}^g}{\partial \theta} \eta_{mins}^g(t) + H_{mns}^g(r, \theta) \gamma_{mins}^g(t) \right\}; \quad (3)$$

$$j_{ns}^{gz}(r, \theta, z, t) \cong \sum_{m=1}^{M_{gns}} \sum_{i_s} \Delta_i' \frac{D_i^g}{h_i} H_{mns}^g(r, \theta) \left[\zeta_{m(i+1)ns}^g(t) - \zeta_{mins}^g(t) \right]; \quad (4)$$

$$j_{n(12)}^{gz}(r, \theta, t) \cong \Delta_{(I_0-1)}' \frac{D_{(I_0-1)}^g}{h_{(I_0-1)}} \left[\sum_{m=1}^{M_{gn2}} H_{mn2}^g(r, \theta) \zeta_{mI_0n2}^g(t) - \sum_{m=1}^{M_{gn1}} H_{mn1}^g(r, \theta) \zeta_{m(I_0-1)n1}^g(t) \right]. \quad (5)$$

Equation (5) applies only to mesh point $i = I_0 - 1$ at which the boundary between synthesis regions (within each channel) is located.

A similar set of assumptions is made for the adjoint magnitudes, with the exception of

$$j_{ns}^{gz*}(r, \theta, z, t) \cong \sum_{m=1}^{M_{gns}} \sum_{i_s} \Delta_i' \frac{D_i^g}{h_i} H_{mns}^{g*}(r, \theta) \zeta_{mins}^{g*}(t), \quad (6)$$

where

$g = 1, 2, \dots, G$	index for energy groups
$n = 1, 2$	index for channel
$s = 1, 2$	index for synthesis region
m	index for trial-function number
M_{gns}	number of trial functions in group g , channel n , and region s
i	index for z mesh point

$\sum_{i_s}$

summation over i index for its values in region s .

$$h_i = z_{(i+1)} - z_i$$

increment in the z mesh

 Δ_i

functions of z defined by:

$$\Delta_0 \begin{cases} 1 & z_0 < z < z_0 + \frac{h_0}{2} \\ 0 & \text{all other } z \text{ values} \end{cases}$$

$$\Delta_i \begin{cases} 1 & z_i - \frac{h_{(i-1)}}{2} < z < z_i + \frac{h_i}{2} \\ 0 & \text{all other } z \text{ values} \end{cases}$$

$$\Delta_I \begin{cases} 1 & z_I - \frac{h_{(I-1)}}{2} < z < z_I \\ 0 & \text{all other } z \text{ values} \end{cases}$$

 Δ'_i

functions of z defined by:

$$\Delta'_i \begin{cases} 1 & z_i < z < z_{i+1} \\ 0 & \text{all other } z \text{ values} \end{cases}$$

 D_i^g

diffusion coefficient in group g and $z_i < z < z_{i+1}$; all other material coefficients are defined in a similar manner

$$H_{mns}^g(r, \theta) \text{ and } H_{mns}^{g*}(r, \theta)$$

known functions of r and θ . M_{gns} of them defined for each group, channel, and synthesis region.

The quantities $\psi_{mins}^g(t)$, $\alpha_{mins}^g(t)$, $\beta_{mins}^g(t)$, $\eta_{mins}^g(t)$, $\gamma_{mins}^g(t)$, ζ_{mins}^g , and the corresponding starred quantities are unknown mixing functions of time to be determined.

The determination of the mixing functions ψ^g , ... is done in the usual way: the synthesis assumptions are substituted in the functional selected; the reduced functional is obtained by integration over θ and r ; stationarity is imposed to obtain the reduced Euler equations in the unknown function $\psi_{mins}^g(t)$, ...

A canonical functional of the type proposed by Selengut is used for the determination of the unknowns. It allows for zero-incoming-current boundary conditions and an arbitrary volume source for the adjoint flux.

The reduced Euler equations obtained are a set of simultaneous difference-differential equations relating values of the unknown functions at indexes $(i+1)$, i , $(i-1)$, and at both channels and the time derivatives $d\psi_{\text{mins}}^g/dt$.

The four-dimensional problem is thus reduced to a two-dimensional problem.

The general formulation developed can handle other geometries and coordinate systems; it also includes delayed-neutron precursors. At present the formulation is being extended to allow for discontinuous trial functions in the time dimension.

2. Nuclear Data--Research and Development

a. Burnup Analysis and Fission Yields for Fast Reactors

(i) Determination of the Absolute Fast Fission Yields of Burnup Monitors and Nonfission Nuclear Transformations in a Fast Reactor (R. P. Larsen)

Last Reported: ANL-7460, p. 91 (June 1968).

An irradiation to measure the effective capture-to-fission ratio (α) for plutonium-239 in a soft neutron spectrum is planned by the Reactor Physics Division [see Sect. I.B.4.b.(ii)], with supporting work being performed by the Chemical Engineering and Chemistry Divisions.

Because the very low integrated neutron flux (about 10^{17}) in the irradiation will produce relatively few captures and fissions, the capture rate will be measured with samples of high-purity ^{239}Pu (<10 ppb ^{240}Pu) and the fission rate with samples of normal plutonium. This technique will minimize handling, and therefore the probability of cross contamination, of the high-purity ^{239}Pu . In preparation for the measurements, it was necessary (1) to prepare and encapsulate adequately samples of high-purity ^{239}Pu , (2) to develop a mass-spectrometric technique that would enable a change in ^{240}Pu content of 100 ppb to be measured accurately, and (3) to develop a technique for measuring the number of fissions accurately.

Details of the preparation of samples for the capture measurements and of the method for measuring the number of fissions are discussed below.

(a) Preparation of High-purity ^{239}Pu Samples for Irradiation.

The preparation of the high-purity ^{239}Pu for capture measurements was carried out in the Chemistry Division isotope separator;* the starting material was 10 mg of ^{239}Pu with a ^{240}Pu content of 8 ppm. The purified material and the aluminum foil target on which it was mounted were dissolved in hydrochloric acid and the solution converted to a nitrate medium. The plutonium was oxidized to the hexavalent state with silver (II) oxide and extracted into ethyl acetate. The ethyl acetate solution was filtered, evaporated to dryness, and the plutonium dissolved in a small quantity of nitric acid. About 500 μg was recovered. A preliminary mass-spectrometric analysis showed the ^{240}Pu content to be <35 ppb. A blank which was carried through the entire procedure was assayed for alpha activity and analyzed mass spectrometrically; both tests demonstrated that there was no plutonium in the reagents.

Seven 50- μg portions of the prepared plutonium were evaporated to dryness on stainless steel discs, which were doubly encapsulated in aluminum by cold welding between circular sheets of aluminum foil. The encapsulated discs then were loaded into an irradiation container along with the fission foils (13-mg discs of normal plutonium encapsulated in aluminum). The remaining 150 μg of high-purity material will be used for preirradiation mass-spectrometric analysis.

To minimize the possibility of cross contamination from normal plutonium, the chemical separations and preparation of the capture foils were carried out with new reagents and glassware in a laboratory of the Biological and Medical Research Division that had no previous history of plutonium work.

(b) Measurement of the Number of Fissions. To measure the number of fissions that occur during the irradiation, the normal plutonium foils will be assayed with a lithium-drifted germanium diode (GeLi) detector for ^{140}Ba activity (half-life of 13 days) by counting the 1.59-MeV gamma emitted by the 40-hr ^{140}La daughter; the assay will be made about 21 days after the irradiation.

To establish a relationship between ^{140}Ba counts and the number of fissions, a short irradiation (~ 2 hr) will be carried out prior to the longer irradiation (~ 5 days) for which the alpha measurements will be made.

The short irradiation will be made with fission foils of the type to be used in the long irradiation and on fission track counters. The fission track counters will consist of mica discs alternately stacked with platinum discs on which are mounted nanogram amounts of plutonium (the amount will be determined by alpha counting). The fission foils will be assayed for ^{140}Ba and plutonium; the mica discs will be treated with

*This work was done by J. L. Lerner of the Chemistry Division.

hydrofluoric acid, and the number of fission tracks counted under a microscope. From the ^{140}Ba counts, the number of fission tracks, and the weights of plutonium in the fission foils and fission counters, a fission/count factor can be calculated. This factor will be used to determine the number of fissions that occur in the fission foils in the long irradiation.

Two preliminary short irradiations have been carried out to evaluate this method of calibrating the detector-fission foil system for determining numbers of fissions. The fissions/count factors calculated from the data agreed within 3%. An examination of the various parameters in the preliminary calibration experiment indicates that the final calibration can be carried out in a manner which will reduce the error to less than 1%. By increasing the length of the irradiation by a factor of 5, the statistical error in the ^{140}Ba counting can be reduced from 1 to 0.5%; by adjusting the amount of plutonium on the platinum plates and dispersing the plutonium more uniformly on the platinum, the statistical error in track counting can be reduced from 2 to 0.5%.

(ii) Supplemental Determination of the Absolute Fast Fission Yields of Burnup Monitors for Fast Reactor Fuels
(R. P. Larsen)

Not reported previously.

An irradiation of 100-mg quantities of various fissionable isotopes is being conducted in EBR-II with the objective of obtaining materials for the determination of fast-fission yields of isotopes that are useful burnup monitors. (This irradiation is being carried out in a special sub-assembly, XO18, which is largely composed of samples of various structural materials.) The fissionable materials when discharged from the reactor will have burnups of 20 to 25%. The number of fissions will be determined by pre- and postirradiation mass-spectrometric analysis of the heavy elements. It is anticipated that the number of fissions can be determined with an accuracy of $\pm 0.5\%$ and fission yields with an accuracy of $\pm 1.0\%$. It was originally expected that the irradiation, which began in December 1966, would be completed by 1970. It now appears that the irradiation will not be completed before 1973, and work could not begin on these samples (because of cooling requirements) until 1974.

To provide the much-needed fast-fission yield data at an earlier date, plans are being made to carry out an irradiation of 100-mg quantities of the various fissionable isotopes (^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu) in driver subassemblies in EBR-II. This irradiation should be completed by 1970; the burnup will be about 3%. The number of fissions that have occurred in each material will be determined by summing the number of atoms of heavy-element fission products produced. This involves four determinations (total rare earth, cesium, barium, and xenon) and an

estimation of the concentrations of three of the minor heavy-element fission products (iodine, antimony, and tellurium) from existing fission yield data. This approach was not feasible when the long-term irradiation was initiated since suitable analytical techniques were not available for determining the rare earths (50% of the heavy fission products). However, the development of the total rare earth and the X-ray spectrometric rare earth methods (see Progress Report for May 1968, ANL-7457, pp. 97-98), along with established methods for cesium, barium, and xenon, will enable the total number of heavy fission product atoms to be determined with the requisite accuracy. It is estimated that the number of fissions can be determined by this summing technique with an accuracy of $\pm 1\%$, and that the values for the fission yields will have an accuracy of $\pm 2\%$.

b. Determination of Nuclear Constants (A. B. Smith)

Last Reported: ANL-7457, pp. 98-103 (May 1968).

(i) Fast Neutron Cross Sections and Other Nuclear Constants

(a) Scattering of Neutrons from Germanium. Elastic and inelastic neutron scattering cross sections of natural germanium were measured in the incident neutron-energy range from 0.3 to 1.5 MeV with incident energy resolutions of ~ 20 keV. Neutron-velocity and gamma-ray-spectroscopic methods were employed to resolve the scattering to individual low-lying states. The results were interpreted in terms of the optical model and the statistical-compound-nucleus model. Cross sections for inelastic neutron scattering from a number of excited states in even germanium isotopes were found to be reasonably well-described by the unmodified statistical-compound-nucleus model of Hauser and Feshbach.

