ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

December 1967

Robert B. Duffield, Laboratory Director
Stephen Lawroski, Associate Laboratory Director

<u>Division</u>	<u>Director</u>
Chemical Engineering	R. C. Vogel
Idaho	M. Novick
Metallurgy	M. V. Nevitt
Reactor Engineering	L. J. Koch
Reactor Physics	R. Avery
Remote Control	D. P. Mingesz (Acting)

Report coordinated by
C. L. Chernick and A. Glassner

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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:

June 1967	ANL-7349
July 1967	ANL-7357
August 1967	ANL-7371
September 1967	ANL-7382
October 1967	ANL-7391
November 1967	ANL-7399

REACTOR DEVELOPMENT PROGRAM

Highlights of Project Activities for December 1967

EBR-II

Reactor operation in Run 26C amounted to 435 MWdt during December and to 619 MWdt since the start of the run on November 22. During this run, small fission product releases occurred in the primary tank. The first of these, at the beginning of the run, was soon identified with two newly loaded driver fuel subassemblies; replacement of these subassemblies solved the problem. Another release, consisting of a very slow and minor buildup of ^{133}Xe and ^{135}Xe , was still under investigation at the end of December.

Power-coefficient measurements continued to indicate a near-zero coefficient between 15 and 25 MWt. At the end of Run 26C, with control rods at a high bank position, the power coefficient at 25 MWt reached the operating limit of 0.5 lh/MW. The power-reactivity decrement from 0 to 45 MWt continued to be about 40 lh throughout the run.

Scheduled production operations were carried out in the FCF hot and cold lines, but emphasis was placed on experimental and surveillance programs. A computerized system was put into operation for processing FCF production and surveillance data.

ZPR-3

Initial criticality was achieved for Assembly 51, the simple non-split cylindrical core of the FFTF Phase-B critical program in ZPR-3, with 215 kg of $^{239+241}\text{Pu}$. The core geometry was then adjusted to the reference critical geometry, and measurement of the worths of core drawers at the core radial boundary was begun.

ZPPR

The contract for the installation of the ZPPR reactor assembly and associated equipment was signed on December 18 with the Hunter-Saucerman Construction Co., Idaho Falls, Idaho.

All electrical components have been installed and checked out. The bed and tables were installed in the reactor pit and aligned satisfactorily. The support wing was accepted for beneficial occupancy with a number of exceptions, such as completion of the room for the Argonne Fast Source Reactor (AFSR). The major work items remaining in the mound area of the facility include leak testing the backup containment structure, properly installing the seal doors, and reworking the cell crane.

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Assembly of the beam-tube test facility is about 75% complete. Installation of piping from the facility to the general-purpose hydraulic loop is 85% complete.

Mathematical models for analyzing stresses in the beam tubes have been completed by Franklin Institute.

Heat-transfer analysis of the neutron windows indicates that forced flow should be used for cooling the neutron windows; natural-convection cooling would cause problems of scale formation and flow oscillations.

Difficulties with use of the FLASH code for loss-of-coolant calculations have been traced to the code-stored physical-property data for water. An accurate set of data is being generated.

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I. PLUTONIUM UTILIZATION--CIVILIAN--EBWR

A. Research and Development

1. Operational Consultation and Support

a. Materials Surveillance (N. Balai)

Last Reported: ANL-7317, pp. 1-2 (March 1967).

When the fuel was removed from the reactor after completion of the Plutonium Recycle Program, the corrosion rate of the SA-212B plate in the pressure vessel, as measured in one reference area, was several times that predicted on the basis of prior out-of-reactor data. (During the Plutonium Recycle Program, EBWR operated for 4901.76 hr above 325°F, of which 4806.76 hr were at 489°F. The nominal water conditions were: pH = 6.9-7.2; resistivity = $1-2 \times 10^6$ ohm-cm; and O_2 = 0.24 ppm in the reactor water and 28 ppm in the steam.)

To investigate the corrosion situation, four months later optical-micrometer measurements were made, the surfaces were examined visually and with dye penetrant, and deposits were analyzed.

The optical-micrometer measurements were made to determine, semiquantitatively, the corrosion of unclad areas of the carbon steel of the pressure vessel (which had been laid bare by grinding and peeling the cladding during previous investigations).^{*} The single initial postoperational reading, which had been made in a steam-zone area, showed an average metal loss of 0.025 in. The readings made four months later, without any cleaning, showed a corrosion-product buildup of about 0.002 in.; this value was verified by additional readings, made after the area was wiped with tissue and scrubbed lightly with glass wool. Measurements at another steam-zone location showed a corrosion-product buildup of about 0.022 in. Two wall areas in the water zone indicated substantial corrosion-product buildup. This is not unexpected and is confirmed, to an extent, by analyses of the corrosion product.

Visual examination of the surface of the stainless steel cladding gave no positive indication of additional cracking of the cladding.

Various areas of the vessel walls were inspected with dye penetrant and photographed. There were no significant differences from the condition of the surfaces prior to the Plutonium Program runs.

^{*}Balai, N., et al., Inspection, Evaluation, and Operation of the EBWR Reactor Vessel, ANL-7117 (Nov 1965), p. 15 et seq.

A small amount of a vermillion deposit was seen in the water zone at the interfaces of the carbon steel and the stainless steel where the cladding had been stripped off during the previous investigations. Samples of this deposit were analyzed with a spectrograph along with a sample of the more usual deposit from the steam zone for reference. Only iron, nickel, and copper oxides were found in the steam-zone deposit. The water-zone deposit also contained aluminum, manganese, chromium, titanium, and zirconium oxides. This indicates that the steam-zone deposits were almost entirely formed in situ, but that the water-zone deposits include a substantial portion of recirculating corrosion products from other areas of the primary system. Although no unusual elements were detected, the strong indication of titanium is unexpected and has not been explained.

Radioactivity levels in the vessel above the present fill line still limit exposure to about 15 min.

II. LIQUID-METAL FAST BREEDER REACTORS--CIVILIAN

A. Fuel Development--LMFBR

1. Metallic

a. Physical Metallurgy of Metallic Fuel Alloys

(i) Density of Irradiated U-Pu-Zr Alloy (W. F. Murphy)

Last Reported: ANL-7382, p. 73 (Sept 1967).

Density measurements have been made on annular fuel sections from U-Pu-Zr elements ND-30, -35, and -43. These fuel elements were irradiated in the EBR-II to burnups of 4.4 a/o. The sample sections were taken from locations near the midlength of the fuel column. The three annular sections from each element were mechanically separated; generally, the rings broke into several pieces. The results of these determinations are shown in Table II.A.1.

TABLE II.A.1. Densities of U-Pu-Zr Alloy from Annular Sections

Fuel Element	Original Fuel Density (g/cm ³)	Postirradiation Density (g/cm ³)		
		Center Zone	Middle Zone	Outer Zone
ND-30	15.3	8.8-9.3(3)	14.7-15.2(3)	10.5-10.6(3)
ND-35	15.3	8.9-9.8(3)	14.3-14.5(3)	12.9(2)
ND-43	15.8	7.2-7.9(3)	15.7-16.4(3)	11.7-12.5(4)

Note: Numbers in parentheses denote number of determinations on one sample.

The least dense fuel is consistently in the center zone and the most dense fuel in the middle zone. Two density values (16.4 g/cm³) for ND-43 are actually greater than for the original fuel. While the precise density values are in doubt, the trends are clearly established. These results are corroborated by metallographic observations that indicate considerably less porosity in the middle zone. However, it was not possible to ascertain differences in porosity between the center and outer zones. Burnup analyses of these annular sections will be made.

b. Fuel Element Performance (W. F. Murphy)

Last Reported: ANL-7399, pp. 1-5 (Nov 1967).

Two fuel elements of U-15 w/o Pu-10 w/o Zr and U-15 w/o Pu-10 w/o Ti in subassembly XG05, shown in Table II.A.2, are continuing

under irradiation in EBR-II. They have accumulated almost 85% of their target burnup of 7.5 a/o; estimated removal date is Feb 1968.

TABLE II.A.2. LMFBR Metal Fuel Elements in Subassembly XG05

S/A No.	Spec No.	Design Parameters				Operating Conditions			
		Fuel Composition (w/o)	Effective Density (%)	Cladding ^b OD (in.)	Cladding Thickness (in.)	Max (kW/ft)	Max Cladding Temp (°C)	Burnup to Date	
								a/o	fiss/cc x 10 ^{-20a}
XG05	ND-24	U-15 Pu-10 Zr	63.6	0.208	0.015	10.0	535	6.2	14.2
XG05	NC-17	U-15 Pu-10 Ti	66.7	0.204	0.016	10.0	540	6.4	14.8

^aBased on effective density.

^bCladding was V-20 w/o Ti.

Fifteen U-15 w/o Pu-10 w/o Zr fuel elements (Group M-3) were inserted into the EBR-II on Nov. 21, 1967. The primary objectives of the M-3 experiment are to confirm the excellent performance shown by an earlier group in subassembly XA07 and to extend experience to 10 a/o burnup, with interim inspections at 5 and 7.5 a/o. The fuel elements are identified in Table II.A.3. Experimental subassembly XO28 containing these elements is located in core position 4E2. To date, the elements have accumulated about 0.25 a/o burnup. The elements are operating with a nominal maximum cladding temperature of 625°C at a linear power rating of 9.4 kW/ft. The effective density of the fuel is 75% of theoretical.

TABLE II.A.3. LMFBR Metal Fuel Elements (Group M-3) in Subassembly XO28

Fuel Material: U-15 w/o Pu-10 w/o Zr
Fuel Dimensions: 13.5-in. long by 0.144-in. diameter

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

^aCladding is 0.166-in. ID.

A group of U-Pu-Zr fuel elements, designated M-4 (see Sect. II.A.1.c.(i)), has been designed with the objective of investigating fuel performance in EBR-II at a linear power rating of 14 kW/ft, an increase of about 4 kW/ft over that of other EBR-II metal-fuel experiments. The higher

power rating is achieved by increasing the fuel diameter from 0.144 to 0.163 in. The target burnup for these elements is 10%. The fuel has a nominal composition of U-15 w/o Pu-12 w/o Zr with 93% enriched uranium. The fuel columns are 14.2 in. long. Forty injection-cast pins have been made and about 75% of the pins have been assembled into fuel elements. The nominal effective density of the fuel is 75%. Cladding materials are Types 304, 316, and 318 stainless steel, Inconel 625, Hastelloy-X, V-15 w/o Ti-7.5 w/o Cr, and V-15 w/o Cr-5 w/o Ti. The claddings are 0.220-in. OD and 0.188-in. ID.

c. Fabrication Development

Last Reported: ANL-7391, pp. 3-5 (Oct 1967).

(i) Fabrication of Group M-4 Fuel Elements (H. F. Jelinek)

Forty fuel elements (M-4) are being fabricated for irradiation experiments as part of the metallic-fuel development program (see Sect. II.A.1.b). The design of the 40 elements, including 4 elements with a segmented fuel column consisting of 5 fuel pieces of equal length, is illustrated in Fig. II.A.1. The fuel material is a U-15 w/o Pu-12 w/o Zr alloy. It is contained in jacket materials of stainless steel, or nickel- or vanadium-base alloys. A triple indent in the jacket restrains the fuel pin from ratcheting above the sodium column in the jacket. The fabrication and assembly of the M-4 fuel elements is being carried out concurrently with several supporting programs. These include furnace modifications, crucible compatibility studies, bonding studies, and hardware-evaluation tests.

Fig. II.A.1. Design of EBR-II M-4 Fuel Elements (all dimensions in inches)

A total of 47 acceptable pins were fabricated from 5 melts, including 7 pins as spares and samples for chemical analyses, compatibility testing, and metallographic evaluation. The pins were cast in two steps.

A binary alloy of U-14.5 w/o Zr (93.2% enriched) was made by gravity casting. The plutonium was then added and the material was injection cast into precision molds of 0.169-in. ID and 18 in. long.

The cast pins were inspected for surface quality, warp, length, diameter, internal porosity, and density. Acceptable castings were sampled and sheared to the approximate desired length. The ends of the pins were machined for the final sizing operation.

Density measurements to determine composition and homogeneity of cast pins were made by immersion in bromobenzene. Density was determined on fuel length casting, as well as top and bottom end sections. A relationship between density and zirconium composition has been developed and has proven reliable for estimating the zirconium composition in this alloy.

A number of tests of sample elements of the several jacket materials are being conducted. Corrosion tests in sodium at 600°C are being carried out. Each tube used has a triple indent and end plugs welded in place. After corrosion testing, the specimens shall be pressure tested at 600°C. Compression tests to determine the force required to push a 0.165-in.-dia pin by the triple indent will be made at room temperature. Comparison will be made with control samples.

All elements have been loaded and bonded except for six elements to be loaded in V-15 w/o Cr-5 w/o Ti jackets. Unacceptable welds were found in top end closures on two Type 304 stainless steel-clad elements. The end plugs were removed and replaced with acceptable closures.

Sodium-bonding parameters are being examined during the production of these elements. Batches of six fuel elements are being assembled and bonded by various methods to determine the effects on bond quality of the process variations. Results of the evaluation are expected to increase understanding of the bonding process and to aid in the design of equipment that will produce bonded elements of more consistent quality. Elements that fail to respond to the new bonding technique are being reclaimed by impact bonding.

Bonded elements are eddy current inspected at 150°C by the masked-aperture point probe and by the differential encircling coil. Sodium levels are determined by hot radiography at 150°C. Evaluation is by comparison with standards and by means of destructive examination. To date 19 elements have been examined. None were rejected for incorrect sodium level, but two were rejected for defects in the bond. The rejects are being decanned for reloading.

(ii) Alloying Development (H. F. Jelinek)

The fabrication of irradiation specimens containing high-melting alloys of uranium-zirconium and uranium-plutonium-zirconium requires methods for making master alloys. The master alloys will be produced by arc-melting in a glovebox.

Commercially available arc-melting furnaces have not been acceptable for glovebox operation without modifications, as based on (1) accessibility to hearth for loading and unloading, (2) handling characteristics of the electrode, (3) size of the chamber, and (4) location of service connections. During 1967 an arc-melting furnace was designed and fabricated at ANL for glovebox use, that could be used with existing vacuum equipment, feed-throughs, and power supply. The vacuum equipment and feed-throughs exist as part of the D-350 hoodline equipment. The power supply is a 600-Amp, dc welder that is currently the power unit for equipment housed in hoodline PF-15. Casters were attached to the unit to make it movable for servicing both furnaces.

(iii) Controlled-density Fuels (D. Walker)

The formation of gaseous fission products contributes to swelling in nuclear fuels. This condition requires that the nuclear reaction be terminated prior to obtaining the desired burnup in order to prevent cladding failure. The use of controlled-density fuel bodies with interconnected porosity would allow the gases developed during fission to diffuse rapidly to an annulus where the accumulation could be accommodated by design. To be useful, however, this controlled porosity must be stable and the fuel body not subject to densification through additional sintering during operation. Controlled-density metallic bodies are readily available commercially as filter bodies, fuel-cell membranes, acoustical resistances, and gas diffusers. Commercial manufacturers have developed products that do not continue to sinter when used at a temperature that would normally cause sintering to occur.

Work has been started to develop controlled-density fuel bodies from U-2 w/o Zr alloy powder. It is intended that the technology developed will be applicable to the uranium-plutonium-zirconium metallic fuel in the form of the metallic sponge from pyrometallurgical refining and vacuum-volatilization.

Attempts were made to obtain either uranium or U-2 w/o Zr powder from commercial sources in accordance with the following simple specifications:

Particle Size: -200 +325 mesh (74-44 μ size)

Purity: Equivalent to commercial grade uranium.
Oxygen not to exceed 0.6 w/o.

General Conditions: Material should not be pyrophoric to the extent that exposure to air immediately results in ignition. Do not use the direct hydride-decomposition method unless the resultant powders are passed through a plasma torch or otherwise atomized.

Uranium powder produced by reduction of UO_2 was offered by Nuclear Fuel Services, Irwin, Tennessee, as powder with 3- to 8- μ particle size and with an oxygen content <0.6 w/o. The powder does not ignite upon exposure to air, but burns vigorously if exposed to an open flame. This material, as well as material produced by the Chemical Engineering Division, was used as input to the controlled-density work. Approximately 1 kg of U-2 w/o Zr was produced in Chemical Engineering by the hydriding process. This material, as well as a small amount of hydride-produced uranium powder made in the Metallurgy Division, has been studied. Cylindrical samples, 0.40-in. in diameter and approximately 0.18 in. long, were pressed at 16,000 psi in dies in a dry box containing a helium atmosphere. No lubricant or binder was added to the powder. The green bodies were then transferred to a sintering furnace external to the dry box thereby exposing the sample to the air. As an additional variable, some samples were pressed in a steel die and then isostatically compacted at 25,000 psi.

The maximum sintered density attained with the hydride-produced U-2 w/o Zr powder is 13.3 g/cm^3 (73% of theoretical density). Pressing and sintering parameters for this sample were:

Steel Die Compaction:	16,000 psi
Isostatic Compaction:	25,000 psi
Sintering Temperature:	950°C
Atmosphere:	Vacuum (5×10^{-5} mm Hg)
Time:	4 hr

No work has been done with the commercial uranium powder from Nuclear Fuel Services other than to determine that a binder will be necessary to provide adequate green strength in the pressed body.

2. Oxide

a. Fuel Element Performance (F. L. Brown and W. N. Beck)

(i) Mixed-oxide Fuel Performance

Last Reported: ANL-7391, pp. 5-6 (Oct 1967).

Efforts continued in an attempt to locate the capsule in experimental subassembly XO11 that discharged fission-product gas into the EBR-II primary coolant. Two suspect capsules containing failed elements (HOV-4 and HOV-10) were returned to ANL, Illinois, for examination. Evacuation tests for fission-gas leakage, vacuum bubble tests, and weight-change tests were performed. Results for Capsule HOB-10 were all negative and, therefore, it was eliminated as a suspect. Capsule HOV-4, however, decreased in weight equivalent to the weight of bond sodium originally in the capsule, and released a small amount of ^{85}Kr fission product gas in an evacuated chamber. The exact location of the possible leak was not revealed by vacuum bubble testing, nor by helium mass-spectrometric leak detection.