(b) Scattering from Deformed Nuclei of Hf, Sm, and Yb.

This work is nearing completion. Experimental measurements are essentially finished. All available data has been analyzed and cross sections obtained. The complex interpretive procedures are now well along. A representative result pertaining to the cross section for the inelastic excitation of the first (2+) state at 0.094 keV is shown in Fig. II.A.3.

Fig. II.A.3

Cross Section for Inelastic Excitation of the First (2+) State in Hafnium. Experimental values are indicated by crosses. The two curves are: a) ABACUS, results of statistical-model calculations using spherical well, b) 2-PLUS results of calculations using a deformed well.

(c) Fast Neutron-capture Processes

(1) ^{238}U Capture Cross Section at 23 keV. A reinterpretation of a spherical shell transmission experiment by Belanova et al. using the Monte-Carlo code AMC has been completed. The final corrected cross section was 495 mb, compared with 412 mb reported by Belanova. This new value agrees very well with recent measurements in the range from 25 to 500 keV by Menlove and Poenitz if a correction for intermediate structure is applied.

(2) Isomeric Cross-section Ratio for Rhodium. Measurements of the isomeric cross-section ratio ($T_{1/2} = 4 \text{ min}/T_{1/2} = 40 \text{ sec}$) were carried out in the region from 20 to 1400 keV. Strong influence of the inelastic scattering was observed due to the 40-90-keV levels and the 530-keV level.

No intermediate structure could be observed in the energy range from 120 to 250 keV where Seth reported such structure in the total cross section. The analysis of these measurements will be completed shortly.

(3) ^{236}U (n, γ). Neutron activation measurements have been made at incident neutron energies of 30 and 480 keV with samples in the form of metal discs of 3/8-in. diameter and 0.010 in. thick. Rough values of the cross section were deduced from observed (60) (208) keV gamma-ray coincidences among transitions in ^{237}Np , the daughter of ^{237}U β -decay. A NaI(Tl)-Ge(Li) detector system was used. Calibration of the system response was made with a calibrated thick ^{237}U β -decay source (^{237}U in equilibrium with a ^{241}Pu source of calibrated α activity). The most serious problem at present is proper handling of the effects of gamma-ray attenuation in the sample, relative detector efficiency, and angular correlation. Because these are complex and not necessarily well-known functions of the emission angles, more extensive calibration will have to be made under conditions more closely approximating the experimental situation. The preliminary results are:

E_n , keV	$\pm \Delta E_n$, keV	$\sigma(n, \gamma)$, mb	$\pm \Delta \sigma(n, \gamma)$, mb
30	15	460	230
480	50	141	70

The large uncertainties in $\sigma(n, \gamma)$ are considered conservative and reflect possible errors associated with various assumptions used to estimate the above-mentioned effects.

(d) Total Neutron Cross Sections

(1) Total Neutron Cross Section of ^7Li . Total neutron cross sections of ^7Li were measured over the energy range from 100 to 650 keV with 2.5-keV resolution. The data were fitted using a multilevel formula and assuming purely scattering resonances. A good fit was obtained by assuming that only one resonance at higher energy produced a significant effect. The parameters assumed and calculated were

$$\text{Nuclear Radius} = 0.336 \times 10^{-12} \text{ cm}$$

	<u>Assumed Resonance</u>	<u>Unknown Resonance</u>
E, keV	1220	261.2 ± 0.1
Γ , keV	1000	35.9 ± 0.1
J, parity	1^-	3^+
ℓ	0	1

The errors were obtained from the scattering of the data about the calculated curve and do not include any systematic or experimental effects.

(2) σ_T Measurements in the Range from 0.6 to 1.5 MeV.

Automated total cross-section measurements with the monoenergetic beam have continued in the high-energy range (0.650 to 1.5 MeV) using the scintillator detecting system with gamma-ray and second neutron-group discriminations. The ^7Li isotope was measured from 0.650 to 1.5 MeV in 2-keV increments. Natural uranium was measured from 0.650 to 1.5 MeV in 4-keV increments, and spot measurements were made for hafnium at 0.650, 0.850, 1.050, 1.250, and 1.450 MeV.

In an effort to compare the precision of the two measuring systems, i.e., BF_3 counters from 0.100 to 0.650 MeV and scintillation detector from 0.650 to 1.5 MeV, a comparison of the cross section at the crossover point (0.650 MeV) from both methods was made. The matchup for the hafnium and the natural uranium was very good at this point, but the match for the ^7Li isotope was only fair (within 10%). A projected matchup for corresponding measurements on carbon indicated a very good match. The fact that two different samples were used may explain the slight mismatch obtained with the ^7Li isotope.

(e) $^7\text{Li}(p,n)^7\text{Be}$ Cross Sections. Neutron angular distributions have been measured with a hydrogen-recoil counter at about 50 energies in the region $1.9 \leq E_p \leq 2.7$ MeV. Relative integrated cross sections were also measured at about 30 energies in the region $1.89 \leq E_p \leq 3.09$ MeV with a flat-response 4π neutron detector. In addition, the absolute integrated cross section was measured at $E_p = 2.0$ MeV. The integrated yields determined in the two groups of relative measurements agree well with each other (within 3%, below the threshold of the second neutron group),

and they, together with the absolute measurement, are in good agreement with the integrated cross sections previously measured by Machlin and Gibbons at ORNL.

(ii) Instrumentation and Digital Techniques

(a) Gray Neutron Detector Efficiency as a Flux Monitor.

The relative efficiency of the gray neutron detector has been measured using the associated activity method. Vanadium, copper, and lithium targets were used. An agreement within 2% with theoretical calculations has been found.

Measurements of the $\text{Li}(p,n)$ cross section showed an agreement (within $\pm 3\%$) with results from recoil-proton measurements and the graphite sphere detector. These results are particularly pertinent to our recent precision determinations of $\sigma_f(^{235}\text{U})$.

(b) Fission Detector. A newly developed double-chamber gas scintillation fission detector was tested with ^{235}U foils. The results promise accurate ^{235}U and ^{239}Pu fission cross-section and cross-section-ratio measurements in the keV energy range. Preparations for such measurements were completed.

(c) Operations. The accelerator was manned for 508 hr, of which 450 hr or 89% was running time.

c. Reactor Code Center (M. Butler)

Last Reported: ANL-7460, p. 88 (June 1968).

During the second quarter of 1968 the Center distributed a total of 244 program packages, bringing the number distributed this fiscal year to 848--a sixteen percent increase over fiscal 1967.

Of this total, 131 went to non-U.S. requesters and the cooperating ENEA Computer Programme Library; 717 were disseminated within the United States. Universities received a total of 119, with 571 going to AEC contractors and the nuclear industry, 23 to U.S. computer manufacturers, and 4 to private individuals.

B. Reactor Fuels and Materials Development

1. Fuels and Claddings

a. Research and Development

(i) Behavior of Reactor Materials

(a) Fission-gas Behavior in Uranium-Plutonium Carbide Fuels (L. C. Michels)

Last Reported: ANL-7460, p. 92 (June 1968).

Postirradiation metallography of sections from specimens irradiated in Capsules ANL-56-12 and ANL-56-13 has been completed. The observed variation of density in certain of the fuel specimens was found to be due to an inhomogeneous plutonium distribution: a larger proportion of plutonium was present in the top portion of the fuel columns.

Specimens MV-2 and MV-6 irradiated in the MTR in Capsule ANL-56-11 are being examined. These specimens, which contained vibratorily compacted mixtures of UC and PuC powders, achieved burnups of 9.3 and 9.8 a/o, respectively. Melting and failure occurred in Specimen MV-2 near the top of the fuel column. The Nb-1 w/o Zr cladding of Specimen MV-6 has a large longitudinal crack. Fuel swelling, presumably, was responsible for initiation of the crack. Postirradiation neutron radiographs also indicated probable inhomogeneous plutonium distribution in both specimens.

Postirradiation examination of Specimens MV-3 and MV-5, irradiated in the MTR in Capsule ANL-56-8, is continuing. The postirradiation neutron radiograph of Specimen MV-3 indicates a larger proportion of plutonium in the top portion of the fuel column.

The Hitachi 200-kV electron microscope (HU-200E) is being installed and will be used to study fission-gas behavior.

(b) Thermal Properties of LMFBR Fuel

(1) Thermal Conductivity of Irradiated Reactor Materials

Not reported previously.

Fuel Materials. The effective thermal diffusivity and conductivity of irradiated U-Pu-Zr alloys will be measured in the near

future. The flash method,* which allows rapid, accurate, and relatively easy measurements in the "hot cell," will be used to examine U-Pu-Zr alloys that have exhibited three annular zones after irradiation (see Progress Report for December 1967, ANL-7403, p. 3). The heat flow in the radial direction across these zones will be investigated. Since the transverse sections normally cut from the fuel pins would not enable the measurement of thermal conductivity across the three zones, the flash method was adopted.

A feasibility study has been completed with a flash technique** that appeared to be suitable for these alloys. A cylindrical sample of Armco iron (2 in. long and 0.144 in. in diameter) was supported at the ends by Lavite holders. After a 0.060-in.-dia hole was drilled through the top half of the holder and half of the sample, a Chromel-Alumel thermocouple was inserted through this hole and forced against the inside surface of the sample. This assembly was then centered within the helical coil of a GE FT-524 xenon flash tube. The sample was kept short so that the heat pulse from the flash tube was uniformly distributed over the outer surface of the sample.

The initial results of the study with Armco iron indicated that correct values of thermal diffusivity can be measured with this system; the results were within 10% of published values. However, the temperature curve of the inner surface was distorted, probably because good thermocouple contact could not be maintained in the sample. We plan to drill a hole ultrasonically of no more than 0.020 in. halfway through the U-Pu-Zr specimen, so that very little of the center zone is cut away. Miniature sheathed thermocouples (of 0.008 in. OD) will be used, and greater pressure can be applied to the sheath to insure optimum thermal contact. Since the simple analysis[†] used to calculate the thermal diffusivity may not give correct values when applied to a cylindrical sample, this approach will be verified.

(2) Cladding Materials. The existing "hot cell" facility will be used to measure thermal diffusivity and conductivity of irradiated stainless steel. Preparation of both irradiated and unirradiated specimens is in progress.

*Parker, W. J., et al., Flash Method of Determining Thermal Diffusivity, Heat Capacity and Thermal Conductivity, J. Appl. Phys. 32(9), 1679-1684 (1960).

**Morrison, B. H., et al., A Parametric Study of Flash Thermal Diffusivity Measurements, LA-DC-8300 (CONF 661032), Los Alamos Scientific Laboratory (1965).

[†]Parker, W. J., op. cit.

(ii) Chemistry of Irradiated Fuel Materials (C. E. Crouthamel)(a) Development of Analytical Facilities, Microstructure Sampling Techniques and Analytical Procedures for Analysis of Irradiated Fuels

Last Reported: ANL-7460, p. 96 (June 1968).

The final phases of construction of a shielded fuel-evaluation facility are proceeding. Installation of the in-cell portion of the pneumatic transfer tube system is complete except for the valve actuators, which are not yet available. Electrical service connections have been made and sealed. This completes major construction within the cell. The access window has been installed, and leak checking of the facility has started. Electrical interlocks have been installed on the waste transfer lock and the turbo-molecular pump.

A "rabbit" to be used in the pneumatic transfer tube is being designed and built; tests of a diverter switch await completion of testing of the "rabbit." The route of the pneumatic tube between the shielded facility and the analytical facility has been laid out. Construction is scheduled to begin shortly.

(b) Postirradiation Studies of Reactor Fuel

Last Reported: ANL-7445, pp. 107-108 (April 1968).

(1) Electron Microprobe Analysis of Irradiated

UO₂-PuO₂. A further study has been made with an unshielded electron probe of the radial distributions of fission products and fuel impurities in a transverse cross section of a UO₂-20 w/o PuO₂ fuel pin (SOV-6) irradiated in EBR-II to an estimated burnup of 2.7 a/o. Electron microprobe studies of this cross section that were reported previously dealt with the interactions between the fuel and cladding (see Progress Report for November 1967, ANL-7399, p. 121) and with the radial distribution of uranium and plutonium as well as the fission products that had segregated as micron-sized metallic inclusions (see Progress Report for February 1968, ANL-7427, p. 102).

X-ray-spectral profiles were obtained for the wavelength region covering the major lines of all the fission products at various locations across the radius of the fuel. The areas analyzed were approximately 10 microns square and were free of metallic inclusions. The fission products that were readily detected in the oxide matrix were zirconium, cesium, barium, lanthanum, cerium, praseodymium, neodymium, and samarium. There was no detectable migration or selective relocation of the fission products zirconium, lanthanum, cerium, praseodymium, neodymium, and samarium. The distribution of the neodymium as a function of radial

position is shown in Fig. II.B.1. The ordinate of Fig. II.B.1 is expressed in three ways, namely, counts/sec neodymium, w/o neodymium, and equivalent burnup of total uranium and plutonium based on theoretical fission yields. The distribution of zirconium, lanthanum, cerium, praseodymium, and samarium as a function of radial position were similar to that of neodymium. The observed concentrations of these elements agree quite well with the expected values based on theoretical fission yields.

Fig. II.B.1. Radial Distribution of Neodymium in $\text{UO}_2\text{-20 w/o PuO}_2$ (2.7 a/o burnup)

No cesium was detected in the columnar grain-growth region. Large amounts were found in the cooler (unsintered) region of the fuel, primarily adjacent to the cladding. Smaller amounts were found in the cooler portion of the equiaxed grain-growth region.

Barium was found in the $\text{UO}_2\text{-PuO}_2$ matrix, as well as in a separate gray phase found distributed throughout the equiaxed region. The composition of the gray phase varied with radial position. This phase was the result of an Al_2O_3 impurity introduced into the fuel during a ball-milling operation. Molten Al_2O_3 which was in the columnar-grain region was not soluble in the $\text{UO}_2\text{-PuO}_2$ matrix and migrated to the cooler equiaxed region where it solidified. The molten alumina acted as a getter for barium oxide and other oxides. The gray phase in the equiaxed region near the columnar grains was primarily $\text{BaO} \cdot 6\text{Al}_2\text{O}_3$ (m.p. 1900°C) with trace amounts of chromium oxide (less than one percent). In the cooler portion of the equiaxed region near the unsintered zone, the phase was either aluminum oxide or aluminum oxide with small amounts of barium oxide, chromium oxide, and, sometimes, strontium and cesium oxides. The amount of barium in the $\text{UO}_2\text{-PuO}_2$ matrix varied with radial positions; it was barely detectable in the columnar-grain region, but was found in larger amounts in the cooler region.

An iron impurity, which was found primarily in a band at the juncture of the equiaxed grains and the unsintered region, was present both as a metal phase (molybdenum-iron eutectic) and as an oxide phase (ternary iron-plutonium-uranium oxide).

There was no marked difference between the distribution of the oxide fission products in the mixed-oxide fuel irradiated in the fast flux of EBR-II and that in enriched UO_2 pins (see Progress Report for May 1967, ANL-7342, p. 88) irradiated in a thermal flux (MTR) except for the $\text{BaO} \cdot 6\text{Al}_2\text{O}_3$ phase in the mixed-oxide fuel pin.

(iii) Thermodynamics of Fuel Materials

(a) Total Vapor Pressures and Carbon Potentials in the Ternary System U-C-Pu (P. E. Blackburn and M. Tetenbaum)

Last Reported: ANL-7445, p. 108 (April 1968).

The study of the vaporization behavior of the uranium-carbon system is continuing. Measurements of the total pressure of uranium-bearing species and carbon activity are made as functions of temperature and UC_x composition, using the transpiration method with hydrogen-methane mixtures as carrier gases.

Total uranium pressures measured over single-phase uranium monocarbide at 2255 and 2355°K were compared with the mass-spectrometric measurements of Storms.* Such a comparison is shown in Fig. II.B.2 for 2255°K, where good agreement is evident. Carbon activities are compared with those of Storms in Fig. II.B.3; the agreement is fair.

Fig. II.B.2. Total Pressure of Uranium-bearing Species as a Function of Composition (2255°K).

Fig. II.B.3. Carbon Activity as a Function of Composition and Temperature for the Uranium-Carbon System.

*Storms, E. K., Thermodynamics of Nuclear Materials, IAEA, Vienna (1966), Vol. I, pp. 315, 330; The Refractory Carbides, Academic Press (1967), p. 211.

The results indicate trends that are consistent with expectations for single-phase uranium monocarbide, namely, a sharp increase in total uranium pressure and a sharp decrease in carbon activity as the composition of the condensed phase approaches the metal-rich phase boundary.

Extension of the measurements to wider composition and temperature ranges is planned.

2. Coolants, Moderators, and Control Materials--Fundamentals of Corrosion in Liquid Metals

a. Corrosion Inhibition by Dissolved Getters in Liquid-sodium Environment (C. A. Youngdahl and W. McFall)

Last Reported: ANL-7460, p. 96 (June 1968).

Tests with solutions of gettered sodium in capsule containers at 650°C have continued. Combinations of compatible materials are being sought for continued study of the solutions with minimum interference from the container material. During previous work, commercially pure nickel containers were found to interact substantially with magnesium in solution in sodium at 650°C (see Progress Report for April 1968, ANL-7445, pp. 103-106). The results of more recent experiments are summarized in Table II.B.1. The capsules were rotated slowly during the experiments and were water quenched from 650°C at the conclusion of the tests. Metallographic study of the insert samples and chemical analyses of the sodium solutions are in progress. The behavior of stainless steel in contact with sodium-magnesium solutions appears promising.

TABLE II.B.1. Weight Change of Solid-metal Samples after Exposure to Sodium Solutions at 650°C for 147 hr

Capsule Number	Material (Capsule and Insert)	Nominal Initial Concentration of Additive (weight ppm)	Insert Weight Change ($\mu\text{g}/\text{cm}^2$)
DG20	Nickel	2210 Ca	+201
DG21	Nickel	7550 Ba	-187
DG22	304 SS	3140 Mg	+22

3. Radiation Damage on Structural Materials--Research and Development-- In-Reactor Creep Studies

a. In-pile Creep (J. A. Tesk)

Last Reported: ANL-7457, pp. 107-108 (May 1968).

An instrumented creep machine was encapsulated to test the auxiliary heating control of the specimen and the capability of the linear variable differential transformer (LVDT) to measure strain. Thermocouples were attached to the specimen and the LVDT. The system behaved satisfactorily in the laboratory.

A creep machine was subsequently assembled with a specimen of documented Type 304 stainless steel and with the LVDT calibrated at temperatures up to 500°C. The apparatus was encapsulated and inserted in hole VT-10 of CP-5. All systems worked satisfactorily. After the reactor went to power, difficulties were experienced in keeping the capsule temperature low and in controlling specimen temperature with variations of reactor power. Difficulties were also encountered in the measurement of strain rate, except for a brief initial success (i.e., at a temperature of 650°C and under a stress of 20,000 psi; creep rate was 8×10^{-4} /hr). Attempts to improve temperature control resulted in a loss of all auxiliary heater power. The capsule was removed from the reactor and is being examined to determine means of improving the performance.

b. Void Formation and Growth (S. D. Harkness)

Last Reported: ANL-7460, pp. 98-100 (June 1968).

(i) A Model of Void Formation. Void formation in structural materials subjected to fast-neutron bombardment has become a problem of major concern to fast-reactor development. A model of void formation was made in an effort to understand the phenomenon and, as a result, to control void formation. Both nucleation and growth of voids are considered.

(a) Nucleation. The vacancy cluster is the most stable configuration for point defects in all face-centered cubic materials when only a few defects are present. As more vacancies condense on the cluster, one of several secondary defects may become the more stable structure. These secondary defects include the void, stacking-fault tetrahedra, Frank loop, and perfect loop. The energy of formation of each of these defects is roughly independent of temperature, but will be a function of defect size, modulus, and the stacking fault and surface energy of the material.

For materials irradiated in a fast flux, however, two important differences exist. First, the damage event almost instantaneously

creates a vacancy-rich core that can easily agglomerate into a cluster, which, in effect, allows the material to start with a cluster size of about 9 Å in diameter. Second, fast-neutron irradiation generates gas homogeneously in the material. The gas then may diffuse into the small clusters and lower the energy of void formation. Clarebrough *et al.** observed void formation in copper and silver only when each was quenched from a hydrogen-bearing atmosphere. Thus, when the void was not the secondary defect of lowest energy, gas generated by irradiation could eventually stabilize void formation.

Voids due to irradiation were observed in three face-centered cubic materials: nickel, aluminum, and austenitic stainless steel. These voids occurred after fast-neutron irradiation in the same homologous temperature range ($1/4$ to $1/2$ T_{MP}). The curves of the energy of formation for secondary defects in the three materials were calculated by using the equations given by Sigler and Kuhlmann-Wilsdorf.** For nickel, the void is stable up to very large sizes, which implies that no stabilization by gas

atoms should be needed. This was verified by Mastel and Brimhall,[†] who found voids in nickel after fluences of only 10^{19} n/cm². For aluminum, the defect energies of formation for the void and the loop are quite similar, which again implies that little gas should be needed for void formation. Experimental verification^{††} was obtained from examinations of aluminum irradiated in the HFIR reactor to fluences less than 10^{22} n/cm².

In austenitic stainless steels, the form of lowest energy up to quite large sizes is the tetrahedron, and the void is the least stable. Figure II.B.4 shows the defect energies for this material. A higher gas concentration would be required to stabilize void formation, which is apparently verified experimentally since voids are not observed in austenitic stainless steel until fluences in EBR-II of about 10^{22} n/cm² are attained.

*Clarebrough, L. M., Humble, P., and Loretto, M. H., Voids in Quenched Copper, Silver, and Gold, Acta Met. **15**, 1007 (1967).

Sigler, J. A., and Kuhlmann-Wilsdorf, D., Calculations on the Mechanical Energy of Vacancy Condensation Loops, Stacking Fault Tetrahedra, and Voids, Phys. stat. sol. **21, 545 (1967).

†Mastel, B., and Brimhall, J. L., BNWL-SA-1746.

††Cunningham, J. E., Severe Radiation Damage to Aluminum Alloys, ORNL-TM-2138 (March 1968).