3. Carbide--Fabrication and Evaluation

a. Fuel Element Performance (F. L. Brown)

(i) Uranium-Plutonium Carbide

Last Reported: ANL-7391, pp. 6-7 (Oct 1967).

The status of uranium-plutonium carbide fuel elements undergoing irradiation in EBR-II is shown in Table II.A.4.

TABLE II.A.4. Status of Mixed-carbide Fuel Irradiations in Progress in EBR-II

S/A No.	Specimen Number	Design Parameters						Operating Conditions			
		Fuel Composition (w/o)	Fuel Form ^a	Effective Density (%)	Cladding Composition (w/o)	Cladding OD (in.)	Cladding Thickness (in.)	Max Cladding Temp (°C)	Max (kW/ft)	Burnup to Date a/o	fiss/cc x 10 ^{-20b}
XG05	SMV-2	UC-20 PuC	VIPAC	83.8	304 SS	0.296	0.021	24.0	675	6.9	19.2
XG05	HMV-5	UC-20 PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	26.6	685	7.2	18.9
XG05	NMV-11	(U _{0.8} Pu _{0.2})C	VIPAC	82.9	Nb-1 Zr	0.281	0.012	25.5	635	7.2	19.7
XO08	NMP-2	(U _{0.8} Pu _{0.2})C	PELLET	81.6	Nb-1 Zr	0.284	0.013	17.4	545	4.5	12.0
XO08	NMV-4	UC-20 PuC	VIPAC	80.0	Nb-1 Zr	0.283	0.013	27.0	645	6.7	17.7
XO08	NMV-7	UC-20 PuC	VIPAC	80.0	Nb-1 Zr	0.283	0.013	26.0	630	6.6	17.4
XO08	NMV-12	(U _{0.8} Pu _{0.2})C	VIPAC	85.5	Nb-1 Zr	0.284	0.014	26.7	645	6.6	18.7
XO08	HMV-1	UC-20 PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	25.2	660	6.1	16.0
XO08	HMV-4	UC-20 PuC	VIPAC	80.0	Hastelloy-X	0.295	0.016	27.2	690	6.5	17.2
XO08	HWMP-1	(U _{0.8} Pu _{0.2})C	PELLET	81.4	Hastelloy-X + W	0.296	0.020	16.9	565	4.4	11.9
XO08	HWMV-1	(U _{0.8} Pu _{0.2})C	VIPAC	82.9	Hastelloy-X + W	0.295	0.018	26.4	685	6.6	18.0
XO15	NMP-1	(U _{0.8} Pu _{0.2})C	PELLET	81.6	Nb-1 Zr	0.284	0.013	16.5	535	1.8	4.7
XO15	NMV-3	UC-20 PuC	VIPAC	80.0	Nb-1 Zr	0.284	0.013	26.0	635	2.6	7.0
XO15	HMV-2	UC-20 PuC	VIPAC	80.0	Hastelloy-X	0.295	0.015	27.0	685	2.6	6.8
XO15	TVMV-1	(U _{0.8} Pu _{0.2})C	VIPAC	83.7	V-20 Ti	0.298	0.021	23.4	640	2.4	6.6

^aVIPAC = Vibratorily compacted into cladding; PELLET = Pressed and sintered pellets.

^bBased on effective density.

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

(i) Type 304 Stainless Steel

Last Reported: ANL-7349, p. 79 (June 1967).

Further examination of two specimens containing Type 304 stainless steel in contact with stoichiometric and hyperstoichiometric (U,Pu)C for 4000 hr at 800°C* has shown evidence of carbon diffusion from the (U,Pu)C into the steel. Evidence for the existence of a higher carbon content in the stainless steel at the (U,Pu)C interface was provided by the absence of sigma phase as well as by heavier carbide precipitation along slip lines within approximately 300 μ of the interface.

A difference between the thermal-expansion coefficients of the compatibility-test specimens and of the frame in which they were held resulted in slight deformation of the stainless steel specimens at the beginning of their heat treatment. Consequently, carbides precipitated not only at grain boundaries, but also along distorted slip lines. The increase in carbide precipitation near the interface evidently decreased the chromium content of the stainless steel below that required to form the sigma phase under the heat-treating conditions. No sigma-phase formation was observed in the stainless steel after 1000 hr at 800°C, but microprobe results indicated a slight carbon concentration gradient in the stainless steel near the interface.** The composition of these specimens (18.6 w/o Cr, 8.8 w/o Ni, 0.69 w/o Si, 0.05 w/o C) lies in the gamma-plus-sigma phase field very near the gamma-plus-sigma/gamma phase boundary at 650°C.†

(ii) Incoloy 800

Last Reported: ANL-7391, pp. 7-8 (Oct 1967).

Exposure of Incoloy 800 at 800°C for 1000 hr with a (U_{0.8}Pu_{0.2}) carbide composition containing 6.75 w/o equivalent carbon has shown that this alloy is less compatible with (U,Pu)₂C₃ than it is with (U,Pu)C. Although precipitation surrounding and within the Incoloy grains occurs to approximately the same depth (125 μ average, 160 μ maximum) as in the specimen containing single-phase (U_{0.8}Pu_{0.2})C,†† the number and size of the precipitates is greater in the specimen containing predominately (U,Pu)₂C₃. The (U,Pu)₂C₃ is affected to an average depth of 24 μ . Metallography has revealed

*Reactor Development Program Progress Report, ANL-7302, January 1967, p. 51.

**Reactor Development Program Progress Report, ANL-7204, April 1966, p. 17; Reactor Development Program Progress Report, ANL-7230, June 1966, p. 79.

†Nicholson, M. E., Samans, C. H., and Shortleeve, J. F., Composition Limits of Sigma Formation in Nickel-Chromium Steels at 1200°F (650°C), Trans. Am. Soc. Metals 44, 601 (1952).

††Reactor Development Program Progress Report, ANL-7391, October 1967, p. 7.

three phases in this zone: (1) a bright phase, tentatively identified as $(U,Pu)Ni_5$, to an average depth of $7\ \mu$, (2) a gray phase, tentatively identified as $(U,Pu)O_2$, to an average depth of $14\ \mu$, and (3) $(U,Pu)C$ in the area directly beyond the two aforementioned phases to a depth of $24\ \mu$ from the fuel jacketing interface. Future microprobe analysis should provide positive phase identification.

4. Carbide--Synthesis

a. Pressing and Sintering of Carbide Fuel Pellets (P. A. Nelson)

Last Reported: ANL-7382, pp. 79-80 (Sept 1967).

Pellet pressing of high-purity $(U-15\ w/o\ Pu)C$ is being carried out with a double-acting hydraulic press. The pressing and ejection cycle is automatically controlled. The pellets are made from powder as received from the fluidized-bed process without grinding or binder addition.* The powder is manually fed into the die cavity with a volumetric measuring scoop, and a pressing pressure of 75,000 psi is used; with this technique, the pellet weight has been found to vary by $\pm 5\%$. At the present pressing rate, a maximum of 20 pellets may be produced each hour. Green pellets of satisfactory strength are produced by this technique, and they may be easily measured and weighed without damage. Green densities of the $(U,Pu)C$ pellets are about $8.1 \pm 0.1\ g/cc$.

The pellets are then sintered in a furnace heated by a graphite resistance element. To achieve the required density of $85 \pm 3\%$ theoretical, it has been necessary to sinter at a temperature of $2100^\circ C$ in an atmosphere of argon at 1 to 25 Torr. In runs in which the maximum sintering temperature of $2100^\circ C$ was held for several hours, the weight loss from the pellets was excessive (2 to 6%). Recently, the weight loss was reduced to 0.2% by utilizing a shortened heating cycle: less than 3 hr with approximately $1/2$ hr at the maximum temperature. In the latest run, all eight pellets met the density requirements, and five pellets also met the dimensional specifications.

5. Fuel Cladding and Structure--Jacket Alloys

a. Irradiation Studies of Fuel Jacket Alloys (R. Carlander)

Last Reported: ANL-7399, pp. 13-14 (Nov 1967).

The effect of preirradiation grain size on the postirradiation strength and ductility of irradiated vanadium-base alloys is being studied in order to specify the optimum preirradiation condition of the alloys for

*Although no binder is used, stearic acid is used as a die lubricant by pressing a stearic acid pellet between each carbide pellet pressing.

performance as cladding materials in the LMFBR. The results of tensile tests performed at 400 to 550°C have been reported.* The following results are for tests conducted at 650°C with specimens of V-20 w/o Ti and V-15 w/o Ti-7.5 w/o Cr alloy, which were irradiated in EBR-II subassembly XO13 to a nominal fluence of 5×10^{21} at $600 \pm 100^\circ\text{C}$.

An increase in the preirradiation grain size from 10 to 35 μ had little effect on the 650°C strength and ductility of irradiated V-20 w/o Ti alloy (see Table II.A.5). Increase of the grain size from 10 to 20 μ increases the yield strength from approximately 50 to 53 kg/mm², but an increase in grain size from 20 to 35 μ has no significant effect on the yield strength. The same increase in grain size decreases the postirradiation total elongation from 24 to 21%.

TABLE II.A.5. Tensile Properties at 650°C of Vanadium-base Alloys Irradiated in EBR-II Subassembly XO13 to a Nominal Fluence of 5×10^{21} at $600 \pm 100^\circ\text{C}$

Material (w/o)	Average Grain Diameter (μ)	0.2% Yield Strength (kg/mm ²)	Tensile Strength (kg/mm ²)	Uniform Elongation (%)	Total Elongation (%)
V-20 Ti	10	50.4	69.5	17.7	24.0
V-20 Ti	20	54.6	72.4	17.3	23.2
V-20 Ti	20	52.5	68.9	16.9	22.0
V-20 Ti	35	53.8	70.8	15.5	20.7
V-15 Ti-7.5 Cr	10	63.2	80.2	11.1	16.0
V-15 Ti-7.5 Cr	10	64.8	82.1	10.7	14.8
V-15 Ti-7.5 Cr	25	58.5	68.7	10.5	18.3
V-15 Ti-7.5 Cr	25	59.9	72.3	10.5	18.4
V-15 Ti-7.5 Cr	35	55.4	68.6	10.4	17.0
V-15 Ti-7.5 Cr	35	56.6	68.5	10.2	16.6

An increase in preirradiation grain size from 10 to 35 μ decreased the postirradiation yield strength of irradiated V-15 w/o Ti-7.5 w/o Cr alloy from approximately 64 to 56 kg/mm². The total elongation increases from 15 to 18% on an increase in grain size from 10 to 25 μ ; a further increase in grain size to 35 μ has no effect on the postirradiation total elongation. The postirradiation uniform elongation is unaffected over the entire grain size range.

The effect of grain size on the postirradiation strength and ductility of the irradiated vanadium-base alloys over the temperature range from 400 to 650°C is illustrated in Figs. II.A.2 and II.A.3. The values for

*Reactor Development Program Progress Report, ANL-7371, August 1967, pp. 80-83.

the results at 400 and 550°C were taken from previously reported data and were averaged when duplicate tests were performed.*

Fig. II.A.2. Postirradiation Strength and Ductility of Irradiated V-20 w/o Ti Alloy as a Function of Grain Size and Test Temperature

Fig. II.A.3. Postirradiation Strength and Ductility of Irradiated V-15 w/o Ti-7.5 w/o Cr Alloy as a Function of Grain Size and Test Temperature

The postirradiation yield strength of irradiated coarse-grained V-20 w/o Ti alloy (20 to 35 μ) is higher than that of the fine-grained alloy (10 μ) over the temperature range from 400 to 650°C. There is no consistent relationship between grain size and test temperature in terms of the postirradiation ductility. The ductility of the fine-grained alloy at 400°C is higher than that of coarse-grained material, whereas at a test temperature of 550°C the situation is reversed. The difference in ductility between fine-grained and coarse-grained material is small at any one temperature, and the curves shown in Fig. II.A.2 were drawn through the minimum and maximum points to represent the spread in ductility expected in irradiated V-20 w/o Ti alloy over the temperature range of 400 to 650°C.

There is no definitive relationship between grain size and post-irradiation strength of irradiated V-15 w/o Ti-7.5 w/o Cr alloy over the temperature range from 400 to 650°C. The coarse-grained alloy is stronger than the fine-grained alloy at 400 and 550°C, but not at a test temperature of 650°C. The coarse-grained V-15 w/o Ti-7.5 w/o Cr alloy has slightly better ductility than the fine-grained alloy at test temperatures of 400 and 550°C.

*Ibid., p. 12.

B. Physics Development--LMFBR

1. Theoretical Reactor Physics--General Fast Reactor Physics

a. Multigroup Methods (H. H. Hummel)

Last Reported: ANL-7391, pp. 16-20 (Oct 1967).

(i) Calculation of Fundamental-mode Adjoint Fluxes with an Ultrafine Multigroup Structure. A computer code has been written to calculate multigroup quantities adjoint to the results of real multigroup calculations performed by the MC² code.* The fundamental property that makes this calculation different from the traditional adjoint calculations is that it uses an ultrafine group structure employing up to 2100 energy groups, rather than a collapsed cross-section set which in general would employ only from 20 to 70 groups.

The ultrafine group structure has been shown previously** to be very effective in the determination of the fine behavior of the real fluxes in typical fast breeder reactors. Similarly, it has now been found that the ultrafine group structure makes it possible to determine considerable oscillations in the fine behavior of the adjoint fluxes.

Some results for the real and adjoint fluxes (ϕ_0) and currents (ϕ_1) of ZPR-3 Assembly No. 48† are shown in Figs. II.B.1 and II.B.2. The results of the adjoint calculations are approximately the same as would be obtained with a coarser group treatment in the energy range above 1 keV, where the adjoint is rather smooth and constant with energy. The oscillations that appear in this range can be traced to an imperfect treatment of the inelastic scattering matrix in the current MC² formulation. In the energy range below 1 keV, however, the behavior of the adjoint flux obtained with the ultrafine group treatment is especially striking, and the large variations within small energy ranges might affect considerably the results of perturbation calculations.

* Toppel, B. J., Rago, A. L., and O'Shea, D. M., MC²--A Code to Calculate Multigroup Cross Sections, ANL-7318 (to be published).

** Travelli, A., "A Comment on the Comparison of Theoretical with Experimental Neutron Spectra in Fast Critical Assemblies," ANL-7320, Proc. Int. Conf. on Fast Critical Experiments and their Analysis, pp. 481-485, Argonne, Illinois, October 10-13, 1966; Toppel, B. J., and Hummel, H. H., "Comparative Physics Analysis of Fast Reactors as a Function of Refinement in Multigroup Cross Section Preparation," Proc. Int. Conf. on Research Reactor Utilization and Reactor Mathematics, Mexico City, May 2-4, 1967.

† Broomfield, A. M., Amundson, P. I., Davey, W. G., Gasidlo, J. M., Hess, A. L., Kenney, W. P., and Long, J. K., "ZPR-3 Assembly 48: Studies of a Dilute Plutonium-fueled Assembly," ANL-7320, Proc. Int. Conf. on Fast Critical Experiments and their Analysis, pp. 205-215, Argonne, Illinois, October 10-13, 1966.

Fig. II.B.1. Real Flux and Current in ZPR-3 Assembly No. 48

Fig. II.B.2. Adjoint Flux and Current in ZPR-3 Assembly No. 48

b. Dynamics Studies (H. H. Hummel)

Last Reported: ANL-7399, p. 15 (Oct 1967).

(i) System for Accident Initiation and Feedback Simulation in the QX1 Program. The primary purpose of the system is to mockup external initiating mechanisms such as control-rod movement or fuel

movement, coolant-velocity change or coolant-temperature changes. External responses to the system conditions, such as rod movement triggered by power or period conditions, and coolant velocity change triggered by coolant exit temperature, are also included. The secondary purpose is to provide crude internal response to internal conditions, such as coolant-density changes triggered by coolant vaporization temperature or clad rupture (i.e., clad temperature).

The method used is as follows: The user specifies time-dependent multiplying factors which operate on the temperature, volume fraction, or velocity of a given material in a given region. These operations are triggered by any one of a number of conditions as described below. Once triggered, the multiplying factors are applied on a fixed time base independent of other system conditions until they are turned off by another set of conditions. Up to 100 different sets of modifiers and conditions may be given. The system is referred to as the "MI Trigger System."

The time origin of a given MI (modifier) is set by any one of the following conditions:

- (1) fixed time relative to the origin of the problem;
- (2) temperature of a given material in a given region;
- (3) volume fraction of a given material in a given region;
- (4) power level relative to the initial power level in a given region;
- (5) the inverse period α .

The MI may be triggered by a "greater than" or "less than" criterion in comparing the current value of the specified parameter with the given trigger value. It must be equal to 1.0 at its origin and have equal time intervals between later values. The real time origin will be set at the trigger time, and will always be one of the normal time-step end points, even though the trigger conditions may have been met somewhere during the last normal time step. Since these time steps will be short in any case, it was believed to be unnecessary to interpolate to the exact initiation time. After initiation, the value of MI at any time will be found by three-point interpolation in the MI values.

Initiation or termination of any modifier forces flux shape recalculation at the appropriate time point.

2. Experimental Reactor Physics--Fast Critical Experiments-- Experimental Support

Last Reported: ANL-7399, pp. 16-22 (Nov 1967).

a. Neutron Spectrometry (W. G. Davey)

Tests of the proton-recoil spectrometer continued with preliminary measurements in both ZPR-3 and AFSR. Results indicate that the problem of electronic noise created by the computer (see ANL-7399, p. 17) has been resolved. Cables were run from AFSR and ZPR-3 to a centrally located room located approximately 150 ft from each reactor. With the pre-amp and detector this far from the central processor, no detectable pickup from the computer clock could be observed in the fast channel.

As a final check before measuring the spectrum in ZPR-3, a test spectrum will be run in AFSR. This will be done in the thermal column, which has been fitted with a boron and lead sleeve to create an approximately $1/E$ flux. Tests for gamma-to-neutron ratio and neutron intensity are presently being carried out.

b. Heterogeneity Studies (W. G. Davey)

Work continued on the development of suitable foil-counting techniques for heterogeneity studies in large, fast, power reactors. The initial work is aimed at precise determination of relative fission rates of a number of fissile species.

The technique being investigated is directed at determining the relative concentrations of individual fission-product isotopes by means of their characteristic gamma rays, using the high resolution available from lithium-drifted germanium detectors. The initial work on the products of ^{235}U fission was reported in ANL-7399. The work has now been extended to longer "cooling" times and higher energies, revealing additional isotopes and possible interferences. Within the limits of gamma-ray energies greater than 200 keV and less than about 3 MeV, and half-lives between 30 min and 65 days, the major gamma-ray lines expected from all fission-product chains having fission yields greater than about 2% have been found and identified.

Of these, approximately 20 apparently well-resolved gamma rays were selected for close study of possible interferences. The apparent half-life of each of these was determined in satisfactory agreement with literature values in most cases. Where possible, the apparent intensity ratios of different gamma rays from the same isotope, and of gamma rays of different isotopes in secular equilibrium, were determined as a function of time. This procedure uncovered a number of cases of interference. A number of isotopes remain, distributed on both peaks of the mass-yield curve, for which interferences have not been detected.

Ideally, the method would make use of four or more fission-product isotopes or equilibrated mass chains, each having two or more interference-free gamma rays. The initial conclusion is that this is probably feasible, provided that the required statistical precision can be obtained with reasonable exposure time (in ZPPR) and reasonable counting time.

The limiting precision for low-level exposure of ^{235}U , ^{238}U , and ^{239}Pu will be determined in measurements now beginning. The first step is a set of survey exposures of ^{238}U and ^{239}Pu . Although major differences in gamma spectra are not expected, minor differences are expected to help reveal any additional interferences.

Preliminary systems design of equipment for the application of this technique to ZPPR measurements has been started. Multichannel analysis will be required.

c. Computer Applications (W. G. Davey)

Design work continued on the ZPPR computer system and its interactions with the experimental equipment. The Request for Proposal, which will be submitted to computer firms for the basic system components, has been completed. Areas of recent effort have included a study of readout provisions for the experimental electronics, and the preliminary designing of the multiplexer system needed for scanning the outputs of the digital position encoders on the reactor control rods. Parts for the prototype models of the above systems have been ordered.

d. Reaction Rates Related to Fuel Cycles (R. Gold)

(i) ^{236}U Capture Rate Measurements. A counting technique for measuring rates of capture of neutrons by ^{236}U in a zero-power fast reactor has been developed. Capture in ^{236}U was measured by observation of ^{237}U decay, whose rate of decay was determined with an absolutely calibrated gamma-gamma coincidence system. Germanium detectors were used in the coincidence system.

With this method, the ^{236}U capture rate was measured in ZPR-9 Assembly 18 with a standard error of less than 3%. The in-core ratio of the number of captures by ^{236}U to the number of fissions of ^{235}U at the irradiation position was found to be 0.248, with a standard error of approximately 4%.

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

Last Reported: ANL-7399, pp. 22-30 (Nov 1967).

a. Doppler Coefficient. The newly modified Doppler equipment for ZPR-3 was vacuum tested. Because of a deviation between the prints supplied by the Andar Corporation for the Torrgett Ion Pump and the pump supplied by them, the connection of the Doppler vacuum equipment to the pump had to be reworked. The new connection is being leak checked preparatory to use of the Doppler equipment for measurements.

b. Reactor Technique Development. The sample changer to be used for reactivity traverses in ZPR-3 has been assembled and bench-operated. The operation of the changer has proved to be more straightforward than originally planned, eliminating the need for several operational interlocks called for in the initial design. A 5-sec application of electric power is sufficient to position a new sample precisely in line with the traverse tube.

Whereas initially air pressure from commercial vacuum-sweeper-type units to transfer the sample from the changer to the in-core position was planned, subsequent tests have shown that the vacuum available from these units will work equally well, with possible advantages.

Present plans call for using the vacuum to hold the sample against the traverse mechanism rod for most measurements, at least until the effect of the sample on the reactor can be semiquantitatively evaluated. A small detector has been fabricated which can be used to measure the speed of a dummy sample through the traverse tube. The additional parts needed for these bench tests are being fabricated.

c. Mockup Studies. Initial criticality was achieved for Assembly 51, the simple nonsplit cylindrical core of the FFTF Phase-B critical program in ZPR-3, with 215 kg of $^{239+241}\text{Pu}$. The approach to critical was made by replacing depleted uranium in the preloaded core zone with plutonium plates. The approach was interrupted at various stages to permit evaluation of tests with the proton-recoil spectrometer. The spectrum measurement could be made only in the subcritical configuration because of limitations for the neutron count rate. Thus, after completion of the live-time clock and the necessary computer programs, part of the core will be unloaded to do the spectrum measurement.

After criticality was achieved, the sensitivity of the reactor instrumentation was adjusted to permit operation at power levels up to 250 W. This was necessary because of the low-density reflector and high neutron leakage of this assembly.

The safety-rod worth was determined by use of inverse kinetics measurements to be less than 1.5% $\Delta k/k$. The worth of the rods was then

increased by removing plutonium from adjacent reactor matrix positions and loading it in the safety rods. In this manner the total worth of the rods was increased to 1.8% $\Delta k/k$ without changing the reactor plutonium inventory.

Zones I and II of the radial reflector were then loaded to their maximum radial thickness within the existing inventory of materials and the core geometry was adjusted to the reference critical geometry. With this geometry the measurement of the worths of core drawers at the core radial boundary was begun.

The "as-built" compositions of the core and the various reflector zones are being computed, using the materials constants of the new plutonium and sodium. These compositions, along with the geometric description of the reference critical, will be published in the next monthly report.

4. ZPR Fuel and Nonfissile Materials--Technical Assistance

a. Fuel Element Fabrication Development (J. E. Ayer)

(i) Oxide Rod Elements

Last Reported: ANL-7391, pp. 31-33 (Oct 1967).

Thirty tubes were cut to length as shown in Fig. II.B.3. One hundred end plugs have been made of Type 304 stainless steel, and the bottom plugs have been TIG welded into the jackets.

Fig. II.B.3. ZPRR Oxide Rod Element

A testing apparatus to thermal cycle samples of the jackets to be loaded has been built and is in operation. Three of the thirty assembled fuel tubes were randomly selected for test. One sample has been cycled over two thousand times between 100 and 600°C in air. The time at 600°C was of the order of seconds. The purpose of this test is to obtain a quick assessment of the adherence of the oxide layer on the jacket surface. A second assembled jacket is being cycled under the same conditions except that it is being cycled two hundred times and held at 600°C for 4 hr. Upon completion of thermal-cycling tests the weight gain or loss of each sample will be determined as well as any dimensional changes that may have occurred during testing. Each jacket will then be sectioned and metallographically examined to ascertain the extent of corrosion.

(ii) Instrument Traverse Elements

Last Reported: ANL-7391, p. 33 (Oct 1967).

The jacket hardware for the 50 traverse elements has been fabricated. The jackets are standard ZPPR jackets with a ferrule welded into the sides. The holes were made by electrical-discharge machining, and the ferrules were attached by TIG welding.

Core plates for the traverse elements will be made from standard ZPPR core plates and elements submitted to ANL by commercial vendors as evidence of their readiness to begin full-scale production of standard elements. A sufficient number of plates are presently on hand to fabricate the required 50 instrument traverse elements.

(iii) Special Measurement Elements

Last Reported: ANL-7317, p. 46 (March 1967).

The elements requested for measurement of the Doppler coefficient are described in Table II.B.1.

TABLE II.B.1. Doppler Coefficient Elements

No. of Elements	Composition		Jacket		Description
	% PuO ₂	% UO ₂	ID (in.)	Material	
4	100	0	1	Invar	Non-nonexpanding
2	100	0	1	Inconel	Nonexpanding
3	100	0	1	Inconel	Freely expanding
6	100	0	1/2	Inconel	Freely expanding
1	67	33	1	Inconel	Freely expanding
2	50	50	1	Inconel	Freely expanding
3	50	50	1/2	Inconel	Freely expanding
1	50	50	1	Invar	Non-nonexpanding
1	33	67	1	Inconel	Freely expanding
2	12.5	87.5	1	Inconel	Freely expanding
1	12.5	87.5	1/2	Inconel	Freely expanding
1	0	100 ²³³ UO ₂	1	Invar	Non-nonexpanding
1	0	100 ²³³ UO ₂	1	Inconel	Freely expanding
4	0	100 ²³³ UO ₂	1/2	Inconel	Freely expanding

A 1-in.-dia non-nonexpanding, a 1-in.-dia freely expanding, and a one-half-in.-dia freely expanding Doppler element of the type requested for fabrication this year are shown in Fig. II.B.4. During this calendar year, the fuel material necessary to fabricate all the elements shown in Table II.B.1 has been obtained, blended where necessary, and converted to unfired granules. The $^{233}\text{UO}_2$ and 100% PuO_2 composition granules have been fired and loaded into jacket components. The elements have been assembled, welded, inspected, and accepted.

Fig. II.B.4. Doppler Coefficient Element Types

5. ZPR Fuel and Nonfissile Materials--Procurement

a. Properties of Zero-power U-Pu Base Alloys (H. V. Rhude)

Last Reported: ANL-7382, p. 61 (Sept 1967).

Corrosion tests in air of specimens R418 (U-36 w/o Pu-2.5 w/o Mo) and R420 (U-36 w/o Pu-3.7 w/o Mo). After five months of exposure, both specimens have tarnished to a dark black, but are still sound and show no signs of cracking or disintegration. R418 exhibits a very slight amount of surface powdering, which began after the first month. The specimen increased in weight by 0.08% during the first five weeks of the test but has gained no weight since. R420 has shown no surface powdering, but increased in weight by 0.09% during the first five weeks of the test and has shown no weight gain since.

6. FFTF Critical Experiment Program (D. Meneghetti)

Last Reported: ANL-7399, pp. 37-39 (Nov 1967).

a. Proposed Assemblies IIa and IIb of Phase B. The FFTF split-core inverted-cone concept is geometrically complex. Furthermore, the

core is designed with nonvertical fuel assemblies such that a 30 to 35% variation in cross-sectional area occurs over a core height of approximately 100 cm. This axial variation in composition together with the rather large gap-region produced by the test region necessitates separation of these two effects by critical experiments involving idealized simple configurations. These will allow a

basis for testing the methods of analysis in addition to indicating possible problem areas which may subsequently arise.

Fig. II.B.5. FFTF Phase-B Critical Experiment Assembly IIa

The proposed Assembly IIa is a cylindrical-shaped uniform-composition core similar to Assembly I except for the presence of an axial gap which separates the core into two driver regions (see Fig. II.B.5). It is proposed that in this assembly no fuel material be in the gap region. Instead, the gap would contain about 80% sodium and 20% stainless steel by volume. This approximates the gap-region composition in the absence of test loops. Furthermore, since the primary purpose of this assembly is to study the gap effect, a gap region without the added complications of fuel-plate heterogeneities would be analytically more tractable.

The volume of the core will depend upon the gap width and upon the cross-sectional area profile of each driver half. Estimates indicate that with a three-drawer-wide gap (~6 in.) it is likely that criticality may be attainable within the present maximum allowable, total plutonium loading of 430 kg. In any event, formation of the

gap region would occur first by formation of a one-drawer-wide gap, a two-drawer gap, and then a three-drawer-wide gap. If criticality is not attainable with the largest gap, the experiments on this idealized system can be carried out with the system having a two-drawer-wide gap.

The proposed Assembly IIb (see Fig. II.B.6) is a modification of Assembly IIa for the purposes of studying the effects of fueled vertical test loops in the gap region and their interaction upon each other, as well as to study the effects of core-driver dilution because of presence of control-rod channels. For both the simulated test-loop drawer clusters

Fig. II.B.6. FFTF Phase-B Critical Experiment
Assembly IIb

and the simulated control-position drawer clusters the suggested composition is about 50 v/o of the drawer composition with additional 30% sodium and 20% stainless steel to account for the remaining volume fraction. The control-simulating positions and number of positions shown in Fig. II.B.6 are only indicative, since the present maximum loading limit of 430 kg of total plutonium may severely curtail the number of such clusters of low fuel density.

b. Experimental Work on Assembly I of Phase B (ZPR-3 Assembly 51). Initial criticality was achieved for Assembly 51, the simple nonsplit cylindrical core of the FFTF Phase-B critical program, in ZPR-3 with 215 kg of ^{239}Pu and ^{241}Pu . The approach to critical was made by placing depleted uranium in the preloaded core zone with plutonium plates. The approach to critical was interrupted at various stages to permit the evaluation of the proton-recoil spectrometer to be made. It was determined that the

subsequent spectral measurements will be made in a subcritical configuration due to neutron count-rate limitations.

After achieving criticality, the sensitivity of reactor instrumentation was decreased to permit operation at powers up to 250 W. This is necessary because of the differing neutron leakages depending on the characteristics of a given assembly.

The combined worth of the eight safety rods was checked using inverse kinetics and found to be less than the required 1.5% $\Delta k/k$. The worth of the rods was increased by removing plutonium from adjacent reactor matrix positions and loading it in the safety rods. In this manner the total worth of the rods was increased to 1.8% $\Delta k/k$ without changing the reactor plutonium inventory.

The Radial Reflector Zones I and II were then loaded to their maximum radial thickness within the existing inventory and the core

geometry adjusted to the reference critical geometry. In this geometry the measurement of the worths of core drawers at the core radial boundary was begun.

The computation of the "as-built" compositions of the core and the various reflector zones is being done using the materials constants of the new plutonium and sodium. These compositions with the geometric description of the reference critical will be published in the next report.

c. Calculated Estimate of Effect of Spontaneous Plutonium Fission upon Subcritical Measurements of Multiplication Change due to Fuel Removal. Previous studies of the subcritical measurement of k_{eff} (Progress Reports for January 1967, ANL-7302, pp. 27-28, and for February 1967, ANL-7308, pp. 13-15) were concerned with the dependence of the inverse of multiplication on k_{eff} when poison is added to produce the subcriticality. When the subcriticality is caused by the removal of part of the plutonium fuel, the plutonium spontaneous-fission source is depleted also. Then the change in multiplication must be corrected for the change in the value of the source.

The size of this correction was investigated by comparison of two computations for Assembly 51 using the MACH1 code. In both of the computations a subcritical $\Delta k = -0.0101$ was produced by substitution of canned sodium for core material in the P-16 drawers. In this comparison the subcritical multiplication was taken to be proportional to the computed ^{10}B reaction rate at the outer edge of radial reflector II. First the ^{10}B reaction rate was found to be proportional to 2.54 with the plutonium spontaneous-fission source still present with the sodium in the central drawers. In the second computation the spontaneous-fission source was removed from the sodium to give a reaction rate proportional to 2.40. The corresponding multiplication is analogous to the one experimentally measured. Thus a correction of about +6% must be applied to the Δk obtained from the second multiplication to obtain the desired Δk of -0.0101 which corresponds to the first multiplication.

7. ZPPR Construction (H. Lawroski)

Last Reported: ANL-7399, pp. 31-33 (Nov 1967).

The major items procured by ANL for the reactor assembly have been received or are in transit. The second rod-drive mounting plate for the reactor assembly was received on November 27, inspected, and accepted. Air-sampling pumps for the alpha air monitors were received on November 20, and the electronics components were shipped from the factory on December 15.

The contract for the installation of the ZPPR reactor assembly and associated equipment was signed on December 18 with the Hunter-Saucerman Construction Co., Idaho Falls. Arrangements have been made to furnish the contractor a work area to start the cleaning of the matrix tubes.

The construction contractor completed placement of gravel on the reactor cell roof. The remaining filter frame was welded into position. The frame containing the access door was reworked to remove interferences and was installed. The one remaining flexboard panel of the covering around the backup containment structure was completed; about 50 percent of the holes in the covering due to misfits during assembly have been filled. The high-efficiency filters were placed in the backup containment structure but have not been installed in the filter frames. One of the two outside access stairs to the backup containment structure from the ground level was completed.

All electrical components have been installed and checked out. The contractor has started preparation of the as-built drawings. The electrical bulkhead through the frame of seal door D66 was tested and found to leak between the gasket and the seal plate. The leak was caused by separation of the aluminum seal plate from the gasket because of flexing. The contractor was instructed to install a reinforcing angle-iron frame on all aluminum seal plates to correct the problem.

The contractor had difficulty in maintaining alignment of seal door B67 and requested advice. ANL representatives inspected the door and found the hinges out of alignment, which put an unacceptable distortion on the bearings and shafts. The contractor had slotted various bolt holes in attempting to adjust the location of the door. ANL advised the contractor to align properly the hinges of all doors for both height and vertical attitude, and to adjust the doors to the seal frames by adjustment of the end mountings of the door-support arms. This procedure was satisfactorily demonstrated on door D67.

The bed and tables were installed in the reactor pit and aligned satisfactorily. The loading tube between the cell and the workroom was installed and grouted into place.

The cell air-handling system was completed, and preliminary data indicate that the specified temperature variation of $\pm 1/4^{\circ}\text{F}$ can be obtained by proper balancing.

Rework of the cell crane has been held up pending completion by the contractor of revised drawings for the trolley drive.

The support wing was accepted for beneficial occupancy with a number of exceptions. Major exceptions included completion of the AFSR room, adjustment of switches in the suspect exhaust system, installation of emergency lights in the fan loft, installation of unit heater switch in the fan loft, and correction of the imbalance of the AC-101 condenser fan.

Major work items remaining in the mound area of the facility are (a) leak testing the backup containment structure, (b) application of the epoxy floor covering in the cell, corridors, and workroom, (c) balancing the air-handling system, (d) properly installing the seal doors, (e) reworking the cell crane, and (f) leak testing the seal doors and bulkheads.

C. Component Development--LMFBR

1. Sodium Technology Development--Engineering Development (S. B. Skladzien)

a. Sodium Quality Control

Last Reported: ANL-7391, p. 43 (Oct 1967).

Construction of the prototype submerged oxide collector is virtually complete; its installation is being planned.

A method of reacting steam with the contents of a spent cold trap has been considered for use in analyzing the exposed mesh of the trap. In this method water or steam reacts with the residue sodium metal and sodium oxide, and the amount of hydrogen evolved, the water used, and the alkalinity of the resulting solution are measured. Chemical analysis determines the concentrations of hydride, hydroxide, oxide, sodium, and other metals. This method was chosen over the standard practice (extracting a core sample from the cold trap mesh and analyzing for total oxygen, hydrogen, etc.) because it should be more informative; the conventional practice yields misleading results because the amount of sodium metal that is entrained in the cold trap determines to a large degree the concentration of the impurity species. The method selected will enable us to determine the amount of sodium metal in the cold-trap residue as well as the composition of the residue.