(b) Void Growth. To examine the importance of possible influences on void growth, the changes in free energy accompanying a cavity expansion from radius X to $X + dX$ were considered. The increment in free energy for this change is given by

$$dF = -4\pi X^2 dX \left(RT \frac{\rho}{M} \ln \frac{N_V}{N_{V_0}} + P_G + \sigma_A \right) + 8\pi X dX, \quad (1)$$

and, at equilibrium,

$$RT \frac{\rho}{M} \ln \frac{N_V}{N_{V_0}} + P_G + \sigma_A = \frac{2\gamma}{X}, \quad (2)$$

where

R = universal gas constant = 8.3×10^7 ergs/cm²-sec,

T = absolute temperature,

ρ = density of material,

M = molecular weight of material,

N_V = number of vacancies present/cm³ = $C_V + N_{V_0}$,

C_V = concentration of vacancies in the lattice due to neutron irradiation,

N_{V_0} = vacancies in thermal equilibrium,

P_G = internal gas pressure in the cavity,

σ_A = $1/3$ applied tensile stress for isotropic material,

and

γ = surface energy ≈ 1000 ergs/cm².

This equation shows a basic difference between the two types of cavities discussed in the literature. Voids are the result of a dominant-vacancy supersaturation term, whereas bubbles exist when the internal gas-pressure term is the most important.

A calculation of the magnitude of the vacancy supersaturation term for austenitic stainless steel under EBR-II conditions will emphasize the importance of the term. The steady-state vacancy concentration due to the neutron irradiation must first be calculated, which involves solution of the equation

$$\frac{\partial c}{\partial t} = D_V \nabla^2 C_V + \phi \sigma_D - \beta D_V D_I C_V C_I.$$

If recombination is treated as a perturbation, the following solution for the vacancy concentration is found:

$$C_V = \left[\phi \frac{\sigma_D}{15D_V} (r^2) - \frac{\beta(\phi\sigma_D)^2}{72D_V} (0.045 r^6) \right] N_0, \quad (3)$$

where

C_V = concentration of vacancies,

C_I = concentration of interstitials,

ϕ = instantaneous flux $\cong 3 \times 10^{15}$ n/cm² for EBR-II,

σ_D = displacement cross section $\cong 3 \times 10^{-22}$,

D_V = vacancy diffusion coefficient for γ iron $\cong e^{-32,250/RT}$,

D_I = interstitial diffusion coefficient,

r = sink spacing,

β = recombination parameter $= \pi^2/R_0^2 D_V$,

R_0 = recombination distance,

and

N_0 = number of atoms/cm³.

To determine the steady-state vacancy concentration at any particular flux and temperature, the mean free path between defects is assumed to be the sink distance.

Thus, for irradiation at 800°K at a flux of 3×10^{15} n/cm² and with a sink distance of 2×10^{-5} cm, the vacancy concentration is

$$C_V = (1.2 \times 10^{-8}/\text{cm}^3) N_0.$$

The concentration of vacancies in thermal equilibrium at 800°K is

$$N_{V_0} = N_0 e^{-32,000/RT} = 2.1 \times 10^{-9} N_0;$$

therefore, the vacancy supersaturation term at 800°K is equal to 230,000 psi, or much larger than any possible applied stress or internal gas pressure. This result then implies that only vacancy supersaturation should be important in cavity growth at 500°C.

As the temperature is increased, however, the vacancy supersaturation term goes rapidly to zero. For example, at 1000°K,

$$D_V = 10^{-7} \text{ cm}^2/\text{sec},$$

$$C_V = 2.4 \times 10^{-10} N_0,$$

and

$$N_{V_0} = 10^{-7} N_0;$$

therefore,

$$\ln (N_V/N_{V_0}) \cong \ln 1 = 0.$$

At lower temperatures the vacancy supersaturation term becomes very high under EBR-II conditions. The mobility of the vacancy, however, is severely decreased with $D_V = 10^{-18}$ at 400°K . The structure found experimentally at low temperatures is the black dot, which is frequently observed in irradiated samples. The excess vacancies could well be associated with this defect.

4. Techniques of Fabrication and Testing--Research and Development

a. Nondestructive Testing--Research and Development

(i) Development of Nondestructive Testing Techniques (C. J. Renken)

Last Reported: ANL-7460, pp. 100-101 (June 1968).

(a) Neutron Techniques. Several measurements have been made with the new neutron-image-intensifier tube (of 15-cm diameter). Brightness measurements indicate that the actual improvement in brightness is by a factor of 5.68 (the original estimate was an improvement greater by a factor of three). The brightness of the output phosphor of the new tube measured 444 ft-L in a thermal-neutron intensity of $2 \times 10^7 \text{ n/cm}^2\text{-sec}$ and with an applied accelerating voltage of 22.4 kV. Under similar conditions, the previous experimental tube (of the same diameter) measured 78 ft-L. The tubes normally operate at an anode voltage of 27 kV. The extremely high-brightness yield of the tubes forces operation at reduced anode voltage with the neutron intensity available in the Juggernaut-reactor beam. The anode voltage is adjusted to keep the brightness of the output phosphor below 100 ft-L.

Measurements of relative neutron-gamma response also confirmed expectations of a substantial improvement for the new tube. The improvement was expected because the window of the new tube is lead glass, which has a high neutron transmission (~90%) and attenuates some lower-energy gamma radiation. The tubes formerly had a boron glass window (which matched the expansion coefficient of the metal flange used on X-ray-image-intensifier tubes) that transmitted only about 50% of the incident neutrons.

Measurements with cobalt-60 gamma radiation indicated that the new tube requires a neutron intensity of only 4.75×10^4 n/cm²-sec to yield a brightness of output phosphor equal to that produced by an intensity of cobalt-60 gamma radiation of 1 R/min. Under similar conditions, the previous tube required a neutron intensity of 1.37×10^5 n/cm²-sec. The neutron-gamma ratio of the new tube is improved by a factor of about three. In a neutron intensity of 2×10^7 n/cm²-sec, which is available at the Juggernaut reactor, the cobalt-60 gamma radiation that would be required to yield an equivalent tube brightness with the new tube is approximately 25,000 R/hr.

Initial measurements of the motion response of the new tube indicate that the tube follows high-contrast motion as fast as 5 m/min with relatively negligible image blurr. This result is similar to that obtained with the other image-intensifier tubes tested, and represents a limitation of motion capability of the vidicon television system used to view remotely the phosphor screen of the neutron-image-intensifier tube (rather than the image intensifier tube).

(b) Passive Ultrasonic Test Techniques. Experiments to improve joining techniques for high-temperature acoustic sensors are being conducted.

One problem under investigation concerns the tendency of the Nicro brazing alloy to remove part or all of the platinum-palladium-gold electrode on the transducer element during the brazing process. Means of reducing this effect, such as preheat temperature and time, brazing temperature, holding time, and conditioning of the element surface, are being evaluated.

(c) Development of an Electrodynamic Transducer. The electrodynamic transducer was checked with the pancake coil of optimum design (see Progress Report for May 1968, ANL-7457, p. 109) to determine the present frequency range.

An aluminum test piece of low attenuation was used, and tuning circuits appropriate for each test frequency were inserted near the coil to insure maximum response. The system, which operated in through-transmission, detected the first received pulse of up to 35.0 MHz through the 12-in.-long aluminum rod. Response of the system was too poor for accurate amplitude measurements.

5. Engineering Properties of Reactor Materials--Research and Development--High Temperature Mechanical Properties of Ceramic Fuels

a. High Temperature Mechanical Properties of Fuel Oxides (R. J. Beals)

Last Reported: ANL-7427, pp. 108-109 (February 1968).

(i) Maximum Fiber Stress. The maximum stress of fibers of sintered uranium dioxide was determined from room temperature to 1250°C in vacuo by cross-bending under four-point loading. Dry-pressed urania specimens were sintered at 1700°C for 4 hr in dry hydrogen to a density of approximately 93% of theoretical.

Preliminary data on the maximum fiber stress, or modulus of rupture, are shown in Fig. II.B.5 with each datum point representing a minimum of ten tests. Also shown are the deflections of the test bars under four-point loading. The average maximum fiber stress of the sintered urania at 25°C was 807.0 kg/cm², with an average deflection of 0.0096 cm over the 3.175-cm span. The average stress increased to 830.5 kg/cm² at 500°C, to 897.3 kg/cm² at 1000°C, and to 1079.7 kg/cm² at 1250°C. Deflections increased to 0.0131 cm at 500°C, to 0.0138 cm at 1000°C, and to 0.0169 cm at 1250°C. All tests were carried out at a crossheat movement of 0.005 cm/min.

Fig. II.B.5

Effect of Temperature on the Maximum Fiber Stress of Uranium Dioxide

Tests are continuing at 1375, 1500, and 1700°C in vacuo at the slow strain rate. Additional tests will be carried out at faster strain rates and with more dense urania. Additional variables to be investigated include stoichiometry, grain size, and pore placement. These data are preliminary to the determination of the maximum fiber stress of mixed urania-plutonia specimens.

(ii) Dynamic Flow. Tungsten tooling to measure the dynamic flow of ceramic materials under compressive loads has been designed and is scheduled for immediate delivery. Right-cylindrical specimens of uranium dioxide will be evaluated preliminary to the determination of compressive stresses of dynamic flow in urania-plutonia.

C. Engineering Development

1. Research and Development

a. Development of Master-Slave Manipulator Systems (D. P. Mingesz and R. C. Goertz)

Last Reported: ANL-7460, pp. 104-105 (June 1968).

(i) Mark-E4A Electric Master-Slave Manipulator. The servo drive unit which consists of four servomotors, a tachometer, a synchro, a solenoid operated brake, a power gear train, and a synchro gear train, has now been irradiated to a total exposure of 2.5×10^9 R.

At 1×10^9 R, only one item on the servo drive unit had changed enough to call it a failure; the brake friction increased to 1.5 times its value before irradiation. This change would not prevent further use of the slave arm. It would increase the risk of damage to the arm if a high external force were applied to the arm while the brakes were set.

At 2.5×10^9 R, three items had changed enough to call them failures. The brake friction increased to twice its value before irradiation. The viscous friction, as measured at the output shaft of the servo drive unit, increased to about five times its original value. The resistance of the rotor winding to ground of the synchro dropped from many megohms to about 700 ohms. Again, none of these failures would completely prevent further use of the arm.

Cyclic testing of the complete manipulator has brought to light several desirable modifications:

With a 30-lb weight in the slave tongs and the master arm driven in a circular motion, the fixed fields would automatically switch to a higher voltage during a fair part of the cycle. This fixed-field switching caused a little too much jerking of the arms, and a new set of voltages and gains have been chosen. The switching is much smoother now.

The master motors were getting too hot. One developed shorted turns and was replaced. A small fan was added to each servo drive unit on the master to prevent further problems in this respect.

As a temporary shortcut, friction couplings were originally used for some of the drive components between the servo drive units and the arm members of the master. Some worked loose. All have now been modified to have positive drives (not depending on friction).

Two fabricated parts, one on the upper arm and one on the lower arm of the master, are a little too springy and show signs of

failing. We plan to replace these with castings. They will be designed to be more rigid and stronger; the increase in inertia will be moderate.

(ii) Study of Remote Handling for LMFBR Facilities. We visited Westinghouse, Waltz Mill Site, to discuss remote handling problems on LMFBR work. They limited the discussion to their plans on the 1000-MWe Sodium-cooled FBR Plant. Many of the principal handling problems were covered, and there was considerable interchange of ideas.

b. Instrumentation and Control

(i) Boiling Detector (T. T. Anderson)

Last Reported: ANL-7457, p. 113 (May 1968).

(a) Acoustic Method

(1) High-temperature Detector Development. The bonds between the crystal and stainless steel components have been unsatisfactory, apparently because a nickel oxide film forms at the interface and because of the solubility of the fired film in several high-temperature brazing alloys. Alternative procedures are being attempted to resolve the problem.

(2) Sound Measurements in EBR-II. Acoustical noise arising from flow and boiling in a dummy EBR-II subassembly has been measured in two experimental facilities: (1) a water-filled pressure-drop test loop and (2) a large water-filled tank, 1.67 m in diameter by 3.36 m high.

In the test loop, cavitation noise was emitted during 80- and 120-gpm flows at 8 and 30 psig. The spectrum of this noise, in the range from 0 to 50 kHz, was 3 to 7 times the spectrum of noise emitted by nonflow boiling from the calrod in the center of an 81-fuel-pin EBR-II-type test subassembly. Because cavitation has the same broad-band frequency spectrum as boiling, it was difficult to detect boiling during flow. However, boiling in the presence of cavitation was detected as a slight increase in the average noise level of the acoustic signals. The average was obtained by rectifying and integrating the amplified acoustic signal from the hydrophone.

In the development of an acoustic-boiling detection system, the acoustic properties of boiling were simulated by driving a high-frequency (600 kHz) hydrophone with random pulses to produce band-limited white noise. Two tests were performed. In the first, one hydrophone was driven by the random pulses; another hydrophone in the tank received the pressure signal. In the second test, the transmitting

hydrophone was placed inside the EBR-II subassembly and returned to the same relative position in the tank. With this simulation technique, no change was observed in the received signal, which implies that an EBR-II subassembly does not appreciably attenuate the acoustic emission from steady boiling.

The random-pulse technique also has been used (1) to determine acoustic reflections of high-frequency pulses, and thereby to have an independent measure of tank reverberation on steady-state sound levels and (2) to identify the occurrence of high-frequency pulses during steady boiling.

c. Heat Transfer and Fluid Flow

(i) Heat Transfer in Liquid Metal Cooled Reactor Channels
(R. P. Stein and J. V. Tokar)

Last Reported: ANL-7460, pp. 107-108 (June 1968).

Design work continues on the stainless steel flow loop that will circulate sodium through electron-bombardment-heated (EBH) test sections. The loop is being designed for experiments with operating conditions typical of LMFBR designs; the heat flux will be increased and/or the coolant velocity will be decreased until a heat-transfer-limited condition occurs. The heat-transfer limit probably will result from incipient boiling, which will cause flow instability. Because the pressure-flow characteristics of the loop can strongly influence the characteristics of the flow instability, the loop is being designed so that its pressures and flows resemble those expected in LMFBRs. The possible occurrence of a critical heat flux without gross flow instabilities is also being anticipated.

In addition to providing a more fundamental understanding of heat transfer in liquid-metal-cooled reactor channels, the information from the experiments may be directly usable in specifying the equivalent of a "burnout heat flux" for various LMFBR core designs.

(ii) Heat Transfer in Liquid Metal Heat Exchangers
(R. P. Stein)

Last Reported: ANL-7460, pp. 105-107 (June 1968).

(a) Liquid-metal-heated Steam Generators. The piping for Freon and mercury in the experimental loop has been completed, and leakchecking of the system has begun. The instruments for reading and recording data have been installed in racks adjacent to the loop. When all leakchecks have been completed, thermocouples will be attached to the test section and to other points on the loop.

Operating procedures for the loop are being drafted for use during initial test runs.

d. Engineering Mechanics

(i) Structural Dynamics Studies--Parallel-flow-induced Vibrations (M. W. Wambsganss, Jr.)

Last Reported: ANL-7460, pp. 108-109 (June 1968).

(a) Transducer Development. Development and evaluation of instrumentation designed to measure fluid pressures and component displacements associated with flow-induced vibration are continuing.

Test results have been obtained from seven miniature pressure transducers mounted longitudinally and circumferentially on a test rod. The transducer signals have been filtered and correlated; time responses and correlated outputs are being studied. Preliminary tests showed good results; results from later tests indicate a malfunction due to high-resistance shorts in cable connectors.

An experiment has been designed to gain further insight into the dynamic behavior of the internally mounted displacement transducers. The test section (containing the test specimen with internally mounted displacement transducers) has been fitted with end caps that will allow filling it with water. The water-filled assembly will be agitated on a shaker to determine component natural frequencies and potential response peaks due to coupling. This information will be used in evaluating the measured transducer outputs.

D. Chemistry and Chemical Separations

1. Aqueous and Volatility Processes--Research and Development--Fluoride Volatility Process

a. Development of Head-end Steps (M. J. Steindler and N. M. Levitz)

Last Reported: ANL-7445, p. 117 (April 1968)

(i) Chemistry of Stainless Steel. An investigation of the reactions of Types 304 and 316 stainless steel with HF-oxygen mixtures has been started to determine whether the fluoride-catalyzed destructive oxidation of stainless steel, which is under consideration as a procedure for decladding fast reactor fuels, is easily controlled and results in powdered products.

Preliminary experiments have been performed with Types 304 and 316 stainless steel tubing (of 3/8-in. diameter) at 500 and 550°C, using a gas mixture containing 40 v/o HF and 60 v/o oxygen, and reaction times between 1/4 and 2 hr. The data obtained showed that Type 304 reacts to a greater extent than does Type 316. The oxide scale formed on Type 316 was much more adherent and less easily powdered than that formed on Type 304. X-ray-diffraction analysis of the scales showed the presence of α -Fe₂O₃, a minor phase which is either γ -Fe₂O₃ or Fe₃O₄, and a very minor phase which is probably nickel. The emission spectrographic analysis of off-gases which had been passed through cold traps at 0 and -78°C did not show that any of the stainless steel constituent elements were volatilized during the reaction. Further study of the fluoride-catalyzed oxidation of Types 304 and 316 stainless steel will include investigation of the effects of irradiation of stainless steel, temperature, and HF concentration on the reaction.

(ii) Mechanical Decladding. Exploratory ball-mill tests have been made to examine the potential of milling for separating fuel from cladding. In three tests, simulated chopped fuel-pin sections (1/2 to 1-in. lengths of 10-mil stainless steel tubing filled with porcelain) were ball-milled on a laboratory scale with balls of 5/32- to 1-in. diameter. Recoveries of porcelain (a stand-in for fuel oxide) were 93, 96.5, and 97%, which are promising for initial tests on a small scale. A process consisting of scoring fuel pins and then splitting and crushing them (ANL-7460, p. 112) is also being evaluated.

b. Purification Procedures and Fission Product Chemistry
(M. J. Steindler)

Last Reported: ANL-7460, pp. 110-111 (June 1968).

(i) Decontamination of PuF₆. In the step of the fluid-bed fluoride volatility process in which nuclear fuel is reacted with fluorine, a mixture of the hexafluorides of uranium and plutonium with volatile fluorides of ruthenium, niobium, molybdenum, technetium, and antimony will be produced and transported away from the fuel residue. One procedure for separating volatile ruthenium fluoride from PuF₆ is by passage of the gas stream through a bed of granulated LiF, which sorbs fission products and a portion of the plutonium as complexes. This is followed by preferential removal of the plutonium by reaction with fluorine. The solid complexes formed by the reaction of PuF₆ with LiF are LiF·PuF₄ and 4LiF·PuF₄.

Initial experiments were carried out to determine the lower stability limit for 4LiF·PuF₄ (i.e., the minimum temperature at which 4LiF·PuF₄ is stable with respect to dissociation to LiF·PuF₄ and LiF). The procedure consisted of heating samples of a mixture of LiF·PuF₄ and LiF,

each at a different temperature between 270 and 400°C, for several hours and then analyzing the solid product for the presence of $4\text{LiF}\cdot\text{PuF}_4$. Results of experiments indicate that $4\text{LiF}\cdot\text{PuF}_4$ is formed at 400°C and possibly at lower temperatures. Additional data will be obtained by holding samples of $4\text{LiF}\cdot\text{PuF}_4$ at constant temperatures for several hours and analyzing them for the presence of $\text{LiF}\cdot\text{PuF}_4$.

Other work is in progress to determine whether exposure of solid LiF and $\text{LiF}\cdot\text{PuF}_4$ to gaseous RuF_5 results in the fixation of ruthenium by the solid phase. Solids which have been exposed to RuF_5 are awaiting analysis for ruthenium.

c. Engineering Design, Analysis, and Evaluation (N. M. Levitz)

Last Reported: ANL-7460, p. 112 (June 1968).

Conceptual design studies of a fluoride volatility processing plant for fast breeder reactor fuels (one metric ton of actinides per day) are in progress. The composition of the projected fuel feed was presented earlier (see ANL-7457, p. 121).

As part of the development of an engineering flowsheet for a fluoride volatility processing plant, several types of cold traps were considered. An externally cooled "egg-crate" trap* was selected for study. In this trap the process gas is cooled as it passes through an arrangement of fins which are bonded to the shell of the trap. The cooling medium passes through tubes attached to the outside of the shell. An advantage of this design is that it minimizes the chance of mixing the coolant with the process gas, since only by the rupture of both the coolant tube and the shell could such mixing occur. A preliminary estimate has been made of the number of traps and the sizing that would be needed for the conceptual plant for processing fast breeder reactor fuels.

2. Closed Cycle Processes--Research and Development--Compact Pyrochemical Processes

a. Laboratory Process Development (I. Johnson)

Last Reported: ANL-7460, p. 114 (June 1968).

Investigations of the mechanisms through which plutonium, uranium, and fission product elements distribute in the reduction step of the pyrochemical salt-transport process flowsheet are being made. The form of the elements as they distribute is also being studied and related to criticality considerations. For example, plutonium as an intermetallic compound would be less of a criticality problem than metallic plutonium.

*Smiley, S. H., Brater, D. C., Littlefield, C. C., and Pashley, J. H., Quantitative Recovery of Uranium Hexafluoride from a Process Gas Stream, Ind. Eng. Chem. 51(2), 191-196 (1959).

A small-scale reduction of UO_2 powder (~40 g) was made to establish a frame of reference for subsequent experiments with plutonium. The UO_2 was reduced through use of a CaCl_2 -20 m/o CaF_2 salt and a Cu-33 w/o Mg-5 w/o Ca alloy at 800°C in a tantalum crucible. The melt was agitated for 5 hr and allowed to settle for 60 hr before being cooled to room temperature. The long mixing and settling times insured complete equilibration and are not representative of actual processing times. The crucible was sectioned vertically, and samples were prepared for metallographic examination.

Metallographic examination revealed that the system had stratified into three layers: metallic uranium at the bottom, a Cu-Mg alloy in the middle, and a salt layer at the top. X-ray examination of the samples established that the Cu-Mg alloy consisted of primary Cu_2Mg and a eutectic of Cu_2Mg and Mg_2Cu . In addition, a phase tentatively identified as CaCu_5 formed prior to the formation of Cu_2Mg . The metallic uranium layer contained small spherical particles that were tentatively identified as CaO . The salt phase was free of both alloy and uranium.