In preliminary tests of the analytical method, the amounts of hydrogen gas evolved when known quantities of sodium metal and sodium metal-sodium oxide mixtures were reacted with steam in a closed vessel were measured. The results, shown in Table II.C.1, are very encouraging. If the reaction between the cold-trap residue and water is carried to completion so that the resulting products are in a water solution and in the

TABLE II.C.1. Tests of Steam-reaction Method for
Determining Content of Cold Trap

Weight of Na (g)	Weight of Na ₂ O (g)	Hydrogen Produced (Liters at STP)		
		Calculated	Observed	Error (%)
15.24	0	7.42	7.35	-0.94
17.77	0	8.65	8.63	-0.23
24.54	0	11.94	12.01	+0.58
11.05	130.98	5.38	5.08 ^a	-6.0
13.65	24.25	6.64	6.63	-0.15
9.22	4.32	4.48	4.48	0

^aIncomplete reaction; solid residue remained in reaction chamber.

gaseous state, the kinds and amounts of products obtained are in proportion to the initial composition of the sample. Thus, if water reacts with a mixture of sodium oxide and sodium, the amount of hydrogen given off will be proportional to the amount of sodium metal initially present in the mixture and will be independent of whatever intermediate compounds are formed. Four of the possible intermediate reaction steps and their net reactions are:

b. Sodium Quality Measurement

Last Reported: ANL-7399, pp. 40-41 (Nov 1967).

(i) Laboratory Techniques. A second sodium sample from the 1200°F loop in Building 308 has been analyzed for trace impurities. The metallic-impurity concentrations measured agree with those measured earlier and are:

<u>Impurity</u>	<u>ppm</u>	<u>Impurity</u>	<u>ppm</u>
Cr	1.7	Mn	0.6
Cu	1.7	Bi	<1.0
Fe	12.7	Sn	<5.0

Oxygen analyses made of this sample by mercury amalgamation showed 73 and 70 ppm, and vacuum distillation indicated 71 and 97 ppm, which are considerably greater than in the earlier sample, probably because of contamination during sampling.

A gas chromatograph is being set up and calibrated to identify and measure trace gaseous impurities in samples of the gas covering the sodium. The instrument is equipped with a helium-ionization-type detector that, under proper operating conditions, should detect the permanent gases down to the low-ppm range.

(ii) In-line Techniques. Further temperature-response tests of the resistivity meter and the United Nuclear Corp. oxygen-meter cells Nos. 1 and 2 were carried out. Two tests at 312-316°C showed that for a 1°C change the resistivity meter changed 0.70 microhm. The resistivity

meter has been operating for more than 30 days and has varied not more than 1 microhm. The variation was due mainly to drift of the constant-power supply, because the temperature variation was less than 0.2°C . However, the response of the oxygen-meter cells varied between 0.42 and $0.52\text{ mV}/^{\circ}\text{C}$ for cell No. 1 and 0.945 and $0.60\text{ mV}/^{\circ}\text{C}$ for cell No. 2. Cell No. 1 responded sluggishly in both tests; cell No. 2 responded quicker but unsteadily. Daily monitoring of voltages from the oxygen meters has shown changes occasionally as much as 0.5 mV at a constant 316°C . These changes are probably caused by system ground-current variations, room-air variations, or pressure changes of the argon blanket gas.

The oxygen content of the analytical loop is being increased by adding sodium hydroxide in pellet form. The responses of all instruments will be compared with chemical analyses, and the hydrogen content of the blanket gas will be monitored.

2. Reactor Mechanisms Development--Materials Evaluation

a. Sodium Effects on Wear Properties of Materials (E. S. Sowa)

Last Reported: ANL-7357, pp. 77-78 (July 1967).

Two runs were made with a Carboloy-98B shaft and vee-blocks of Clarite-T1 alloy (0.73% C, 18.00% W, 4.0% Cr, 1.10% V, and the balance Fe), materials that have very different surface energies.

In each run two pairs of these materials, one in sodium and the other in inert gas, were used. The tests started at 300°F . With the temperature held steady, the equipment was operated for approximately 2 hr while data were collected on wear and friction coefficients. Then the tests were repeated at 500, 700, 900, 1100, and 1200°F .

The vee-block samples had been vapor-blasted to a very clean and uniform surface finish, so the initial friction coefficients in the sodium were approximately 0.650, dropped to 0.450 as the sodium tried to lubricate the surface, and rose to 0.700 as higher-temperature wetting of the surface was encountered. By contrast, the sample in inert gas started with a friction coefficient of approximately 0.180 and dropped to 0.120 at 1200°F .

The wear rates, on the other hand, were higher for the sample in inert gas than for the sample in sodium at 300°F , the point at which the data were examined.

Further reduction of data is in progress.

Initial heating of the Component and Material Evaluation Loop (CAMEL), the facility that will be used for dynamic tests, has started before

the introduction of sodium. Several leakchecks revealed that gas was escaping from the loop when it was pressurized at 250°F. After the leaks were isolated and corrected, the loop held a vacuum overnight. The pressure transducers are being calibrated to produce normalization curves that will be used during high-temperature operation.

b. Fuel-pin Thermocouple Development (A. E. Knox)

Last Reported: ANL-7399, pp. 42-43 (Nov 1967).

(i) Electrical Insulators. The analysis of heated thoria beads with a laser microprobe was not successful. The beads have been submitted for ordinary spectrochemical analysis.

(ii) Thermoelement Properties. While the cause for the failures of the barewire thermocouples is being determined, the stability of a tantalum-sheathed, thoria-insulated W-3% Re/W-25% Re thermocouple is being measured at 2400°C. This thermocouple has had a nearly constant emf output for a period of 30 hr and will be tested for at least 200 hr.

(iii) Material Compatibility. Tests of the UO₂-tantalum compatibility sample TU-3 have been rescheduled for January because of furnace problems.

(iv) In-pile Tests. An evaluation of the ANL in-pile test requirements has indicated that it would be advisable to subcontract the in-pile test program. Accordingly, a letter of inquiry has been sent to industry for an expression of interest. A request for proposal is in preparation and will be sent to interested vendors.

c. Fission Gas Pressure Transducer Development (J. R. Folkrod)

Last Reported: ANL-7399, p. 43 (Nov 1967).

(i) NaK-filled Capillary System. The 1/8-in.-dia bellows system has been filled with NaK successfully. A voidless fill was accomplished by (1) properly cleaning the 25-ft-long, 0.012-in.-ID capillary tube and (2) heating the entire system while filling. Very little bellows movement was required to obtain full deflection of the transducer diaphragm, giving a 15-mV output. A nickel bellows was used for this demonstration because of its availability and low cost; however, it became quite brittle after heating and broke near the weld joint before further tests could be made. Nevertheless, it has been demonstrated that a working, miniature, fluid-filled device can be built. A full report is being written.

(ii) Null-balance System. The first of two MSA null-balance devices has been tested. It worked well between 72 and 400°F, but failed

between 400 and 600°F. The device works by movement of a bellows against a contact probe to switch an electrical circuit, which allows a solenoid valve to open and reference gas to enter, balancing the pressure across the bellows. A very slight overpressure of the reference gas causes the bellows to break contact, closing the solenoid valve. The reference-gas pressure is taken to be equal to the gas pressure on the opposite side of the bellows, which in a fuel pin would be the fission gas.

In the test runs between 72 and 400°F, the reference-gas pressure was in close agreement with the simulated fission-gas pressure, being only approximately 1/4 psi off over the range. Between 400 and 600°F, shorting of the probe, or another part previously insulated from the ground, took place and caused the solenoid valve to open. The reference gas entered, and a correct reading was no longer possible. At times there was as much as 100 psi differential across the bellows. The system became very erratic and of no practical use. These tests were repeated many times to confirm the results. X rays of the device have been inconclusive in defining the cause of the shorting.

The second unit is now being tested and is showing the same pattern as the first.

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

Last Reported: ANL-7399, pp. 43-44 (Nov 1967).

(i) Thermal and Mechanical Tests of Magnetic Materials. The Lavite furnace assembly and magnetometer are being used to determine the reversible effects on Alnico-5, -6, and -8 magnets having various length-to-diameter (L/D) ratios and stabilizing treatments. The results of these tests will be used to prepare the in-pile test samples and will aid measurements of the preirradiation sample.

A castable ceramic material has been used to prepare a new furnace body for the magnetometer test apparatus. Preliminary heat cycling indicates that this new material will be much more durable than the original Lavite. The finished surface of this new material is not as hard as desirable, but will probably be acceptable. Further tests on the new furnace body are continuing.

The B-H tester has been used to check samples of Alnico-8 magnets and stainless steel rods. The measured B_r and H_{ci} values of the Alnico-8 samples are very close to those derived from published curves for this material. The sensitivity of the tester was sufficient to display the hysteresis loop of the stainless steel samples, and the peak magnetizing force available was sufficient to put Alnico-8 well into saturation.

Further tests are being made on Alnico-5, -6, and -8 rod magnets to determine the material and irreversible temperature effects.

(ii) Preliminary Prototype Magnetic Flowmeter. Loop tests of the preliminary prototype permanent-magnet flowmeter have been completed. Check calibrations were made at 400 and 700°F after the expansion bellows in the flowmeter leg of the calibration loop was replaced and the loop made operational.

The flowmeter was operated an additional 1102 hr between the calibrations of August and November 1967. The flowmeter sensitivity decreased since the previous calibrations by 8.3% at 700°F and 9.5% at 400°F. After loop repairs in the immediate vicinity of the flowmeter, sensitivity decreased again. Although the exact cause or causes of these changes in flowmeter sensitivity due to temperature is not known, the changes will be correlated to the reversible temperature effects being determined in the magnetic-materials tests.

The flowmeter will be removed from the loop and dismantled for examination of all materials.

(iii) Eddy-current Flowmeter. A typical eddy-current in-core flowmeter has been designed and is being fabricated. A strike at a major supplier of high-temperature wire threatens to delay delivery unless an alternate can be found.

We plan to test three different flowmeter assemblies. The first will have the coils wound with low-temperature Class-A magnet wire and will be tested dry and at room temperature, with moving aluminum rods and cylinders simulating the sodium flow. The second unit will more closely resemble an in-core prototype; it will be wound with high-temperature wire and will be tested in a furnace to optimize the high-temperature coil construction. After the first two phases are completed, a third unit will be fabricated for use in the Sodium Calibration Loop.

e. Signal-lead Connectors for Sodium Service (A. P. Grunwald)

Last Reported: ANL-7399, pp. 44-45 (Nov 1967).

(i) Test of Electrical Connector in Sodium Vapor. As part of the program to develop signal-lead connectors for the use in the FFTF, the insulation resistance of a Physical Science Corp. 36-pin connector (plug) was measured while it was exposed to sodium vapor at various temperatures in a stainless steel capsule. The insulation between each pin (of the 36 pins, 18 are Alumel, 9 are Chromel, and 9 are stainless steel) and the stainless steel body of the connector is Durock, a proprietary material that contains six metallic oxides, and is boron free and nonhygroscopic when correctly processed.

The test capsule was heated in the furnace through a range of 100 to 209°C over 5½ hr. With 150 V (dc) applied in series with a vacuum-tube voltmeter having an input resistance of 11 megohms between the body of the capsule and two pins of the connector, no lowering of the insulation resistance was observed. In other words, the voltmeter continued to read zero throughout this part of the test; however, maximum sensitivity of the voltmeter was not realized then because the 150-V scale was used (whereas the 5-V scale was used for most of the remaining testing). At this point, the furnace was shut down.

When the test was resumed 64 hr later, measurements were:

<u>Time</u>	<u>°C</u>	<u>Insulation Resistance</u>
0900	100	16,500 megohms
1000	225	9,660 megohms
1115	258	5,680 megohms
1230	255	4,600 megohms
1400	299	2,880 megohms
1530	308	3,290 megohms
1630	245	8,650 megohms

The following day:

0845	235	11,750 megohms
0930	315	4,580 megohms
1030	355	1,060 megohms
1215	342	2,060 megohms
1255	407	69 megohms
1339	400	~0 ohms
1420	355	~0 ohms

Thus it is apparent that the insulation resistance is seriously lowered when the connector is exposed to sodium vapor at these temperatures. (The normal resistance of the Durock is $\sim 10^5$ megohms at 300°C and $\sim 4 \times 10^3$ megohms at 400°C.*) Results of a subsequent test are being analyzed.

A similar test will be performed on another connector at a constant 176°C (350°F) for approximately 48 hr.

f. Failed Fuel Locating Method Development (F. Verber)

Last Reported: ANL-7399, p. 45 (Nov 1967).

Work continues on the design of the sodium-recirculating system that, hopefully, will demonstrate that fission gas can be separated from circulating sodium to locate a failed fuel element in the FFTF. One

*Popper, Glen F., and Knox, Alvin E., FARET In-core Instrument Development, ANL-7161 (July 1966).

improvement involves lowering the return end of the 1/2-in.-dia recirculating gas loop and directing it downward so that the returning gas will impinge on the surface of the sodium. This should preclude the formation of dead space between the surface of the sodium and the elevation of the low point of the gas-return tube.

An electromagnetic pump and a sodium vapor trap are on hand and ready for use in the test assembly; 1-in. bellow-sealed valves are being repaired and will be ready for use.

D. Systems and Plant--LMFBR

1. 1000-MWe Plant

a. Contract Management and Liaison (L. W. Fromm)

Last Reported: ANL-7391, pp. 48-50 (Oct 1967).

(i) The Babcock & Wilcox Co. Subcontract. Work has concentrated on refining the reference design and performing trade-off studies. Two trade-off studies have been completed and seven others are underway.

A new cross-section library for plutonium-239 and plutonium-240 isotopes was completed. Additional reactivity calculations that have been completed for the reference design will be used in the safety-analysis work. The effects on safety of the release of stored energy in the fuel to sodium are being studied.

B&W has supplied functional requirements to the architect-engineer (Sargent & Lundy) and soon will provide the reference-design steam-cycle conditions. The architect-engineer is providing structural feasibility, design, and cost information for several reference concepts derived from the Task-I study.

(ii) Westinghouse Corp. Subcontract. Most technical work in Task I has been finished. The architect-engineer has essentially completed system layout drawings and cost estimates. Westinghouse has begun writing the Task-I report and associated topical reports.

Task-II work is underway; shop orders have been written for the work packages, and functional groups have started working on reactor core, reactivity control, fuel management, and turbine-generator systems.

(iii) General Electric Co. Subcontract. Fuel-cycle cost has been estimated, and plant-optimization studies are being adjusted accordingly. Instrumentation requirements to prevent propagation of fuel melt-downs are being studied.

Preliminary core arrangements and building designs have been completed; system capital cost differentials are being estimated.

(iv) Combustion Engineering Inc., Subcontract. The Task-I report has been distributed. In addition, the quarterly technical report (CEND-326) summarizing work during the quarter ending October 31 presents work finished after the Task-I report was prepared.

Design descriptions for the subsystems are being started as part of Task II. Calculations of control-rod requirements and worths in the reference design have been evaluated. The layout of reactivity control rods and drive mechanisms is being designed.

Calculations have been started for analyzing the following controlled accidents: (a) loss of flow, (b) control-rod ejection, (c) fuel-bundle drop, (d) ganged-rod withdrawal, (e) local plugging, (f) gas bubbles, and (g) startup.

(v) Atomics International Subcontract. Revision of the Task-I work plans to include parametric investigation of several major core variables has increased the work to be accomplished prior to selection of the reference core. This revision, together with the changed sequence of activities in Task-I, will delay the selection of the reference core and the preparation of the system design descriptions. However, it is anticipated that the Task-I work will be completed on schedule.

E. EBR-II

1. Research and Development

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

Last Reported: ANL-7399, pp. 48-56 (Nov 1967).

(i) Nuclear Analysis (R. R. Smith)

(a) Fission Product Release, November 23, 1967. A fission product release on November 23, 1967, was not immediately indicated by the results of gas analysis, since the reactor system was in the startup mode; however, the release was indicated by an abnormal increase in the Fission Gas Monitor (FGM) signal (see also Sect. II.E.3.a) and by the preliminary radioactivity-meter survey of the first gas sample (at 0940 hours). The meter survey indicated an activity level of 2 mR/hr compared to the usual saturation level of 1.0 mR/hr under 45-MW operating conditions.

A graphical summary of events and of the gas activity sensed by the FGM during the November 23 fission gas release are given in Fig. II.E.1. Assuming a time lag of 5 to 10 min in FGM response, the time of release is placed at 0855 hours by an extrapolation of the data given in Fig. II.E.1. From the similar data given in Fig. II.E.2, the time of the second release is placed at 1420 hours, November 24.

Fig. II.E.1. FGM Response to the Original Release: 08:55 11/23/67

three techniques, a significant radioactivity background exists which is the consequence of fissions in two ever-present components: (a) natural uranium impurities in the coolant (0.005 ppm) and (b) the contamination of fuel-element surfaces with enriched fuel material. Of the two, the contamination component is by far the larger and the more variable.

(2) Activity Change during the Period of December 6-10, 1967. A summary of radiometric analysis data for ^{133}Xe and ^{135}Xe is given in Fig. II.E.3. Following the completion of power coefficient measurements on December 1, the reactor was operated at 45 MW until 0900 December 10. At that time a series of reduced-power, reduced-flow tests was begun. During the period of December 1-6 the growth rates of ^{133}Xe and ^{135}Xe are considered normal. Since the half-lives of ^{135}Xe and its precursor ^{135}I are both shorter than those for the ^{133}Xe - ^{133}I pair, the ^{135}Xe activity saturates (i.e., reaches radioactive equilibrium) at an earlier time.

Fig. II.E.3. EBR-II Argon Cover Gas Activity Run 26C from December 1 to December 13

On December 6 the ^{135}Xe had reached its saturation value while the ^{133}Xe was approaching its saturation value. At approximately 1800 hours December 6, both activities began increasing, with ^{135}Xe tending to level off at a higher saturation value during the period December 9-11. Actually, reactor power was gradually reduced on December 10 from 45 MW at 0900 to shutdown conditions at 1954 hours. A restart was made the following day beginning at 1630 and the system was shut down, after reaching 25 MW, at 2300. The increase in ^{133}Xe after the shutdown of December 10 is the consequence of continued buildup through ^{133}I decay. Observations during the three-day period December 7-10 may now be interpreted as a fission product release. Since there was no indication of any meaningful or significant increase in the FGM and FERD signals at any time during this period, however, it was difficult to reconcile the results of radiometric gas analyses with an actual fission product release. The recordings from all three FERD channels and the FGM have been compressed and examined, but no abnormal indications have been found.