Future experiments will determine the distribution of plutonium among the phases present after the reduction step of the process.

b. Supporting Chemical Studies (I. Johnson)

Last Reported: ANL-7460, p. 113 (June 1968).

Experiments are being conducted to study the physical and chemical aspects of the reduction reaction of the salt-transport process. In these experiments UO_2 pellets are being exposed to various molten salts and alloys under a variety of conditions and then examined metallographically.

The following tentative conclusions have been drawn:

- (1) The reduction occurs at the surface of the UO_2 pellet or particle.
- (2) The products of the reduction (uranium and CaO or MgO) do not disperse spontaneously into either the salt or the alloy phase.
- (3) The UO_2 must be wet by the reductant for appreciable reaction to occur.
- (4) The metallic uranium layer formed around the UO_2 when magnesium is used as the reductant appears to be more dense than the layer formed when calcium is used as the reductant.

The CaCl_2 -rich region of the CaCl_2 - CaF_2 - CaO phase diagram was studied in connection with investigations of various salt compositions

for the reduction step. The liquidus surface in the area of interest was determined, and the diagram is presented in Fig. II.D.1. The compositions of two incongruently melting compounds, $\text{CaCl}_2 \cdot \text{CaF}_2$ and $\text{CaO} \cdot 2\text{CaCl}_2$, are shown on the diagram.

Fig. II.D.1. CaCl_2 -rich Region of CaCl_2 - CaF_2 - CaO System

c. Process Engineering Studies (R. D. Pierce)

Last Reported: ANL-7457, pp. 125-126 (May 1968).

Work has continued on an engineering flowsheet for pyrochemical processing of liquid metal fast breeder reactor fuels. The principal effort has been concentrated on the head-end operations, and zinc dissolution has been selected as the decladding method. Fuel subassemblies will be trimmed of their bottom hardware and immersed in molten zinc with the wrapper tube or shroud still in place. The upper hardware will remain above the molten zinc, and will be removed after the fuel and blanket portions of the subassembly have dissolved. The presence of sodium on or in the fuel elements is not expected to create a problem. The manner in which fission product gases will be released from unvented fuel pins is being investigated. Plutonium accountability will be achieved by sampling the Cu-Mg reduction alloy after the reduction step.

d. Experimental Flowsheet Investigations (R. D. Pierce)

Last Reported: ANL-7457, pp. 126-127 (May 1968).

Work is continuing on the Plutonium Salt Transport Facility (see Progress Report for May 1968, ANL-7457, pp. 126-127). All of the gloveboxes have been installed. The design of the helium-purification system required for the helium gloveboxes has been completed and is being checked. Drafting work for new filter chambers and gas distributors for the gloveboxes is nearing completion. Shop work is underway on the purifier for the helium system. A hygrometer and an oxygen analyzer that will be used to monitor the water and oxygen content of the helium atmosphere have been received.

Six 13-in.-ID furnace vessels are being fabricated by Union Carbide* from Type 304 stainless steel. Six water-cooled adapter flanges have been built to mount the reaction vessels below the floor of the processing glovebox; six holes will be cut in the floor of the box to accommodate the vessels and flanges. Top flanges that will incorporate movable transfer lines are being designed for the processing vessels. Four 20-kW furnaces for heating four of the vessels are scheduled for shipment in August.

Two large shear-formed tungsten crucibles (of $11\frac{1}{8}$ -in. OD by 20 in. high) and two smaller crucibles (of 6-in. OD by 14 in. high) have been received from Metallwerk Plansee.** Three additional crucibles (of $5\frac{3}{4}$ -in. OD by 14 in. high) have been shipped to complete the order. Motors that will be used to drive the agitators in the reaction vessels have been received, and Mo-30 w/o W and Croloy 16-1 tubing for the transfer lines is on hand.

3. General Chemistry and Chemical Engineering--Research and Development

a. Identification of Mechanism(s) of Carbon Transport in Sodium Systems (F. A. Cafasso)

Last Reported: ANL-7445, p. 122 (April 1968).

Efforts to characterize the carbon-bearing species that migrate between structural materials immersed in liquid sodium are continuing. For this purpose, the sodium partaking in the transport of carbon from a ^{14}C -labeled ferritic Fe-C alloy was analyzed. Radioactive carbon was used to increase the sensitivity of the analysis as well as to distinguish between

*Union Carbide, Nuclear Division, Paducah, Kentucky.

**Metallwerk Plansee, Innsbruck, Austria.

transported and contaminant species of carbon. Identification and assay of the carbon species was made by dissolving the sodium samples in water and determining the gaseous reaction products* by radiogas chromatography.

In one such experiment, sodium and a tab of the ^{14}C -labeled Fe-C alloy were sealed in a small tantalum capsule and heated at $\sim 650^\circ\text{C}$ for 120 hr. Analysis of the entire sample of sodium at the end of the experiment showed that the major ^{14}C -containing off-gas from dissolution was acetylene. The specific activity of the carbon in this acetylene was close to that of the carbon in the starting Fe-C alloy, indicating that little exchange with contaminant carbon had occurred. Because iron carbide does not yield acetylene on hydrolysis, these results suggest that Na_2C_2 (which does) is the most likely precursor of the acetylene observed, and that it is involved in the transport of carbon from the Fe-C alloy. In 1965, Flint** postulated on thermodynamic arguments that formation of Na_2C_2 is involved in the transfer of carbon through liquid sodium. These results appear to substantiate his suggestion.

In other experiments, a different experimental arrangement was used so as to sample the liquid sodium more than once. Sodium, either with or without a Type 304 stainless steel tab immersed in it, was held at $\sim 650^\circ\text{C}$ in a crucible made of the ^{14}C -labeled Fe-C alloy. The crucible was heated in a helium atmosphere within a stainless steel furnace tube. Liquid sodium was pipetted from the "open" crucible via a sampling lock on the furnace tube. Dissolution of the samples yielded both acetylene and carbon dioxide as major ^{14}C -labeled off-gases. The specific activities of carbon were lower than in the starting Fe-C alloy by factors of ~ 5 and ~ 100 , respectively. Thus, a considerable amount of contaminant carbon species had exchanged with the transported ^{14}C -bearing species. The contamination may have resulted from oxygen introduced during sampling, which subsequently interacted with components of the system. Although formation of Na_2C_2 may be the same as in the capsule experiment, the problem of contamination appears to be a serious drawback to using this kind of experimental arrangement for the study of carbon transport.

b. Total Vapor Pressure and Oxygen Potentials in the Ternary U-Pu-O System (P. E. Blackburn and J. W. Reishus)

Last Reported: ANL-7445, pp. 123-124 (April 1968).

The transpiration apparatus is being incorporated into a glovebox facility for studies of the Pu-O and U-Pu-O systems. The purpose of these investigations is to determine both the oxygen potentials and the total pressures of plutonium-bearing or uranium-plutonium-bearing species over samples of known composition.

*Preliminary experiments had shown that $> 90\%$ of the ^{14}C -labeled carbon in sodium appeared in the off-gas from aqueous dissolution; the remaining carbon was distributed between the solution and a small amount of insoluble product.

**Mott, B. W., private communication.

Tests of various components of the transpiration system have been conducted recently. The stability and reproducibility of the electro-balance were tested under experimental conditions by operating in vacuum, flowing He-6% H₂, and static He-6% H₂ atmospheres while the furnace was cycled between 500 and 2300°C. During these tests, the noise level of the balance signal remained at the background level of 4 μ g or less. However, apparent weight decreases of 56, 150, and 175 μ g, respectively, were observed in the three environments when the temperature of the furnace was raised to 2300°C. Additional investigations will be conducted to determine the cause and nature of these apparent weight changes.

Further assembling of the gas-flow system and connecting of the glovebox to the helium-purification system are awaiting delivery of various components. It is anticipated that these components will be available in the near future so that full-scale tests of the transpiration apparatus can be conducted. For these tests, the U₄O₉-U₃O₈ system will be investigated prior to the initiation of the plutonium studies.

c. Vapor Species Partial Pressures in the Ternary U-Pu-O System (P. E. Blackburn and J. E. Battles)

Last Reported: ANL-7445, pp. 124-127 (April 1968).

Mass-spectrometric studies of the volatilization behavior of the uranium-plutonium-oxygen system continue to determine (1) the apparent composition of the vapor phase in equilibrium with the condensed phase(s), (2) the partial pressures of the vapor species as a function of temperature, and (3) the thermodynamic properties of the vapor species. The first phase of this study is being devoted to the plutonium-oxygen system, since mass-spectrometric data for this system is not available.

The initial work relating to the apparent congruently vaporizing plutonia (~PuO_{1.83}) has been completed, and the study of the binary condensed phase Pu₂O₃ + PuO_{1.61} and the hypostoichiometric plutonia (PuO_{1.61}-PuO_{1.83}) initiated.

The two-phase material (Pu₂O₃ + PuO_{1.61}) was prepared by mixing the appropriate amounts of PuH₃(s) and PuO₂(s), and heating slowly to about 550°C in vacuum. After heating at 550°C for 12-16 hr, the material was then heated slowly to 1500-1600°C and maintained at this temperature for about 4 hr, or until all reaction was completed, as determined by mass-spectrometric observations. Chemical analysis showed that the material had an O/Pu atom ratio of 1.57.

Effusion measurements were carried out in tungsten effusion cells. The rate of mass effusion was determined at 2132°K for the (Pu₂O₃ + PuO_{1.61}) solid mixture and at 2163°K for single-phase PuO_{1.67}. The vapor pressure was calculated by applying the Knudsen equation for molecular

flow, considering the vapor phase to be entirely PuO(g) . The O/Pu ratio was determined for the sample residues at the conclusion of the measurements. Because of the large samples (>1 g) and the small amount of material vaporized (<10 mg), the composition of the samples remained essentially constant during the measurements. The results of several measurements on each type of sample are given in Table II.D.1; the observed ion-intensity ratios $I_{\text{PuO}^+}/I_{\text{Pu}^+}$ and $I_{\text{PuO}^+}/I_{\text{PuO}_2^+}$, and an ionizing electron energy of 13 eV, are also included in the table.

TABLE II.D.1. Results of Mass Effusion Measurements for $(\text{Pu}_2\text{O}_3 + \text{PuO}_{1.61})$ Solid Mixture at 2132°K and for $\text{PuO}_{1.67}$ at 2163°K

Run No.	O/Pu Atom Ratio	Ratio $\frac{I_{\text{PuO}^+}}{I_{\text{Pu}^+}}$	Ratio $\frac{I_{\text{PuO}^+}}{I_{\text{PuO}_2^+}}$	Pressure ($\text{atm} \times 10^6$)
<u>$\text{Pu}_2\text{O}_3 + \text{PuO}_{1.61}$</u>				
PW-2A	1.58	10	-	7.391
PW-3B	1.59	11	~300	6.379
PW-3D	1.57	13	~300	5.878
PW-3E	1.57	13	~300	5.809
				Avg 6.364
<u>$\text{PuO}_{1.67}$</u>				
PW-1A	1.68	28	50	2.890
PW-1B	1.67	30	52	2.738
				Avg 2.814

In calculating the vapor pressure, the orifice area was corrected for thermal expansion, and the Clausing correction factor* was applied for molecular flow through short tubes. The data for the ion-intensity ratios for the vapor species are tentative, pending additional measurements of the temperature dependency.

A comparison of the vapor pressure data from this study with those of Ackerman, Faircloth, and Rand** and of Ohse and Ciani† is shown in Table II.D.2 for hypostoichiometric plutonia (PuO_{2-x}) and the two-phase ($\text{Pu}_2\text{O}_3 + \text{PuO}_{1.61}$) solid mixture. (For additional data on hypostoichiometric plutonia see ANL-7445.) The vapor pressures reported in Table II.D.2 were determined by considering the vapor phase to be entirely $\text{PuO}_2(\text{g})$ over $\text{PuO}_{2-x}(\text{s})$ and $\text{PuO}(\text{g})$ over the $(\text{Pu}_2\text{O}_3 + \text{PuO}_{1.61})$ solid mixture. The agreement among the three sets of data is fair.

*Clausing, P., Ann. Physik 12, 961 (1932); see also, Dushman, S., Scientific Foundations of Vacuum Technique, John Wiley and Sons, Inc., N. Y. (1962), pp. 80-117.

**Ackerman, R. J., Faircloth, R. L., and Rand, M. H., J. Phys. Chem. 70, 3698 (1966).

†Ohse, R. W., and Ciani, V., paper presented at Third International Symposium on High Temperature Technology, Aslomar, California (September 17-20, 1967).

TABLE II.D.2. Comparison of Vapor Pressure Data in the Plutonium-Oxygen System at 2132°K

	O/Pu Atom Ratio	Vapor Pressure (atm)
<u>Pu₂O₃ + PuO_{1.61}</u>		
This study	1.58	6.364×10^{-6}
Ackermann <i>et al.</i> *	1.52	1.13×10^{-5}
Ohse and Ciani**	-	1.55×10^{-5}
<u>PuO_{2-x}</u>		
This study	1.83	6.76×10^{-7}
Ackermann <i>et al.</i> *	1.83	5.970×10^{-7}
Ohse and Ciani**	1.84	1.65×10^{-6}

*J. Phys. Chem. 70, 3698 (1966).

**Paper presented at Third International Symposium on High Temperature Technology, Aslomar, California (September 17-20, 1967).

PUBLICATIONS

Laboratory Investigations in Support of Fluid-bed Fluoride Volatility Processes. Part XVI. The Fluorination of UO₂-PuO₂-fission-product Oxide Pellets with Fluorine in a 2-inch-diameter Fluid-bed Reactor

L. J. Anastasia, P. G. Alfredson,* M. J. Steindler, G. W. Redding, J. G. Riha, and M. Haas

ANL-7372 (December 1967)

Helium Pumping with a Sputter-Ion Pump

P. M. Danielson

J. Vacuum Sci. Technol. 5(3), 89 (1968)

Dynamics and Control of a Batch Reactor

L. B. Koppel

Ind. Eng. Chem. Process. Design Develop. 7(3), 416-421 (July 1968)

Thermal Conductivity and Heat Capacity of the Monophosphide and Monosulfide of Plutonium

J. B. Moser and O. L. Kruger

J. Am. Ceram. Soc. 51, 369-372 (July 1968)

Ultrasonic Buffer-Rod Technique for the High-Temperature Measurement of the Elastic Moduli of Short Specimens

Allen Sather

J. Acoust. Soc. Am. 43, 1291 (June 1968)

Comparative Physics Analysis of Fast Reactors as a Function of Refinement in Multigroup Cross Section Preparation

B. J. Toppel and H. H. Hummel

Proc. Intern. Conf. on the Utilization of Research Reactors and Reactor Mathematics and Computation, Mexico City, May 2-4, 1967. Centro Nuclear de Mexico, 1968, Vol. 2, p. 611

*Australian AEC.

III. NUCLEAR SAFETY

A. Reactor Kinetics--Other Reactor Kinetics

1. Research and Development

a. Reactor Control and Stability

(i) Reactor Stability Design Criteria for Spatially Dependent Systems (C. Hsu)

Last Reported: ANL-7457, p. 136 (May 1968).

The development of methods for analyzing the stability of nonlinear reactor systems has continued. The method derived using spectrum analysis has been extended using Lyapunov's method, which can eliminate the spectrum computation but usually yields a more restrictive stability condition. Applications of these methods to practical LMFBR systems will be compared.

(ii) Transfer Function Techniques to Measure Large Fast Reactor Stability (L. Habegger)

Last Reported: ANL-7457, p. 137 (May 1968).

The computer program that uses Galerkin's method to approximate eigenvalues in linear, space-time-dependent systems has been used in trial cases with simple system models. The resulting approximate eigenvalues agreed satisfactorily with known exact eigenvalues.

The Galerkin method will be applied to more realistic reactor systems to demonstrate its value.

b. Coolant Dynamics

(i) Critical Flow (H. K. Fauske)

Last Reported: ANL-7460, pp. 122-123 (June 1968).

(a) Sodium Tests. Additional data on flashing of liquid sodium in the cylindrical-divergent test section have been obtained for stagnation temperatures up to 1500°F and for inlet superficial velocities up to 12 ft/sec. Sodium-vapor qualities range from 0 to 4%; corresponding two-phase adiabatic pressure drops range from 0.2 to 5 psi. These will be compared to predicted values using correlations developed for steam-water pressure drops.

Simultaneous measurements of temperatures and pressures in the flashing zone indicate that only slight deviations from thermal equilibrium occur in the high-void-fraction range. Data are being obtained in the low-void-fraction range (0-50%).

(b) Sonic Velocity and Pressure-wave Propagation. Measurements have been made of the propagation velocity of compression and

Fig. III.A.1. Propagation Velocity of Compression and Rarefaction Waves in Steam-Water Bubble Flow at Saturation Temperature of $265 \pm 10^\circ\text{F}$

rarefaction pressure waves in two-phase steam-water bubble flows. As illustrated by Fig. III.A.1, which shows the propagation velocity of the front of the pressure pulse, there is little difference between the velocity of compression and rarefaction waves within the variation of the experimental data. It has been suggested* that rarefaction waves may propagate at a velocity characteristic of thermodynamic equilibrium. However, it is seen that the fronts of both waves travel at a velocity more nearly that of the homogeneous frozen speed, given by

$$a_{TP}^2 = \left\{ \frac{\alpha \rho_{TP}}{\rho_g a_g^2} + \frac{(1-\alpha) \rho_{TP}}{\rho_l a_l^2} \right\}^{-1} \quad (1)$$

This is in agreement with previous compression-wave data of Karplus** and Semenov and Kosterin.† The increased deviation of the data from Eq. (1) at higher void fractions can probably be related to strong flow-regime effects on the propagation velocity (see Progress Report for February 1968, ANL-7427, p. 141). These flow-regime effects, as well as a wider range of system pressures in one-component systems, will be studied.

(ii) Coolant Dynamics (R. M. Singer and R. E. Holtz)

Last Reported: ANL-7460, p. 123 (June 1968).

(a) Sodium Superheat Experiments. The boiler with the sandblasted interior surface is being operated, and data are being reduced and analyzed. A boiler with the as-received surface is being prepared for testing.

*Davies, A. L., The Speed of Sound in Mixtures of Water and Steam, AEEW-M452 (Oct 1965).

**Karplus, H. B., Propagation of Pressure Waves in a Mixture of Water and Steam, ARF 4132-12 (Jan 1961).

†Semenov, N. I., and Kosterin, S. I., Results of Studying the Speed of Sound in Moving Gas-Liquid Systems, Teplotemerika 11(6), 59 (1964).

A sodium pool boiler has been placed in the core of the Juggernaut reactor to measure incipient superheats in a radiation environment. Preliminary results indicate that radiation intensities at powers up to 200 kW do not significantly affect the incipient superheat; a final conclusion cannot be made until a statistically significant amount of data becomes available.

(b) Coolant-expulsion Experiments. Instrumentation to measure the transient void fraction and vapor-growth rate in sodium in the heated section is being installed and calibrated.

(c) Transient Test Experiments. Fabrication and assembly of the Transient Test Loop are continuing. The tube-flow-type electron-bombardment heater (EBH), which will serve as the primary heater, and the thermal-radiation shields to be used around the EBH have been fabricated. The vacuum pumping station has been checked and installed. Development and fabrication of instrumentation are continuing.

c. Core Structural Safety (C. K. Youngdahl)

Last Reported: ANL-7460, pp. 123-126 (June 1968).

(i) Coolant-expulsion Coupling to Cylindrical Channels. Studies pertaining to the influence of the response of the parallel-plate-assembly structure on the pressure pulse and coolant expulsion during a power transient have been reported (see Progress Reports for February 1967, ANL-7308, pp. 84-85 and July 1967, ANL-7357, pp. 143-144). Those analyses are being extended to a structure in the form of a cylindrical shell. Two cases are considered: one considers the variation of shell-wall displacement and pressure in the axial direction, whereas the other neglects it.

To compare the results, calculations were made for a cylindrical shell of the same cross-sectional area as that of the parallel-plate structure, taking into account the variation of dependent variables in the axial direction. The shell radius was 0.1784 in.; wall thickness was 0.01 in.; all other parameters were kept the same as those used for the parallel-plate structure. The peak pressure developed in the cylindrical shell was 67.128 psi; in an equivalent rigid bounding structure, the peak pressure would be 67.179 psi. Thus, as was expected, the influence of the bounding structure in the form of a cylindrical shell on the peak pressure is negligible for the case considered. The more severe case of no variation of dependent variables in the axial direction was also considered. The effect was found to be negligible for these dimensions. A general digital-computer program has been developed for both cases.

d. Fuel Meltdown Studies with TREAT

(i) Meltdown Studies with Ceramic Fuels (C. E. Dickerman)

Last Reported: ANL-7457, pp. 134-135 (May 1968).

(a) Ceramic Fuel Experiments. The first TREAT experiment planned for the Mark-II (high-pressure) sodium loop is a high-specific-energy "checkout" experiment. In this experiment, a UO_2 fuel pin consisting of an 11-in.-long stack of 0.250-in.-dia oxide pellets inside a Type 304 stainless steel jacket of 0.290-in. diameter will be run, inside a ring of six dummy pins. The oxide fuel is 13% enriched. A calibration run has been made in TREAT to obtain an experimental value for the ratio of sample power to reactor power. Radiochemical measurements are being made on fully enriched uranium foils attached to the cladding and on pellets from the pin. In order to mock up the nuclear aspects of the experiment, the calibration sample was run in an actual Mark-II loop. The loop is being shipped back to the Illinois Site for outfitting after the prototype loop outfitting is finished. For protection in shipping to and from Idaho, the loop is loaded into a "special form" container (see Progress Report for June 1968, ANL-7460, p. 121) built specifically for shipping Mark-II loops and transparent capsules.

(ii) Development of Experimental Methods (C. E. Dickerman)

Last Reported: ANL-7460, p. 121 (June 1968).

(a) Experimental Apparatus

(1) Mark-II Integral Sodium TREAT Loop Construction.

The prototype Mark-II integral sodium loop is nearly completely outfitted, and the facility for proof-testing of the system has been prepared. This consists of a temporary structure built of existing 8-ton shielding blocks, located on the west side of Building D206. The blocks are 3 ft square in cross section and $7\frac{1}{2}$ ft long, and are erected to form a massive enclosure which is $10\frac{1}{2}$ ft high, with a 3 x 3-ft central open shaft to contain the 9 ft 5 in.-long integral loop in vertical suspension. Control console, power supplies, pressure source, and auxiliaries will be located inside the building. The electrical and pressure instrumentation as well as the coolant lines will be carried through a port in the west wall to the loop within the test cell. In the interest of safety to personnel and vehicles, proof-tests of the sodium-filled loops at elevated temperatures will be carried out on weekends, with the D206-208 parking lot evacuated.