(3) Radiochemical Analyses for ^{131}I and ^{137}Cs .

Table II.E.1 summarizes the results of radiochemical analyses of sodium samples. It may be seen that no discernible increase occurred in the ^{137}Cs content of the primary coolant when compared with the November 23 sample, which was collected before the indicated time of activity increase.

TABLE II.E.1. Radiochemical Analysis for ^{131}I and ^{137}Cs (a)

Sample Date	^{137}Cs ($\mu\text{Ci/g}$ sodium $\times 10^2$)	^{131}I (d/min-g sodium $\times 10^4$)(b)
11/23/67	1.7	-
12/14/67	1.5	0.94
12/15/67	1.7	1.3
12/18/67	1.7	1.0

(a) Precision limits of analysis: ± 10 -20%.

(b) Values given have been corrected for decay as of 12/19/67.

Although comparable reference information for ^{131}I is not available, the result of a calculation of the saturated activity expected under normal 45-MW operating conditions is about

250 d/min-g sodium (corrected to 12/19/67). Although no increase was noted in the ^{137}Cs activity, the activity associated with ^{131}I increased approximately fortyfold. The failure to observe a significant increase in the ^{137}Cs content is the result of the relatively large amount of this species released to the system during the fission product release of May 24. Since ^{137}Cs has a long half-life (30 yr), radioactive decay has not reduced its concentration. On the other hand, all ^{131}I released during the May 24 failure has decayed. Accordingly, the effects of any ^{137}Cs released on December 6 are masked by strong background effects while those of ^{131}I are not.

(4) Postulated Failure Models. Because of the contrary evidence supplied by the FGM and FERD records, consideration has been given to a variety of failure models which conceivably could account for the increase in cover-gas activities (as indicated by radiometric analyses) and yet would give no indication on the FGM and FERD systems. Cover-gas samples were taken from the same pumped-gas line which supplies cover gas to the FGM system; thus, no difference in gas sampling was involved.

Several models involving gas-type failure mechanisms were considered, but each was rejected because gas-type failures should result in a rapid signal increase (over a period of minutes or perhaps hours) and then a decrease with continued operation. For example, during the gas-type failures of May 24 and November 23, the signal increases following failure, as indicated by the FGM and the results of radiometric analyses, were rapid, large, and unmistakable. The confirmatory release on November 24 was followed by 8 hr of operation at 25 MW, during which time the FGM signal decreased and was nearing the normal background level when the reactor was shut down. Such behavior indicates that after a gas bubble is released, the driving force for additional release (at that power level) decreases to nearly zero. Following pressure relief, the major mechanism for additional release is diffusion through a small hole.

Once the FGM signal reaches its maximum, no mechanism exists which would continue to supply precursors at the rate corresponding to the maximum signal. Thus the signal after reaching maximum decreases even though reactor power continues uninterrupted:

It is suggested that an extremely small defect below sodium level could account for the conflicting information indicated by the three monitoring techniques. Consider the case of a driver element or experimental irradiation capsule which has developed a very small hole below sodium level. For the conditions existing at the indicated time of failure it is reasonable to assume that all index species for all three monitoring techniques were saturated. These species include the following: the delayed-neutron-emitting species which are sensed by the FERD; ^{133}I and ^{135}I which feed ^{133}Xe and ^{135}Xe for the radiometric cover-gas analyses; and ^{88}Br , ^{87}Br , and ^{138}I , which are the parents of the ^{88}Kr , ^{89}Kr , and ^{138}Xe sensed by the FGM. The formation of ^{133}Xe and ^{135}Xe involves growth from their parents ^{133}I and ^{135}I , since the independent fission yields for ^{133}Xe and ^{135}Xe are essentially negligible.

Of the species mentioned above, only ^{133}I and ^{135}I have half-lives long enough to permit migration from their place of birth and more or less homogeneous mixture with the sodium bond. All the other halogens, which have much shorter half-lives, travel only 2-3 mils into the bond (by recoil) before they decay to rare-gas species which diffuse upward into the gas plenum of the fuel element. When bond sodium is forced through a defect, halogens, which travel with the sodium, are preferentially weighted by a large factor in favor of those with relatively long half-lives. Upon escape to the primary coolant, ^{133}I and ^{135}I , fixed chemically as halide ions in the bond, eventually decay to ^{133}Xe and ^{135}Xe , which diffuse upward into the cover gas plenum. The ^{133}Xe and ^{135}Xe trapped in the gas plenum of the fuel element do not enter the primary coolant; these species appear in the cover gas plenum only as a consequence of parental decay. The much-shorter-lived halogens ^{88}Br , ^{89}Br , and ^{138}I do not survive long enough to permit significant quantities to escape through the defect, but die a few mils from their place of birth.

The growth-rate curves of ^{133}Xe and ^{135}Xe activities given in Fig. II.E.3 approximate the behavior expected for a very slow and constant introduction rate of ^{133}I and ^{135}I species into the primary coolant. From the viewpoint of gas-activity kinetics, it does not matter whether the iodine precursors are being added mechanically to the system at a fixed rate (through a constant loss of bond) or through the hypothetical introduction of additional tramp uranium, for example. The additional ^{133}Xe and ^{135}Xe species will be indicated by a second buildup curve which will saturate at a rate dependent only on the half-lives of the respective iodine and xenon pairs. The amplitude of the second buildup curve, however, does depend on the leak rate.

Since the amplitude of the second buildup curve at saturation is a function of leak rate, the minimum fraction of the bond forced through the defect of a typical driver element during the period December 6-10 may be estimated if the signal-to-noise ratio (S/N) for a complete release is known. The S/N ratio is defined as the signal increase noted when a bare pin is inserted in the reactor divided by a signal associated with the ever-present tramp uranium component.

From the results of earlier "bare pin" studies,* a value of six may be estimated for the S/N ratio. Assuming that the new saturated value of ^{135}Xe is $0.0175 \mu\text{Ci/ml}$ and its original value was $0.0105 \mu\text{Ci/ml}$, a 100% bond release would cause an increase of 6×0.0105 or $0.0630 \mu\text{Ci/ml}$. Since the increase is only $0.0175 - 0.0105$ or $0.0070 \mu\text{Ci/ml}$, the fraction of bond released during the period December 7-10 is only $0.0070/0.0630$, or about 11%.

(5) Conclusions

(A) A significant increase was noted in the ^{133}Xe and ^{135}Xe activities in the cover gas beginning about 1800 December 6.

(B) No significant increase was noted in the signals from the FGM or FERD systems before, during, or following the indicated time of increase.

(C) No abnormal indication was noted in the readings of outlet subassembly thermocouples during this period of time.

(D) An approximate fortyfold increase in the ^{131}I content of the primary sodium was noted. No estimate of the ^{137}Cs increase could be made. Assuming an S/N ratio of 6, the entire ^{131}I bond inventories of approximately seven driver elements or a single experimental capsule would be needed to explain the ^{131}I release.

(E) The qualitative behavior of the indicated release is consistent with a mechanism involving a very small defect which releases bond sodium slowly over a period of many days.

(F) If a driver fuel element is involved, at least 11% of the bond has been lost.

(G) No significant amount of fuel has been exposed to the primary coolant; otherwise, increases would have been noted in the FGM and FERD signals.

(H) The fission product releases of November 23, 1967 and of December 7, 1967 are considered to be unrelated.

*Smith, R. R., and Doe, C. B., Cladding Failure Simulation Tests in EBR-II, ANL-7067 (Dec 1966).

(c) Postirradiation Inspection of Subassemblies Discharged from Run 25. Postirradiation examinations were completed on subassemblies discharged from Run 25: eleven driver-fuel subassemblies, six Row-6-type driver-fuel subassemblies, and six control rods. Measurements for straightness, twist, and dimensional changes across the buttons were made on the 17 driver subassemblies. Since control rods are not equipped with spacer buttons, measurements conducted on these were limited to straightness and twist. Preirradiation twist and dimensions were measured with empty hex tubes, whereas the corresponding postirradiation studies were made with the completed subassemblies.

(1) Twist. Specifications permit a tolerance of 1° twist in the tubing used for subassembly fabrication. The maximum twist found in a discharged subassembly, including control rods, amounted to 0.9° .

(2) Straightness. Specifications permit a maximum tolerance of 80 mils (± 40 mils from true straight) in the straightness of a subassembly. The results of measurements conducted on 17 driver subassemblies are summarized in Fig. II.E.4. In this figure the vertical axes define indicator positions 1-6 where the actual elevations may be established from the following: position 1, 6.00 in. from the top of the hex tube; 2, 19.25 in.; 3, 32.50 in.; 4, 45.875 in.; 5, 59.125 in.; and 6, 72.50 in. In most of the subassemblies the original deviations from straightness were preserved even after irradiation and discharge, and in no case was the 80-mil tolerance exceeded.

Fig. II.E.4. Straightness Measurements of Power Subassemblies after Run 25

(3) Button-to-Button Dimensional Changes. The actual results of dimensional changes across the spacer buttons were reported previously (see Progress Report for October 1967, ANL-7391, p. 60, and for November 1967, ANL-7399, p. 55). Of a total of 51 sets of measurements, 13 indicated no change within the tolerance of ± 0.001 in., 11 indicated an increase, and 27 indicated a decrease.

(4) Effects of Heating on Dimensional Changes. To establish the effects of heating on the deformation of subassemblies, two experiments were conducted. In one, careful measurements were made across the flats of experimental subassembly X900 used in a study of sodium-immersion, water-wash effects. The subassembly was "soaked" for 15 days under 700°F sodium in the storage basket during Run 26, after which the dimensions across the buttons were remeasured. For all flats a net increase of 3 mils was noted.

In another series of tests, three hex sections containing spacer buttons were heated for 3 hr in a furnace at 900°F and then air cooled. Measurements across the buttons indicated that the average increase across the nine sets of flats was 2 mils, and in no case was a decrease noted.

The results of the heating measurements described above indicate that stress relief through heating is manifested by dimensional increases. Such results imply that the decreases in button dimensions noted for Run-25 subassemblies are very likely real, and that consideration should be given to subassembly compressional effects in formulating a physical model for understanding the Run-25 power coefficient.

(d) Operating Stability at Power. The reactor was operated for $2\frac{1}{2}$ hr at 45 MWt with no adjustment in control-rod position or flow of sodium in the secondary system, and with the control rods banked at 12.75 in. The reactor exhibited a slow power oscillation with a maximum amplitude of ± 0.6 MW and an oscillation period of 17 min. On the following day, the rods were banked at 14 in. and the test was repeated. For these conditions, the maximum amplitude was ± 0.7 MW; the oscillation period remained at 17 min.

(e) Delayed Recovery of Reactivity Effects. Several instances have been observed recently of slow recovery of the expected reactivity of the reactor. These delayed effects seem to be associated with the operating history of the reactor over the previous few days, and are called collectively "hysteresis effects." An earlier illustration of these effects was given in Progress Report for August 1967, ANL-7371, p. 10.

(f) Reactor Behavior

(1) Reactor Behavior, November 7, 1967. The reactor was operating at a power level of 45 MW with 48.8 lh excess reactivity in control rod No. 5 at 1100 hours November 7, 1967, at which time power was reduced to 22.5 MW. Upon return to 45 MW, after $7\frac{1}{2}$ hr at 22.5 MW, the excess reactivity in control rod No. 5, corrected for burnup to 1100 hours, was 55.6 lh. During continued operation at 45 MW the excess reactivity gradually returned toward the earlier value. The sequence of measurements from the console log is shown in Table II.E.2.

Since the initial discrepancy at 1902 hours was greater than 5 lh and recovery took somewhere between 5 and 13 hr, further investigations of reactivity behavior were made. The records of the previous period of operation at 22.5 MW were examined, and are shown in Table II.E.3. On this occasion also a 5-lh discrepancy occurred immediately after the return to full power, but gradually diminished over the next $8\frac{1}{2}$ hr.

TABLE II.E.2. Excess Reactivity
(November 7 and 8, 1967)

Time	MW	Excess in Control Rod 5 (lh) ^a
0800	45.0	48.2
1100	45.0	48.8
1130	22.5	-
1825	22.5	-
1902	45.0	55.6
2400	45.0	51.6
0800	45.0	49.9

^aCorrected to 1100 hours, November 7, 1967.

TABLE II.E.3. Reactivity Data
(October 22 through 24, 1967)

Time	MW	Excess in Control Rod 5 (lh) ^a
2400	45.0	48.0
0800	45.0	49.1
1000	22.5	-
2325	45.0	54.8
2400	45.0	53.3
0800	45.0	51.0
1600	45.0	51.2

^aCorrected to 0800 hours, October 23, 1967.

TABLE II.E.4. Reactivity Data,
(November 5 and 6, 1967)

Time	MW	Excess in Control Rod 5 (lh) ^a
2400	45.0	53.0
0800	45.0	52.4
0836	Scram	-
1155	45.0	53.6
1600	41.5	56.4
(= 52.4 at 45.0 MW)		

^aCorrected to 2400 hours, November 5, 1967.

The reactivity behavior before and after a scram is shown in Table II.E.4. On this occasion, there was no significant discrepancy on returning to power.

The discrepancies shown upon returning to a power level of 45 MW after 22.5-MW operation are believed to be manifestations of the same sort of hysteresis effects witnessed during descent from power. During the 22.5-MW operation, an equilibrium temperature pattern characteristic of this low power level

was established, and the 5-lh discrepancy after a rather rapid rise to 45 MW was a hysteresis effect which was gradually recovered.

During the scram of November 6, 1967, the duration of reduced-power operation was not long enough to establish a low-temperature equilibrium, so there was no significant discrepancy in excess reactivity on return to power.

(2) Reactor Behavior, November 29-December 1, 1967.

Reactor power was reduced to 22.5 MW on November 29, 1967, for a reduced-flow measurement, part of the planned sequence of measurements on Run 26C. Upon return to 45 MW, the control-rod position which had been in effect prior to the experiment was not immediately re-established. This was not regarded as unusual because of the previously observed long-term delays in reactivity adjustment. Data were taken on an hourly basis to record the return to the expected control rod position. After 1515 hours on November 29, the reactivity value decayed from 45.2 to 43.8 lh, the same as it had been at 0800 before the reduced-flow experiments.

Unrelated scrams (at 1840, November 29, and 0150, November 30) required restarting the reactor. Power coefficients were measured during the return to power. During these observations, the reactivity of the reactor appeared to increase very slightly during the power increment from 10 to 15 MW. Other tests were performed at this time, including a temporary alteration of the control-rod bank, but continued observation revealed that the increase in reactivity at 15 MWt was apparently part of a continuing long-term drift in reactivity which took place at constant power. Drift occurred between 0547 and 1000 hours on November 30. After 1000 hours, reactivity drift appeared to cease. Reactor power was carefully increased to 45 MW in 5-MW increments, and no further indication of an apparent positive power coefficient was observed.

When a power level of 45 MW was attained, the observed excess reactivity was 50.1 lh, corrected for burnup, as compared to an expected value of about 44 lh based on the experience of November 29. During continued operation at 45 MWt, the excess reactivity diminished slowly to a value of approximately 44 lh and remained constant thereafter. There was no indication of instability or difficulty in reactor control. The final power reactivity decrement was approximately the same as before the scrams.

Reactivity drifts which occur at constant power cannot properly be regarded as part of the power coefficient. Power-coefficient measurements which are taken after a prolonged operating period may include such reactivity drifts. It was this drifting which apparently caused the gain in reactivity at 15 MW and this gain is therefore not a manifestation of a positive power coefficient.

(g) Rod-drop Analysis. The feedback has been calculated for the rod-drop experiments in Run 26B, done at the same power and flow conditions as the oscillator rod experiments for the purpose of comparing results. The power levels were 22.5, 30, and 41.5 MW. The feedback is consistent with that calculated earlier for rod-drop experiments at power levels of 25, 35, and 45 MW in Run 26A, as reported in Progress Report for October 1967, ANL-7391, pp. 57-58, and for November 1967, ANL-7399, pp. 49-52. The qualitative shape of the feedback-versus-time curve is similar in all cases. As might be expected when the steady-state power coefficient is considered, the 22.5-MW feedback was slightly higher than the 25-MW feedback.

Analysis is continuing in the development of sets of feedback time constants and amplitudes which give the observed feedback in both the oscillator-rod and rod-drop experiments.

A special rod-drop experiment, in which a standard control rod was dropped from 5.85 in., was conducted in anticipation of a shortened drop time and greater accuracy of feedback determination, but the drop time was found to be longer by a factor of two (approximately 0.5 sec) because of the dashpot characteristics.

(h) Transfer-function Results, Run 26. All transfer-function data taken during Run 26 have been normalized in a manner which is consistent with the results of rod-drop experiments. The normalization involves an estimate of the prompt fuel-power-coefficient component from rod-drop analyses. Transfer-function data at the higher frequencies are adjusted according to the estimate to reflect the amplitude and time dependency of the prompt feedback.

In computing the zero-power transfer function, the following delayed-neutron parameters were used:

$$\ell^* = 0.155558 \times 10^{-6} \text{ sec}$$

$\beta_1 = 0.259 \times 10^{-3}$	$\lambda_1 = 0.0127$
$\beta_2 = 0.151 \times 10^{-2}$	$\lambda_2 = 0.0317$
$\beta_3 = 0.137 \times 10^{-2}$	$\lambda_3 = 0.115$
$\beta_4 = 0.301 \times 10^{-2}$	$\lambda_4 = 0.311$
$\beta_5 = 0.105 \times 10^{-2}$	$\lambda_5 = 1.40$
$\beta_6 = 0.242 \times 10^{-3}$	$\lambda_6 = 3.87$

Graphical summaries of feedback data for 22.5, 30, and 41.5 MW are given in Figs. II.E.5-II.E.7, respectively.

Fig. II.E.5

Rod Oscillator Data for Run 26B at 22.5 MW. Date: 10-23-67. Bank Rod Position: 11.5 in. The ordinate and abscissa have been multiplied by 10^3 .

Fig. II.E.6

Rod Oscillator Data for Run 26B at 30 MW. Date: 10-18-67. Bank Rod Position: 11.0 in. The ordinate and abscissa have been multiplied by 10^3 .

Fig. II.E.7

Rod Oscillator Data for Run 26B at 41.5 MW. Date: 10-18-67. Bank Rod Position: 11.0 in. The ordinate and abscissa have been multiplied by 10^3 .

(ii) Thermal and Hydraulic Analyses. The results of calculations to determine the temperature gradient across a driver subassembly for constant axial coolant ΔT conditions indicate that the radial ΔT across the hex can is relatively constant as power and flow are reduced proportionately, but the ΔT between edge cooling channels and center channels

decreases. These calculations were made in an effort to find a means of reducing or eliminating the rubbing in the oscillator rod and to gain more information about the bowing phenomenon.

As part of a study to improve the operation of the oscillator rod at power by reducing temperature gradients across the oscillator position, an analysis was made of the temperatures in driver-fuel subassemblies surrounding the two possible types of control-rod locations. A location at the corner of Row 5 (an odd-numbered control-rod location) and one on a Row-5 flat (an even-numbered control-rod location) have very similar conditions. In either location, temperature gradients may exist across a control-rod location at a level corresponding to the top of the driver fuel because of different coolant temperatures in adjacent driver-fuel subassemblies. Mixing of the coolant within the subassembly eliminates this gradient about 6 in. above the fuel. Eliminating the gradient at the top of the fuel by revising the flows in the adjacent driver-fuel subassemblies would therefore create a gradient across the oscillator-rod location in the upper blanket region.

A preliminary analysis of the results of a one-dimensional problem run at Oak Ridge National Laboratory (ORNL) on the gamma-energy distribution in the EBR-II core, reflector, and blanket for the case with stainless steel reflector subassemblies in Rows 7 and 8 indicates that the results are in good agreement with the experimental results obtained by ORNL in July. The gamma heat rate at the axial centerline of Rows 7 and 8 is about $1/2$ to $1/3$ of that previously used in the temperature-distribution calculations in those rows.

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

Last Reported: ANL-7399, pp. 56-61 (Nov 1967).

(i) Investigation of EBR-II Power Coefficient

(a) Linear Feedback. The fuel-temperature feedback has been determined experimentally by measuring the change in reactivity between 41.5 MW, full flow, and 22.5 MW, 54% flow. The ratio of power to flow for these cases is constant, so that one can assume that the only temperature differences are for the fuel. In Table II.E.5 are presented the measured values for the experiments performed during Runs 25 and 26.

TABLE II.E.5. Measured Changes in Reactivity between 41.5 MW, Full Flow, and 22.5 MW, 54% Flow

1. 7.8 Ih (Run 25)	4. 11.6 Ih (Run 26B--end)
2. 8.8 Ih (Run 26A--start)	5. 9.5 Ih (Run 26C--start)
3. 11.3 Ih (Run 26B--start)	6. 14.8 Ih (Run 26C--end)

Of the data in Table II.E.5, values for experiments 3 and 5 should have a large uncertainty factor. For experiment 3, the power was actually 24.2 MW rather than 22.5 MW, so that an estimated correction to the experimental data was necessary. In experiment 5 the data were taken in the descending power mode, so that "hysteresis effects" are involved.

Although the data are probably only accurate to several inhours, the trend of the measurements seems to indicate an increasing fuel-temperature feedback during Run 26. A possible explanation for such an effect would be that the fuel thermal conductivity and the thermal conductivity of the fuel-clad bonding decrease with burnup. For the same power, then, fuel temperatures and associated fuel expansion would be greater at the end of the run than at the beginning.

By use of computed values of the average fuel temperature for various power and flow conditions, the fuel-expansion coefficient (Ih/MW) can be obtained from the data in Table II.E.5. Further, assuming that the sodium temperature coefficient and, in turn, the sodium power coefficient of 0.85 Ih/MW is the same for all cases, the results for the linear reactivity feedback for three cases are those listed in Table II.E.6.

TABLE II.E.6. Linear Reactivity Feedback

26A (start)	1.8 Ih/MW
26B (end)	2.0 Ih/MW
26C (average) ^a	2.1 Ih/MW

^aFor this case, a weighted average, 13.5 Ih, of the two values for Run 26C in Table II.E.5 was used.

(b) Nonlinear Feedback. Subtraction of the values for the linear feedback as given in Table II.E.6 from the measured values of the total power coefficient (normalized for the control-rod bank effect) gives the nonlinear component of reactivity (see Fig. II.E.8). The curve labeled 26C (average) was obtained

Fig. II.E.8
The Nonlinear "Bowing" Component of Total Reactivity Feedback during Run 26 vs. Reactor Power

from the total power coefficient for 26B (end), since the 26C (start) power coefficient measurements did not follow the expected pattern. It appears, therefore, that there has also been an increase in the nonlinear component of reactivity during Run 26. A possible explanation for this behavior could be that subassembly clearances may have been increased by repeated "bowing" cycles. This less-tight core configuration would then result in an increase in the nonlinear feedback.

c. Reactor System Testing, Surveillance, and Evaluation
(B. C. Cerutti)

Last Reported: ANL-7399, p. 62 (Nov 1967).

(i) Guide Bearings on Control-rod Lower Adapters. An inspection of the beryllium copper guide bearings on the lower adapters from control rods L-435 and L-441 was conducted in the FCF air cell. The lower adapters were visually inspected and photographed through a periscope. The results verified the conclusions from the previous inspection of control rod L-434 (3000-MWd exposure). Control rod L-441 (2910 MWd) was corroded slightly more than L-434 had been. Control rod L-435 (4520 MWd) was corroded quite badly and was significantly worse than the others in the upper bearing area. However, L-441 also exhibited more bearing corrosion at the upper end than was found in L-434. The upper bearings on a control-rod lower adapter are subjected to less pressure drop and less flow velocity than the lower bearings, but are in an area of higher neutron flux. The corrosion appears to be approximately proportional to total hours of exposure to sodium at full reactor power, and possibly also to be a function of the total exposure to neutron flux.

(ii) Core Gripper. A test was conducted in the FCF air cell on the highly radioactive gripper mechanism from the core fuel-handling gripper assembly which was removed because of a malfunction (see Progress Report for August 1967, ANL-7371, p. 19). The purpose of the test was to determine the maximum size of fuel-subassembly upper adapter which could have been engaged properly by the gripper when it was in the primary tank. The gripper assembly, including approximately 4 ft of drive shaft, was first cleaned so that the jaws were operable. A carrier loop and a jaw actuator loop were then welded to the upper tubing extensions to permit manipulator operation of the gripper in the vertical position.

Three specially sized upper adapters were fabricated to determine the maximum size of upper adapter which would still allow proper operation of the core gripper. The length of the cylindrical portion of the knob on the B-341 upper adapter was determined as accurately as possible, using an optical comparator, to be 0.155 ± 0.0005 in. The specially sized adapters included a slightly undersized adapter (0.154 in.), a very slightly oversized adapter (0.156 in.), and a more oversized adapter (0.158 in.)

as compared to the upper adapter for subassembly B-341. Each of these adapters was then physically inserted in the gripper to determine if the jaws would operate properly. The results are as follows:

(1) The gripper jaws operated properly on the 0.154-in. adapter with very slight resistance in the open and close directions.

(2) The gripper jaws were difficult to close on the 0.156-in. adapter if the jaw actuator was moved slowly as occurs in the primary tank. The jaws were difficult to open completely without applying impact force to the jaw actuator.

(3) The gripper jaws would not close on the 0.158-in. adapter. If the gripper was slightly misaligned vertically, one jaw would partially engage the adapter, but the jaws could not be closed with reasonable force.

It should be noted that continued operation of the gripper jaws on the 0.156-in. adapter resulted in easier jaw operation. This was attributed to rounding of the lower edge of the adapter knob which decreased the effective size of the adapter. This was confirmed visually outside the cell after the test was completed.

The test shows that while the gripper will operate normally on a 0.154-in. upper adapter, it will not operate properly on a 0.156-in. adapter (grripper did not operate properly on B-341 which was 0.155 ± 0.0005 in.), and will not operate on an adapter nominally 0.003 in. bigger than the B-341 adapter. The maximum drawing tolerance on the upper adapter is 0.155 in.

After the above test, the gripper was disassembled for inspection. Although the radiation level of the gripper components does not permit direct inspection, remote inspection has not disclosed any obvious wear or damage to the gripper components.

(iii) Startup Feedwater Pump. Fabrication of the new pump at the vendor's plant is still progressing satisfactorily. Delivery is expected in late January 1968. Designing of the piping changes needed to accommodate the new pump is in progress.

(iv) Instrument Air Blower for FERD Loop. An evaluation was made of the problem concerning the instrument-cooling air blower for the FERD loop. The presently installed positive-displacement Roots-Connersville blower causes some oil entrainment in the instrument-cooling air supply. Deposition of oil on the four preamplifiers for the FERD loop delayed-neutron detectors reduces tube life and necessitates costly and

time-consuming repairs. Access to the preamplifiers is complicated by the need for shutdown of the system, time for decay, radioactive sodium, and shield removal. The manufacturer's data indicate that the installed blower supplies an estimated 25 cfm of cooling air at approximately 1/3 psi or 9.2 in. of water-pressure drop in the present air-supply pipe.

A proposal was made to replace the existing blower with an oil-free centrifugal type along with a shorter run of larger-diameter duct to provide an increased supply of oil-free cooling air at a lower pressure drop. The proposed blower would supply 62 cfm of air through the new duct with a pressure drop of about 1 in. of water pressure.

(v) Leak-rate Testing of Containment Building. A study is being made to establish allowable leakage rates for all reactor-building penetrations. According to the EBR-II Hazards Summary Report Addendum (ANL-5719), the maximum acceptable leakage rate for the reactor building is 1000 ft³/day at 32°F and 36.2 psia. Drawings of all mechanical and electrical penetrations of the building were reviewed and the total peripheral length for all penetrations was determined to be 2593 in. Dividing the allowable total leakage by the total peripheral length of all building penetrations gives an allowable leakage rate of 0.386 ft³/in.-day. Multiplying this number by the peripheral length of each penetration establishes the allowable leakage rate for each reactor building penetration.

(vi) Turbine Generator. The vapor extractor of the turbine generator was inspected for the first time. All bearings were replaced. Other parts were in excellent condition.

The turbine-driven auxiliary oil pump was inspected and found to be in the same condition as at the time of the general overhaul in 1967. The effects of corrosion observed at that time apparently existed before initial startup.

Delivery of the new motor-driven auxiliary oil pump is expected soon.

(vii) Engineering Consultation. A survey of the latest information concerning irradiation effects on stainless steel is being made to determine when the safety-rod thimbles should be replaced.

d. Higher Power Operation (R. E. Rice)

Last Reported: ANL-7399, pp. 62-63 (Nov 1967).

The proposal for controlled flow experiments in Rows 4, 5, and 6 was revised to incorporate all comments of the review committee and was sent to the Idaho Division Director for review.

Revision of the proposal for 50 MWt operation continued. Temperature distributions within the core were determined for 50 MWt operation with depleted uranium subassemblies returned to Rows 7 and 8. The calculations were carried out for two cases: one assuming experimental subassemblies in Row 4, and the other assuming them in Row 5.

The temperature calculations for the proposal were revised to include new flow rates. The new flow rates resulted from inclusion of (a) calibration data for driver subassemblies with triflute axial end blankets, and (b) changed flow distributions due to lower flow requirements in experimental subassemblies and (where applicable) the stainless steel inner blanket. The use of the new flow rates results in significantly lower calculated temperatures in driver fuel and cladding in the present core because of the higher flow rates in driver subassemblies.

The study of the relocation of experimental subassemblies for operation of the reactor at a power level of 50 MWt with a stainless steel reflector has been augmented to include operation with a depleted uranium blanket (including Rows 7 and 8). The conclusions reached from this added study were that the fueled experiments located in Row 4 must be moved to Row 5 even though they will then be operated at the minimum of the acceptable flux limits and that fueled experiments in other rows must also be relocated. The lower flux value is a consequence of a steeper radial flux gradient and a lower specific power brought about by the use of the depleted uranium blanket.

e. Fuel Swelling and Driver Surveillance (F. G. Foote, K. F. Smith, and S. T. Zegler)

(i) Mark IA--Investigation of Anomalous Fuel Swelling

Last Reported: ANL-7399, pp. 63-65 (Nov 1967).

(a) Extended Burnup of MC-S-type Fuel Pins. Subassemblies C-291, C-292, and C-296 of matched Set 1 reached 98, 99, and 101% of target burnup, respectively, at the end of Reactor Run 26. Subassembly C-291 contains only MC-S-type fuel and will be held in the fuel-transfer storage basket pending surveillance of identical material in subassemblies C-292 and C-296.

(b) Effect of the Change to Mark-IA Design upon Fuel Swelling. Subassemblies C-297 and C-298 reached 97% of the target burnup at the end of Reactor Run 26. After cooling, the fuel elements in these subassemblies will be surveyed for fuel-pin swelling.

(c) Effect of Enrichment Increase for 48% (Mark I) to 52% (Mark IA). Subassembly C-283 reached target burnup about midway through Reactor Run 26 and was removed from the core. Subassembly C-293 reached 96% of target burnup at the end of Reactor Run 26 and will also be removed from the core. No surveillance results are yet available.

(d) Effect of Position and Orientation in the Casting Bundle upon Fuel Swelling. Subassembly C-2035 reached 97% of the target burnup at the end of Reactor Run 26 and will be removed from the core.

(e) Effect of Trace-element Impurities upon Fuel Swelling. Subassemblies C-2111 and C-2113 were inserted in Reactor Grid Positions 1-A-1 and 4-A-3, respectively, about midway through Reactor Run 26. Shortly after startup, when the reactor was at 50% of its normal operating power, the fission-gas monitor signal increased, indicating a fission-product-gas release (see Sect. II.E.1.a). Both subassemblies were removed from the core and the signal remained at background level after restart, which indicated a leaking fuel element in one of the subassemblies. Subassembly C-2113 was subsequently disassembled in FCF; one fuel element was identified as a possible leaker. This fuel element is being examined in detail for jacket defects. If no leak can be identified, subassembly C-2111 will also be disassembled and examined.

In the October 1967 report (ANL-7391), p. 69), the difficulty of adding silicon to EUF-type melts was discussed. The difficulty now

TABLE II.E.7. Carbon and Silicon Analyses of Spiked Melts

Casting Batch	Trace-element Impurities (ppm)		
	Carbon	Silicon	Source
0450	307	210	Idaho Original
0450	317	374	Illinois
0450	-	358	Idaho Recheck
0452	675	140	Idaho Original
0452	653	511	Illinois
0452	-	485	Idaho Recheck

appears to have been in the sampling and chemical analysis rather than with the alloying. Cast pins from melts 0450SR and 0452SR were shipped to Illinois, sampled, and analyzed for silicon and carbon. Samples prepared at Illinois were returned to Idaho for silicon analysis. The results are tabulated in Table II.E.7.

The carbon analyses agree quite well, as do the Illinois and Idaho recheck for silicon. It now appears that these melts do satisfy the composition requirements of the experiment, although the carbon and silicon in casting batch 0452SR somewhat exceed the specifications. Chemical analyses

of other melts in this series are being obtained. Subassemblies C-2110 and C-2112 (the other two in this series) will be inserted in EBR-II midway through Reactor Run 27.

To establish an upper limit for the silicon content, it has been proposed to prepare alloys containing 500, 650, 800, and 1000 ppm silicon by weight. None of these alloys has been prepared yet although existing casting batches prepared during normal operation of FCF correspond to the lower half of the range. These batches include silicon contents up to 655 ppm.

(f) Effect of Pressure on Swelling of U-5 w/o Fs Alloy.

The specimens of U-5 w/o Fs fuel from Capsule P-1 have been given additional density determinations to assess the amount of scatter existing in the data given in the November monthly report (see ANL-7399, p. 63). The calculated values of $\Delta V/V$ as a function of in-reactor ambient pressure are given in Fig. II.E.9. Although the scatter of data points is considerable because of poor specimen condition, the trend to lower swelling with higher pressure is still quite apparent.

Fig. II.E.9. Effect of Pressure on Swelling--Capsule 1

Burnup data for Capsule P-1 have been revised from ~1.5 a/o to 3.2 to 3.5 a/o, on the basis of recently received burnup analyses. These higher figures are much more consistent with the physical appearance of the irradiated fuel pins.

Recent metallographic evidence has now shown that the maximum temperature of irradiation of the P-1 specimens was above

650°C instead of about 595°C, as had been indicated by thermocouples adjacent to the specimens. The higher apparent fuel temperature is believed to be mainly due to the loss in fuel thermal conductivity resulting from extensive void formation.

The postirradiation examination of Capsule P-2 has been completed on the jacketed specimens. One specimen, with an intermediate filling pressure, was ruptured. All others showed insignificant changes in weight, density, length, and diameter. Residual pressure will be measured next. Radiography of the jacketed specimens reveals the appearance of an inhibitory effect of pressure on swelling, assuming that no leakage has occurred. Burnup levels have been revised to 1.8 a/o for these pins on the basis of recently received burnup analyses for Capsule P-1.

Capsule P-3 was removed from the reactor on November 9 at 158 MWd (now 1.3 a/o burnup). Examination of a neutron radiograph of the entire capsule indicated that a spectrum of swelling values as a function of filling pressure was measurable even at this burnup level. The capsule is awaiting availability of postirradiation facilities for further examination.

(g) Effect of Heat Treatment on Swelling of U-5 w/o Fs Alloy. Heat treatment and irradiation parameters for H-1, H-2, and H-3 are given in Table II.E.8; the series of capsules in process to show the effect of various heat treatments on fuel swelling. These contain fuel pins for exposure without containing jackets.