The Inconel X-750 flow tubes for the Mark-II loop Annular Linear Induction Pumps (ALIP) failed to pass final inspection. Since the flaws consisted of interior weld undercuts, the tubes could not be reworked. However, it was possible to salvage the finished central tube

sections. New flanged sections were made which are of greater length than designed to compensate for the loss due to cutoff. Flanged and central sections were joined by electron-beam welding, instead of inert-arc welding. The refabricated flow tubes required a full solution anneal, followed by full age-hardening. This incurred a delay, since the finished flow tubes had to be sent off-site to a commercial heat-treater because of the unavailability of the vacuum furnace in the shops; however, a number of the hardened tubes have been returned and are ready for the final assembly of the pumps.

(b) Plutonium Facility for Mixed Oxide Experiments. The exhaust portion of the once-through nitrogen-atmosphere system for the plutonium glovebox facility being prepared in Building 301 has progressed through the design stage, and contracts have been let for procurement and construction. The system is basically a demand-type tandem duplex (redundant) arrangement. The design incorporates two motor-exhausters in parallel in each of two branches. The first (or primary) branch (with a single exhaustor operating) will have the capacity to carry the normal range of gas flows contemplated for the glovebox--with the greater capacity of a secondary branch immediately available, through a pilot-operated relief valve, to maintain the glovebox pressure negative. To assure continuous operation, automatic transfer switches have been provided so that, within each branch, the associated standby exhaustor will be activated in case of malfunction of a particular operating unit, and power for both will be supplied from the building's normal-emergency bus.

To allow for reasonable reloading and recycle of the Mark-II sodium loops and transparent capsules with low-level fuel packages, the "double-door" transfer (wall penetration) technique has been utilized in developing a suitable end plate for the glovebox. Fabrication of the end plate is complete, and it is in final inspection.

e. Materials Behavior and Energy Transfer

(i) Interactions of Fuel, Cladding, and Coolant (R. O. Ivins)

Last Reported: ANL-7445, pp. 137-138 (April 1968).

Studies relating to the possible physical or chemical interaction between reactor materials and the reactor coolant are continuing. An accident in a nuclear reactor could cause large quantities of molten fuel materials to come into contact with the reactor coolant (either sodium or water), resulting in explosive pressure generation, presumably because of the breakup of the molten materials into small droplets with a greatly increased surface area. Currently, experiments are being conducted on the dynamics of fragmentation of low-melting metals dropped into water.

Most of the experiments done in the last six months, in which molten tin was dropped into water, have been processed using a

computer program which calculates the two-dimensional surface area of each particle and the particle-size distribution of all of the particles (greater than about 0.1-mm diameter) in a particular run.

The results were evaluated by averaging the particle-size distribution data from five runs performed under similar conditions and plotting the averages on log-probability graph paper as cumulative percent surface area. In all cases, the particle-size distributions were bimodal in nature, as indicated by the typical data shown in Fig. III.A.2. Frequency distribution plots were then made on the data from all of the experiments. Again, the double distributions indicated by the breaks in the curves of the log-probability plots were confirmed in all cases. Fig. III.A.3 is the frequency distribution plot of the data presented in Fig. III.A.1; the double distribution is apparent, with the median particle diameters occurring at about 0.21 and 6.7 mm.

Fig. III.A.2
Particle-size Distribution of
Tin Fragmented in Water

Fig. III.A.3
Plot of Percent Frequency
vs. Log Diameter for Tin
Fragmented in Water

The data presented in the figures have been evaluated further. Figure III.A.3 indicates that the median diameter of one part of the double distribution occurs at 0.21 mm and that this distribution extends to about 3.3 mm. The data from this first distribution (neglecting the largest four size fractions), were used to recalculate the particle-size distribution. This is also plotted in Fig. III.A.2. It can be seen that these points fall on a straight line, which indicates compliance of the data to a log-probability distribution and further strengthens the case for multiple distribution.

The remaining experiments have been evaluated in a similar manner. A few general conclusions can be made at this time: (1) in all cases, a double distribution is observed; (2) increasing the drop height (entrance velocity) considerably increases both the number of particles produced and the total fragmented surface area density; (3) at metal temperatures near the critical temperature of water (373°C), the greatest amount of fragmentation occurs, as evidenced by large values of the fragmented surface area density.

Time-exposure photographs, made under illumination of a stroboscopic light, have recently been taken of molten tin falling into water. These photographs provide a clue to the cause of the multiple distributions noted above. They show that an initial breakup of the molten tin occurs at or near the surface of the water, probably as a result of the dynamic forces acting on the molten metal at the time of impact. In addition, at some distance below the surface of the water (about 2 cm when the initial temperature of the tin is 330°C), a second and much more violent fragmentation takes place in which many of the fragments are expelled sideways with considerable force. This second fragmentation would account for the very large number of small particles observed in all of the experiments.

A computer program is being written to fit a bimodal probability distribution to the data. This will simplify the presentation of the data and should provide a more detailed description of fragmentation.

(ii) Violently Sprayed Sodium-Air Reaction (R. O. Ivins)

Last Reported: ANL-7445, pp. 138-140 (April 1968).

The study of the sodium-air reaction has been undertaken to provide knowledge about the reaction rate of molten sodium, in spray form, with air and air depleted in oxygen. This information will be of value in safety analyses of sodium-cooled nuclear reactors.

A mathematical model was developed to predict the burning rate and burning temperature of a single spherical sodium particle moving through air or air depleted in oxygen. The model was based on the

assumption that the reaction rate is controlled by the diffusion of oxygen to a vapor-phase combustion zone surrounding the particle (see Progress Report for January 1968, ANL-7419, pp. 149-152).*

In order to maintain a consistent model, the substitution which expresses the equilibrium constant in terms of the model variables, Eqs. (9) and (10) in ANL-7419, have been altered as follows:

$$K_P = \frac{1}{(P_{Na}^2 P_{O_2}^{1/2})_B}; \quad (9)$$

$$\begin{aligned} 1/K_P &= P_t^{5/2} (Y_{Na}^2 Y_{O_2}^{1/2})_B \\ &= 16 (Y_{O_2,B} P_t)^{5/2}. \end{aligned} \quad (10)$$

This conversion assumes that the sodium is present in a stoichiometric proportion to the oxygen in the combustion zone. Typical results of calculations which incorporate the above modification are shown in Figs. III.A.4 and III.A.5.

Fig. III.A.4. The Effect of Particle Size on the Theoretical Burning Temperature and Burning Rate of Sodium Spheres

Fig. III.A.5: The Effect of Particle Velocity on the Theoretical Burning Temperature and Burning Rate of Sodium Spheres

*One of the equations as originally presented in ANL-7419 is incorrect. Equation (1), which describes the rate of diffusion of oxygen, should read

$$N_{O_2} = \frac{k_y(Y_{O_2,A} - Y_{O_2,B})}{1 - Y_{O_2,B}}. \quad (1)$$

In the next phase of the theoretical analysis, a quasi-steady-state approach and an averaging technique were used in correlating the reaction rates of the individual particles with the burning rate of the spray and the pressure-rise rate of the enclosing volume. The results of these calculations are in good agreement with the experimentally observed variations in the pressure-rise rate with respect to ambient oxygen content, spray velocity, and particle size that were presented in ANL-7445.

A comprehensive discussion of the theoretical and experimental results of the sodium-air reaction studies will be presented in a forthcoming topical report.*

(iii) Segregation in Ceramic Fuels (M. G. Chasanov)

Last Reported: ANL-7445, pp. 141-142 (April 1968).

Work is continuing to determine the extent of migration and segregation in mixed uranium-plutonium fuels under a thermal gradient. Experiments using $(U_{0.8}Pu_{0.2})O_2$ pellets have given some evidence of an increase in plutonia concentration at the hotter end of specimens held in gradients of approximately $1000^\circ C/cm$ for up to 1000 hr (see ANL-7445). Confirmation of this evidence is being sought through further experimentation.

Two additional tests have recently been completed using 1/2-in.-high by 1/2-in.-dia cylindrical pellets. In each experiment the temperature at the sample top was held at about $2350^\circ C$ and that at the sample bottom at about $1150^\circ C$; the heating times were 262 and 432 hr, respectively. The sample heated for the longer time was sectioned and examined ceramographically; a relatively pore-free region was found at the top accompanied by long columnar grains. Both samples currently await electron-microprobe analyses for plutonium and uranium content.

2. TREAT Operations (J. F. Boland)

Last Reported: ANL-7460, pp. 126-127 (June 1968).

a. Reactor Operations. Experiments RP-49-1 and RP-49-2 were each subjected to three transient tests in transparent capsules. Each capsule contained an EBR-II Mark-IA fuel element instrumented with eddy-current transducers for measuring the elongation of the fuel inside the jacket. The photographic film and motion transducer data have been sent to Illinois for analysis.

A 13% enriched UO_2 fuel element was irradiated under steady-state conditions in a mockup of the Mark-II packaged sodium loop capsule.

*Krolikowski, T., Violently Sprayed Sodium-Air Reaction in an Enclosed Volume, ANL-7472.

The purpose of this irradiation was to obtain power calibration data for future experiments on this type of fuel in Mark-II packaged sodium loops.

Neutron radiographs were made of the EBR-II oscillator rod, four groups of EBR-II driver fuel pins, and one BNW capsule from MTR.

b. Development of Automatic Power Level Control System.

Approval to increase the operating temperature of TREAT fuel elements to 600°C was received from the AEC on July 16. The approval will permit proceeding with plans to run transients with greater energy releases than at present. An automatic control system is being designed for TREAT so that the greater energy releases can be used effectively during loss-of-flow or other quasi-steady-state experiments. The system will require the installation of two servo-controlled hydraulic control-rod-drive mechanisms and associated electronics. As a result of the AEC approval, reactivity control requirements have been fixed, and the final specifications for the hydraulic control-rod drives are being prepared for submission to vendors.

PUBLICATIONS

Transient Motion of a Thick Finite Elastic Transversely-Isotropic Shell

G. Cinelli and W. Pilkey*

Proc. Canadian Congr. Applied Mechanics, Laval University, Quebec, May 22-26, 1967. Laval University, Quebec, 1968, Vol. 1, pp. 49-51

Summary and Analysis of TREAT Sodium Loop Experiments on Behavior of Single EBR-II Mark I Pins under Accident Conditions

C. E. Dickerman, R. T. Purviance, L. E. Robinson, W. P. Stephany, and F. L. Willis

Nucl. Eng. Design 7(5), 442-454 (1968)

Incipient Pool Boiling of Sodium

R. E. Holtz and R. M. Singer

AIChE Journal 14(4), 654-656 (July 1968) Letter

Removal of Nitrogen from Argon with Titanium-Metal Sponge

M. L. Kyle, R. D. Pierce, L. F. Coleman, and J. D. Arntzen

Ind. Eng. Chem. Process Design Develop. 7(3), 447-453 (July 1968)

Sodium Boiling Transients in Oxide-Fueled Fast Reactors

D. R. MacFarlane, N. A. McNeal, D. A. Meneley, and C. K. Sanathanan

Nucl. Eng. Design 7(5), 411-426 (1968)

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