TABLE II.E.8. Schedule of Heat Treatments for U-5 w/o Fs Specimens To Be Irradiated in Capsules in CP-5

Capsule Number	Position in Capsule	Cast Pin Number	Melt Number	Heat Treatment	Ambient Pressure	Target Fuel Temp (°C)
H-1	1	56-2	409	As-cast	~Atmospheric	~600
	2	-1	409	1 hr @ 500°C, A.C. ^a		
	3	-3	409	2 hr @ 980°C, W.Q. ^b		
	4	-5	409	2 hr @ 980°C, W.Q., 1 hr @ 500°C, A.C.		
	5	-6	409	16 hr @ 740°C, W.Q.		
	6	-8	409	16 hr @ 740°C, W.Q., 1 hr @ 500°C, A.C.		
H-2	1	13-1	0450SR	As-cast	~Atmospheric	~600
	2	-2	0450SR	1 hr @ 500°C, A.C.		
	3	-3	0450SR	2 hr @ 900°C, W.Q.		
	4	-5	0450SR	2 hr @ 900°C, W.Q., 1 hr @ 500°C, A.C.		
	5	-6	0450SR	16 hr @ 740°C, W.Q.		
	6	-8	0450SR	16 hr @ 740°C, W.Q., 1 hr @ 500°C, A.C.		
H-3	1	54-1	409	1 hr @ 500°C, A.C.	~Atmospheric	~600
	2	-12	409	1 hr @ 500°C, A.C.		
	3	47-1	0450SR	1 hr @ 500°C, A.C.		
	4	-12	0450SR	1 hr @ 500°C, A.C.		
	5	-2	0450SR	2 hr @ 900°C, W.Q., 1 hr @ 500°C, A.C.		
	6	-11	0450SR	2 hr @ 900°C, W.Q., 1 hr @ 500°C, A.C.		

^aA.C. = air cooled.

^bW.Q. = water quenched.

Capsule H-1 has accumulated 1.1 a/o burnup. Upon removal, neutron radiography revealed some incipient swelling, but not enough to assure definitive measurement. The capsule was, therefore, returned to the reactor for additional exposure.

The difficulty with apparently low silicon in casting batch 0450SR, mentioned above in Item (e), has now been resolved, and specimens cut from pins from this melt will be used in Capsules H-2 and H-3.

Capsule H-2 is scheduled for placement in the reactor on December 18, 1967. Capsule H-3 is in heat treatment.

(h) Hot Laboratory Examination of Irradiated Fuel. The results of chemical analyses for certain impurities in EBR-II, Mark-IA fuel alloys were given in Table II.E.9 in the October report (ANL-7391, p. 72 et seq.). Some changes and additions are given in Table II.E.9, with previously reported values in parentheses.

TABLE II.E.9. Trace-element Analyses for EBR-II, Mark-IA Fuel

Casting Batch and Pin No.	Melt Refining No.	Subassembly and Element No.	Trace Elements (ppm by weight)				Analyst
			Carbon	Silicon	Aluminum	Iron	
4003	-	24	Average	215(208)			
4022	-	C-30	Average	156(144)			
4023	-	33	Average	121(132)			
4027	-	46	Average	174(190)			
4032	-	C-36	Average	262(238)	189(151)		
SL-33	-	-	Cast Pin	166	71	63	27
SL-35	-	-	B-343				
SL-36	-	-	B-343				ID
SL-36	-	-	Cast Pin	120	70	51	167
SL-38	-	-	B-343				ID
SL-39	-	-	B-342				ID
SL-39	-	-	B-343				ID
SL-39	-	-	Cast Pin	145	76	64	183
SL-40	-	-	B-343				ID
SL-41	-	-	Cast Pin	144	104	56	201
SL-42	41	-	C-213	78	244	193	ID
SL-42	-	-	Cast Pin	215	109	95	195
SL-42	-	-	Average	215	94	170	194

(i) Characterization of Unirradiated Fuel Pins. Pins from injection-casting batches 0450SR and 0452SR (containing 365 and 500 ppm silicon, respectively) are being examined metallographically to determine the effect of silicon upon the microstructure.

(j) Analysis of EBR-II Fuel-pin Swelling. A figure of merit given the ratio of swelling to average burnup, based only on swelling

data for average burnups exceeding 0.975 a/o, is plotted against the silicon content (see Table II.E.9) in Fig. II.E.10. There is now a very clear separation between high-swelling low-silicon material and low-swelling high-silicon material. Casting batch 4049 (MR-C-45) contains 486 ppm silicon, the highest yet reported, and is a low-swelling material.

Fig. II.E.10

Unit Swelling vs. Silicon Content
for EBR-II, Mark-IA Fuel

(k) Compatibility of U-5 Fs Alloy (Containing Intentionally Added Impurities) with Type 304 Stainless Steel. Shown in Table II.E.10 are fuel and jacket penetrations measured with diffusion couples comprised of Type 304 stainless steel and U-5 w/o Fs alloy containing 790 ppm (by weight) of Si and 1870 ppm of Al (Melt R-410). Included for comparison are similar data obtained previously for U-5 w/o Fs alloy containing 77 ppm of Si and 20 ppm of Al (Melt R-409). Both fissium alloys exhibit essentially the same fuel and jacket penetrations at annealing times up to 1650 hr at 650°C.

TABLE II.E.10. Maximum Fuel and Jacket Penetrations (μ) for
U-5 w/o Fs-type 304 Stainless Steel at 650°C

Annealing Time (hr)	U-5 w/o Fs (R-410)- type 304 SS			U-5 w/o Fs (R-409)- type 304 SS		
	Specimen Number	Fuel	Jacket	Specimen Number	Fuel	Jacket
168	7	160	7			
1000	8	408	12	27	340	12
1650	8E	450	16	a	464	16

^aCalculated from data up to 1000 hr reported in ANL-7382.

f. Mark-II Driver Fuel Element Development

Last Reported: ANL-7399, pp. 65-66 (Nov 1967).

(i) Irradiation of Encapsulated Mark-II Elements in EBR-II
(M. A. Pugacz)

When the capsules are removed from an experimental EBR-II subassembly at regular intervals of burnup, the sodium bond in the capsule freezes from the top, causing shrinkage voids to form in the solidifying sodium. If these voids are not removed before subsequent reinsertion of the capsule in the reactor, they could cause overheating of the experimental fuel element contained in the capsule. Although tests on actual fuel elements have shown shrinkage voids when the sodium is solid, X-ray inspection showed that these voids disappear when the sodium is remelted.

At the present time, there is no equipment available in the Fuel Cycle Facility to inspect the quality of the sodium bond in irradiation capsules. Although such equipment is being built, it will not be available for several months. In order to avoid delays, and the risk of reinserting unbonded capsules in EBR-II, it has been decided to forego the originally scheduled removal of capsules at the 0.5 a/o burnup level. The first capsule removal will be made at the end of the second reactor cycle, when burnup of the experimental elements will be in the range of 1.1 to 1.3 a/o.

(ii) Long-term Compatibility of U-5 w/o Fs Alloy and
Type 304 Stainless Steel (S. T. Zegler)

Fuel and jacket penetrations of 44 and 2 μ , respectively, have been measured for a U-5 w/o Fs (Melt R-409)-type 304 stainless diffusion couple (Specimen 25-A) that was annealed for 1000 hr at 550°C. Both values are closely consistent with those calculated from extrapolation of data for times up to 840 hr. The significantly lower fuel penetration, 22 μ , previously reported for Specimen 25 (see Progress Report for September 1967, ANL-7382, p. 31) is, on the basis of findings in optical metallographic studies, attributed to fracturing of interdiffusion zones either during or after annealing.

g. Equipment--Fuel Related (E. Hutter)

(i) Improved Gripper and Holddown Force-limit Device

Last Reported: ANL-7391, p. 81 (Oct 1967).

Assembly and test of the prototype electronic control system continued. The driver motor was connected to its gear box and

its speed controller; the combination operated satisfactorily after slight malfunctions of the controller were corrected. The motor is being connected to the test rig.

The electronic control system was checked using the sensor potentiometers of the test rig. Omitted connections have been added, adjustments have been made, and the pull-force channel has been energized. Several minor design changes were made to eliminate noise and spurious oscillations; some noise problems remain, but they can be connected with no major design change. After the electronic system tests are completed, the system will be connected to the motor controller and other components for operational tests of the entire mechanical test rig.

(ii) New Control-rod Subassemblies (E. Hutter)

Last Reported: ANL-7399, p. 66 (Nov 1967).

(a) Control Rod of Greater Worth. Studies to develop control rods of greater worth are continuing. Work concentrated on further developing the seven-tube bundle array for an all-absorber-type control rod (reported as Type C in Progress Report for October 1967, ANL-7391, p. 82).

A detailed design layout has been prepared, and calculations have been made to determine the expected volume fractions, quantity of helium-gas generation, gas pressure, and tube-wall stress. The calculations show that the design is capable of operating in EBR-II to 2 a/o burnup of the B₄C.

Even with the most pessimistic condition of 100% gas release from the B₄C (although 100% release is physically not possible), the stresses in the walls of the tubes will still be tolerable with an adequate margin of safety. With 20% gas release, the safety factor increases five times. The design assumption regarding 20% gas release is very conservative; the designs for the B₄C control rods for the Fermi and SEFOR reactors are based on an assumed 10% gas release.

(iii) Oscillator Rod--Mark II

Last Reported: ANL-7391, pp. 82-83 (Oct 1967).

Operation of the present oscillator rods and analysis of reactor data show that the usefulness of the rods would be enhanced if bowing of the rod assembly and internal capsules were eliminated. Bowing variations with changes of reactor power or temperature cause minor changes in the amplitude of reactivity change. Also, the oscillator rod would be more useful if it could be rotated at very slow speeds when the reactor is operating at full power; at speeds below 0.3 rps, the different temperatures of

Fig. II.E.11. General Arrangement of Mark-II B Oscillator-rod Assembly (without lower bearing)

the adjacent subassemblies apparently heat the present oscillator rod unevenly and cause it to rub slightly against the enclosing thimble.

A new oscillator-rod design, the Mark IIB (see Fig. II.E.11), employs tangential jets on the outer surface of the rod to cause the sodium coolant to flow helically between the rod and its guide thimble, thus reducing circumferential temperature differences in the rod. The rod capsules that contain boron carbide and aluminum oxide are wrapped with helical wires and enclosed within shroud tubes, so that the coolant will flow in a spiral and equalize the horizontal temperature differences across the capsules.

About 7 gpm of sodium flow up through grooves in the bottom oscillator bearing (not shown), around the base collar, and up between the hexagonal thimble and the outer tube of the oscillator rod. Another 17 gpm flow up through paths inside the tube on which the bearing is mounted and through the oscillator-rod lower end fitting to the orifice and guide plate; at this plate, the flow divides into three paths: (a) 8.5 gpm spiral up around the boron carbide capsule within its shroud tube, (b) 1.3 gpm spiral up between the aluminum oxide capsule and its shroud tube, and (c) 7.2 gpm flow through the stainless steel structure within the oscillator-rod assembly. After the sodium has cooled the capsules, it leaves the shroud tubes through holes near the top of the shrouds and rejoins the sodium that has flowed within the 1.75-in.-dia outer tube; the sodium then passes up through cutouts in the bottom end fitting of the expansion chamber and rises between the expansion chamber and the outer tube. The sodium leaves the outer tube through tangential horizontal jets. These jets, at

eight elevations through the outer tube, swirl the sodium between the outer tube and the surrounding hexagonal thimble. (Four jets are spaced at 5-in. intervals within the oscillator-rod region; the other four jets are at $5\frac{5}{8}$ -in. intervals in the expansion-chamber region.)

The steel in the top adapter and the top end fitting of the expansion chamber provides 4 in. of radiation shielding between the core and the top of the reactor. Except during oscillator tests, a dummy rod that provides greater shielding is installed in the reactor.

To keep thermal stresses in structures above the oscillator rod within tolerable limits, the temperature of the mixed sodium leaving the top of the oscillator rod is kept within 100°F of that from adjacent subassemblies. This is accomplished by having a total of about 24 gpm of sodium flow through the oscillator rod. Tests will determine the precise flow and distribution needed to equalize the temperature gradients in the various flow paths through the rod assembly.

(iv) Engineering Consultation

Last Reported: ANL-7399, pp. 66-67 (Nov 1967).

(a) Power Coefficient. The BOW code has been completed and is being used to explain EBR-II power-coefficient changes observed when stainless steel reflector subassemblies were substituted for depleted-uranium blanket subassemblies.

The code is being used to evaluate various schemes for improving the EBR-II power coefficients both in the near future and from a long-range viewpoint. Some of the ideas being studied are: (1) inserting additional rows of reflector subassemblies in place of blanket subassemblies, (2) reducing clearance between core region subassemblies by using higher spacer dimples on the hexagonal tubes, (3) increasing clearance between reflector subassemblies by eliminating spacer dimples from the hexagonal tubes, (4) reducing the stiffness of reflector subassemblies, (5) modifying subassembly flow orifices to obtain more favorable temperature distribution, and (6) reducing the thickness of the upper ends of reflector blanket subassemblies to minimize interference of adjacent subassemblies.

h. Equipment--Reactor and Primary Coolant System (B. C. Cerutti)

Last Reported: ANL-7399, pp. 67-69 (Nov 1967).

(i) Design for Sodium Drain Tank. Installation drawings for the primary sodium storage facility are being prepared. The investigation of the use of an EM pump as a backup procedure for draining the primary tank is being continued.

(ii) Primary Cold Trap. Prefabrication of pipe welds for the cold-trap installation is continuing. The schedule for installation of the new cold trap is dependent upon satisfactory checkout of the new sampling system in the FERD loop.

(iii) Primary Tank Pressure Relief. Drafting work for the installation of the primary tank pressure-vacuum relief system is nearing completion.

(iv) Sodium Sampling Pump and Filter. The pump and filter assembly and the water test rig are being readied for tests.

(v) General Improvements

(a) FERD Loop Plugging Meter and Sampler. Following operational testing of the plugging meter in the FERD loop, the facility is in routine use. Samples from the FERD loop sampling system are being examined for quality. The design modification to provide the capability of obtaining samples in glass or quartz beakers is in progress.

(b) Control-rod-drive Lifting Platform. An improved mechanical interlock was installed on the control-rod-drive lifting platform.

(c) Surge Tank in Purification System for Primary Sodium. The engineering package for modification of the surge tank in the purification system for primary sodium was reviewed and approved by the Plant Modification Committee.

i. Secondary Sodium and Power Systems (B. C. Cerutti)

Last Reported: ANL-7399, p. 69 (Nov 1967)..

(i) Steam-drum-level Instrumentation. A new differential pressure cell was installed on the steam drum in parallel with the Eye-Hye level indicator in order to provide low- and high-level alarms that are independent of the level control system.

(ii) Sodium Boiler. Bids for new safety valves for the steam drum have been received and are being evaluated.

(iii) Installation of the New Secondary EM Pump. Installation drawings for the additional footings needed for the new secondary EM pump have been completed. An installation procedure for the new pump is being prepared.

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

(i) Mark-II Core Development

Last Reported: ANL-7399, p. 70 (Nov 1967).

Additional values for total dose have been compiled and evaluated. These computed total dose values for the reactor cover when the Mark-IIA (26-in.) fuel element is used are equal to or lower than those computed originally. These data verify the conservatism of the original computations.

k. Instrumented Subassembly (E. Hutter)

Last Reported: ANL-7399, pp. 70-72 (Nov 1967).

(i) Design

(a) Subassembly Attachment and Lead Severance. The top-end-fixture test gripper has been fabricated. This device, part of the modified top-end-fixture coupling (see Progress Report for October 1967, ANL-7391, pp. 87-88), will support the weight of the instrumented subassembly. Tension tests were conducted in which the gripper was pulled to 3000 lb in 500-lb increments. No detrimental effects were observed, and elastic behavior was indicated. A series of uncoupling tests, in both room-temperature air and 700°F sodium, is planned.

(b) Drywell Concept. A sodium-free annular space, used to extend conventional leads from the sheathed leads in this region, is formed by the 27-ft extension tube (of 2.5-in. OD and 0.120-in. wall), a near-elliptical inner tube, and a bulkhead through which the sheathed leads enter. This arrangement will permit the use of shorter sheathed leads (7 instead of 26 ft) to simplify handling during assembly.

The inner tube (2.156 x 1.593 x 0.049-in.-wall) is shaped to allow passage of the lead-cutoff tool before the subassembly is separated from the extension tube. The specially shaped tubing is obtained by cold drawing; for this purpose a mandrel has been fabricated, and several sections of 5 to 8 ft have been drawn with good results. A 30-ft section will be drawn for further tests.

(c) Dummy Plug Assembly. In the proposed arrangement in EBR-II, the extension tube and gripper mechanism of an instrumented subassembly occupy the space where a control-rod drive mechanism normally would be located. If no experimental subassembly is ready to be installed in the thimble at any time after a control rod and drive mechanism have been removed to make room for it, a "dummy" system will be installed

in the vacated thimble. In one design, this dummy system would be a unit plug assembly that extends from the control-rod lifting platform, through the rotating plug and reactor-vessel cover, and into the core region. It would have the disadvantage of having a large mass of irradiated metal that would to be handled outside the reactor when the plug is withdrawn and stored; the disadvantage could be eliminated by sectioning the dummy plug so that only the upper (unirradiated) section would be withdrawn while the lower section would be removed from the core by the existing fuel-handling equipment and stored in the primary tank. This arrangement has been investigated.

If the upper and lower dummy-plug sections are segregated at the level of the tops of regular fuel subassemblies, the normal-fuel handling procedures could be used. The upper section of the dummy system would consist of a solid rod with a force-sensing mechanism at the lifting platform and a labyrinth seal at the penetration through the reactor-vessel cover. The bottom end of the rod, which would be similar to the holddown fingers in the vessel cover, would clear the top of the instrumented subassembly. The lower section, which would allow for normal coolant passage, uses the existing inlet portion of the control-rod assembly that has inherent hydraulic-holddown capability so that the dummy subassembly cannot be lifted hydraulically.

The dummy subassembly would fit in the control-rod thimble, the storage basket, and the fuel-handling mechanism of the reactor.

1. Packaged Loop (B. C. Cerutti)

Last Reported: ANL-7399, p. 72 (Nov 1967).

Preliminary studies concerning the feasibility of the packaged loop are in progress. An estimate of the sodium activity expected in the loop has been made, and shielding studies based on this estimated activity are in progress. Depleted uranium appears to be the shielding material required because of sodium activity levels and space limitations. In the region where the thimble penetrates the control drive platform, lack of space may require the use of shadow shielding at the distance from the loop components.

A preliminary estimate of weights involved, mostly shielding, shows that there should be no structural problem with the plug bearings or other load-supporting components.

m. Process Chemistry (D. W. Cissel)

Last Reported: ANL-7399, pp. 72-76 (Nov 1967).

(i) Sodium Coolant Quality and Control

(a) Primary Sodium

(1) Sampling. The primary sampling system operated satisfactorily until the failure of a heater of the plugging-meter economizer made shutdown necessary. The heater was repaired after decay of sodium activity to a level permitting work in the purification vault, and the sampling system returned to service.

(2) Studies of Cesium Segregation. Attempts to gain qualitative or semiquantitative information on the dynamics of cesium segregation have been unsuccessful. The lack of precision gamma-scanning equipment has hampered this work. Since a point of diminishing return seems to have been reached in the cesium-segregation studies, they will be discontinued. Future work will concentrate upon sampling and analytical methods which circumvent the impurity-segregation phenomenon by analysis of the total sample.

(3) Carbon Analysis. Low carbon concentrations obtained in analyses of both primary and secondary sodium aroused the suspicion that some part of the analytical procedure might be causing losses of carbon. To check this, a series of six recovery tests was performed in which the reaction vessel and sodium slug taken from the secondary system were spiked with successively smaller amounts of carbon. The recoveries noted, and the amounts of carbon presumed to be in the sodium prior to spiking, determined by subtractions of the amount added, are shown in Table II.E.11.

TABLE II.E.11. Carbon Analyses

Run No.	Weight of Secondary Sodium Sample (g)	Amount C Added (μ g)	Amount C Found (μ g)	Difference (μ g)	C in Na (ppm)
1	0.71	163	163	0	0
2	0.77	122	127	5	7
3	0.69	81	90	9	13
4 Repeat	0.68	81	89	8	13
5	0.70	41	47	6	9
6	0.72	16	19	3	4

These data prove conclusively that the analytical method is accurate after the taking of aliquots from the bulk sample. The possibility remains that carbon segregates near the walls of the sample vessel, thus giving rise to low concentration in the bulk of the sample. This point will be resolved if a satisfactory sampling method can be developed for taking uncontaminated samples of suitable size for analysis of the total sample.

(b) Expanded Sampling Capabilities for Primary and Secondary Sodium

(1) Primary Sodium. Design work continued on modifications to the primary sampling system (see Progress Report for September 1967, ANL-7382, p. 41).

A cup-and-spigot type of sampler is being designed for installation in the FERD loop sampling station. This sampler will permit taking samples in 10-ml Pyrex, quartz, or metal beakers, in addition to the flow-through freeze line samples now taken. When this sampler has been fabricated and installed, the capabilities of the FERD loop sampling station will very nearly duplicate those of the primary sampling system.

A preliminary design is being prepared for a cup-and-spigot sampler which will permit taking small-volume sodium samples (much less than the 10-ml beaker samples presently taken for metals analyses) in quartz vessels for carbon, oxygen, and hydrogen analyses. Provided that small samples can be taken without contamination, this method should successfully circumvent the impurity-segregation problem (see Sect. II.E.1.m.(i).(a).(2)) because the total sample can be analyzed.

(c) Tritium in EBR-II Systems. Samples from the primary and secondary sodium, cover gas, shield- thimble-cooling, and water systems have been analyzed for tritium content. The primary and secondary sodium have also been analyzed for lithium content, since lithium-6 reacts with neutrons to produce tritium. The analytical data are summarized in Table II.E.12. The maximum permissible levels of tritium and tritium oxide in an uncontrolled environment are included for comparison purposes.

n. Experimental Irradiations and Testing

Last Reported: ANL-7399, pp. 77-82 (Nov 1967).

(i) Experimental Irradiations (D. W. Cissel)

(a) Status of Experiments in EBR-II. The status of experimental irradiations in EBR-II as of December 31 is shown in Table II.E.13.

TABLE II.E.12. Tritium Analyses

Sample/Date	Tritium (atoms/liter)	Tritium (pCi/cc)	Tritium, MPL for Uncontrolled Environment (pCi/cc)
Primary Argon			
7/18/67	1.2×10^{12}	60	
7/24/67	1.6×10^{12}	75	
11/8/67	1.5×10^{12}	70	
Shield Cooling Air			
11/9/67	$<2 \times 10^9$	<0.1	$^3\text{H}_2\text{O} = 5$ $^3\text{H}_2 = 400$ (Air)
Thimble Cooling Air	$<2 \times 10^9$	<0.1	
Well Water	$<4 \times 10^{10}$	<2	$^3\text{H}_2\text{O} = 3 \times 10^4$ (Water)
Condenser Hot Well Condensate ^a			
11/8/67	2.7×10^{11}	13	
12/4/67	2.3×10^{11}	11	
Boiler Blowdown ^b			
12/4/67	2.3×10^{11}	11	
Superheated Steam ^c (Condensed Sample)			
12/4/67	2.3×10^{11}	11	
Steam in Sodium Boiler Building ^d (Condensate Sample)			
12/4/67	8.2×10^{10}	4	
Primary Sodium			
11/14/67	3.8×10^{13}	1840	
Secondary Sodium			
11/21/67	8.2×10^{12}	400	
Secondary Argon			
12/4/67	9.1×10^{10}	4.4	
<u>Lithium (ppm)</u>			
Primary Sodium			
11/17/67	0.56		
11/24/67	0.57		
Secondary Sodium			
10/20/67	0.52		

NOTE: Sampling Locations

^aDischarge of condensate pump.^bInfluent to flash tanks.^cFrom steam line entering power plant.^dEffluent of pump discharging steam-trap condensate from collection sump.
(This value is probably low because the condensate from the power cycle steam traps is diluted by condensate from the heating steam system which is independent of the power system and is fed by the domestic oil-fired boilers.)

TABLE II.E.13. Status of EBR-II Experimental Irradiations on December 31, 1967

Subassembly	Date Charged	Capsule Content and Number of Capsules ()	Experimenter	Approx Accum Exp ^a (MWd)	Goal Exp (MWd)
XG02	7/16/65 ^b	UO ₂ -20 w/o PuO ₂ (1) Stainless Dummies (18)	GE	13,032	13,650
XG03	7/16/65 ^b	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	13,032	19,450
XG04	7/16/65 ^b	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	13,032	39,000
XG05	9/3/65	UO ₂ -20 w/o PuO ₂ (9) U-15 w/o Pu-10 w/o Zr (1) U-15 w/o Pu-10 w/o Ti (1) UC-20 w/o PuC (3) Structural (5)	GE ANL ANL ANL GE	12,455	13,750
XA08	12/13/65	UC-20 w/o PuC (8) Structural (11)	ANL	10,633	19,800 ^b
XO10	3/24/66	UO ₂ -20 w/o PuO ₂ (4) Structural (11) Structural (4)	GE ANL PNL	9,963	19,600
XO12	8/10/66	UO ₂ -20 w/o PuO ₂ (19)	NUMEC	6,263	20,100
XO15	11/15/66	PuO ₂ -UO ₂ (11) (Pu _{0.2} U _{0.8})C ^b (4) PuO ₂ -UO ₂ ^b (2) Mk-1A (Metal) ^b (2)	NUMEC ANL GE ANL	4,608	11,000
XO16	1/13/67	Structural (9) Structural (10)	ANL GE	3,288	5,400
XO17	11/15/66	PuO ₂ -UO ₂ (11) (PuU)C (3) Mk-1A (Metal) (5)	NUMEC UNC ANL	4,608	6,500
XO18	12/6/66	Structural (3) Structural (1) Structural (2) Structural and Heavy Metal Fission Yield Samples (1)	GE PNL ANL ANL	3,918	21,300
XO19	1/13/67	UO ₂ -20 w/o PuO ₂ (7) (U _{0.8} Pu _{0.2})C ^b (3) Structural (8) Graphite (1)	GE UNC PNL PNL	3,288	12,500
XO20	1/13/67	Mixed Oxides (9) Mixed Carbides (3) Structural (4) Structural (2) Graphite (1)	GE UNC PNL ANL PNL	3,288	25,000
XO21	2/27/67	Structural (7)	PNL	3,288	21,500 ^b
XO22	2/27/67	Structural (7)	PNL	3,288	18,000 ^b
XO25	10/10/67	Structural (19)	GE	1,648	5,000
XO26	10/11/67	Structural (7)	NRL	1,648	3,000 ^b
XO27	11/22/67	Mixed Oxides (18) Structural (1)	GE PNL	619	7,200
XO28	11/22/67	Metal Alloy ^b (19)	ANL	619	9,200
XO30	10/10/67	Thermocouple Material (2) Graphite (3) Structural (1) Stainless Dummies (13)	PNL PNL PNL	1,648	No Limit
XO31	11/22/67	Mixed Oxides (19)	PNL	619	2,200
XO32	11/22/67	Mixed Oxides (19)	PNL	619	11,000

^aAccumulated exposure during December was 435 MWd for all subassemblies shown in the table.^bUnlisted or incorrectly listed in Progress Report for November 1967.

Experimental subassemblies XO29 and XO33 were loaded into the primary-tank storage basket to await installation on the grid. The following is a brief description of the contents of the two subassemblies with the grid positions into which they will be loaded:

Subassembly	Experimenter	Capsule Contents and Number of Capsules ()	Grid Assignment	Goal Exposure (MWd)
XO29	ANL	Mk-II Fuel (37)	4-E-3	10,200
XO33	UNC	(U _{0.8} Pu _{0.2})C (19)	5-E-2	10,000

Following the insertion of these two experimental subassemblies, the reactor core will be very near its capacity for low-worth subassemblies, i.e., subassemblies in which the reactivity worth is significantly less than for standard driver subassemblies. With the installation of the ORNL-sponsored structural materials subassembly XO34, which is approved in principle for Row 2, the core will be fully loaded with low-worth experimental subassemblies for the start of Run 28.

(b) Status of Irradiation Experiment Proposals Given Administrative Approval-in-Principle (AIP). Table II.E.14 summarizes the present status of all irradiation experiments which have been approved in principle by the LDO but which have not been inserted in the reactor.

TABLE II.E.14. Status of Experiments with AIP

Sponsor	Material of Test	No. of Subassemblies	No. of Capsules	Suggested Reactor Row ^a	Date of AIP	Estimated Idaho Site Delivery
ANL	Oxide Fuel, Group O-3	1	19	3	9/22/67	4/68
	Carbide Fuel, Group C-5	1	19	2	9/22/67	4/68
	Oxide Fuel (unencap) Safety Program	1/2	19	4	9/6/66	6/67
	Mk-II Fuel	5	91 (ea)	Core	9/22/67	6/68
	Structural Materials, Group S-9	1	19	4	9/22/67	2/68
	Tritium Yield	1 (IB)	-	7	6/8/67	5/68
	Magnetic Materials	1	7	7	6/8/67	6/68
GE	Oxide Fuel, Group 8B	1	19	7	6/7/67	2/68
	Oxide Fuel (unencap) Group 3	1/2	16	4	9/6/66	6/67
INC	Structural Materials Fatigue No. 1	1	7	7	6/6/67	2/68
	Structural Materials Fatigue No. 2	1	7	7	9/21/67	4/68
LASL	Oxide Insulators	1	7	8	6/6/67	4/68
	Carbide Fuels	1/3	7	3	2/27/67	3/68
NRL	Structural Materials	1	7	7	9/6/67	6/68
ORNL	Oxide Insulators	1	7	8	9/21/67	4/68
	Structural Materials	1	7	2	1/20/67	1/68
	Structural Materials	1	7	7	1/20/67	1/68
PNWL	Oxide Fuel (unencap) (low)	1	37	3	9/21/67	5/68
	Oxide Fuel (unencap) (norm)	1	37	4	9/21/67	5/68
	Oxide Fuel (unencap) (high)	1	37	3	9/21/67	6/68
	Graphite Materials	1	7	7	7/31/67	(No estimate)
	Thermocouple Materials	1	7	4	7/31/67	2/68
UNC	Carbide Fuel (129-147)	1	19	7	6/6/67	7/68
	Carbide Fuel (high power)	1/3	6	3	1/20/67	6/68
W	Carbide Fuel (+ V-alloy)	1/5	2	3	9/21/67	1/68
	Carbide Fuel (+ V-alloy)	1/2	9	4	9/21/67	3/68

^aBased upon fission rates at a reactor power level of 45 MWt.

(c) Procurement and Inventory Control of Experimental Hardware. The vendor for fabrication of experimental capsule hardware was selected and an initial order placed. When filled, this order will provide a maximum stock of end fittings for all sizes of experimental capsules. When the supply decreases to the established minimum, a new order will be placed.

Additional bar stock and 0.375-in.-OD tubing were ordered to fill out the stock requirements for these materials.

(d) Other Work. Experimental irradiation subassemblies XO29, a Mark-B-37 subassembly containing 37 ANL-MET encapsulated Mark-II driver fuel elements, and XO33, a Mark-A subassembly containing 19 UNC mixed-carbide fuel capsules, were flow tested. The prototype of the reorificed driver fuel subassembly having upper inlet holes enlarged to 0.359 in. was flow tested for Rows 1 through 5.

Depleted-uranium inner-blanket subassembly A793, with a modified type B lower part, was flow tested.

A temperature test capsule of Type 304 stainless steel containing melt-wire monitors and having borated stainless steel follower weights was heated for 128 hr at 1100°F. No effects on the capsule were apparent from external or X-ray examination. Neutron radiographs were made with this test capsule installed in a slot in a subassembly top adapter and a trifluted axial reflector section. Resolution was significantly improved with the borated stainless followers. Elevation of the followers in the capsule could be determined in the heaviest reflector section.

(ii) Nondestructive Testing (D. W. Cissel)

(a) Capsules Examination. United Nuclear Corp. carbide Capsules Nos. 92 and 106 were X rayed and determined acceptable for insertion in the reactor. Capsule No. 92 has previously been rejected because of poor weld penetration on the lower capsule weld and was re-encapsulated by the experimenter.

Four General Electric Co. structural irradiation capsules from Group L-15 were received and examined. One will be returned to the experimenter for rework of a lower weld.

Eight Pacific Northwest Laboratory structural capsules were inspected and found satisfactory.

Singly clad oxide elements from ANL (Reactor Physics Division) were X rayed. All welds and the general condition of the elements were satisfactory except for element No. 001. This element

had evidently been damaged in transit; X radiography indicated pellet separation and chips. The experimenter has been requested to review the X rays and advise ANL-ID concerning the acceptability of this element for irradiation.

(b) Other Nondestructive Testing. The electronic equipment for the eddy-current tubing-flaw detector was completed; it is now ready for testing and calibration to determine sensitivity and establish a correlation between test indications and flaw size and configuration.

(c) Antimony-Beryllium Neutron Source. Preliminary measurements of neutron flux from an antimony tetroxide-beryllium gamma source which had been irradiated in MTR for one cycle (approximately 800 Ci) indicated a flux of 5×10^4 to 1×10^5 n/cm²-sec as measured at the exit of a conical collimator 24 in. long. Radiography of experimental capsules with this flux is expected to require an exposure time of 6 to 8 hr.

A neutron radiograph made with this source indicated a lack of definition which could not be accounted for on the basis of geometry alone. Measurements demonstrated that approximately 20 to 30% of the neutrons incident upon the foil were of epithermal energy, originating outside the collimator. The application of Boral plate within the source slot effectively improved definition. This source is not expected to provide radiographs with the degree of resolution attainable with the much greater flux intensity of TREAT; however, the in-cell radiography facility will permit greatly simplified screening of irradiated capsules and fuel elements. Those which show internal features requiring more detailed examination will be radiographed at TREAT.

Collimator aperture and configuration are being further evaluated with the source in the EBR-I repair cell. This work will serve to fix the collimator design to be used initially with the source in FCF.

(iii) Handling and Examination (D. W. Cissel)

(a) Irradiated Capsules from XO11. No further handling or examination has been done on the 13 capsules from subassembly XO11 that remain in Idaho. The capsules are awaiting transfer to the Argon Cell for storage until return to the reactor or other disposition.

(1) 5P-9 and 5P-12. Pacific Northwest Laboratory metallographically examined the capsule walls and reported them to be in satisfactory condition; PNL believed that the capsules could have been safely returned to the reactor for further exposure.

(2) F4D. General Electric Co. examined the wall of Capsule F4D and reported that the microstructure of the wall was essentially unchanged as a result of irradiation and subsequent handling. On the basis of its review, GE also considers that the balance of the capsules from experimental subassembly XO11 are in satisfactory condition for further irradiation.

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

(1) Cyclical Testing of Types 304L and 304 Stainless Steel Tubing Exposed at 1100°F, Washed in Water, and Stored in Air. Mounting of tube samples for an electron microprobe examination is in progress.

(2) Cyclical Testing of Stressed Capsules (Type 304 Stainless Steel) at EBR-II. Experimental subassembly X900 containing the pressure-stressed tubing samples is in the fourth and final soaking cycle in the primary tank storage basket. The next residence in primary sodium is proposed to be in Row 7 of the reactor.

Preliminary metallographic examination of the first two sets of pressure-stressed tube samples shows no evidence of deleterious corrosion. Samples from the third cycle are presently being examined.

(iv) Safety Review of Experiments (D. W. Cissel)

After review of experimental irradiation Capsule GE-E1H-36 to determine whether it met Irradiation Guide stress criteria with an upper capsules wall defect of 11%, the capsule was accepted. Experimental subassembly XO33, a Mark-A-type subassembly containing 19 mixed-carbide fuel capsules, was reviewed and approved for irradiation in Row 5.

(v) Electromagnetic Test Equipment for the In-cave Testing of Sodium Bonds in Irradiated Capsules (C. J. Renken)

Electromagnetic test equipment is being developed to test the sodium-bond quality of capsules that have undergone some irradiation in EBR-II and are withdrawn for interim inspection. The prototype equipment is now complete and is being tested. The actual equipment that will be installed in the Argon Cell in the FCF is under construction.

The tests on the prototype equipment indicate that in the Mark-A capsules, voids of a volume equivalent to a 1.6-mm-dia sphere can be detected under a sodium layer of 0.76 mm. A good indication of depth of the void in the annulus is obtained independent of void size. While 100%

detection of nonwetted areas on the inside of the capsule is not expected, simulated tests indicate that annular gaps between sodium and jacket of less than 0.0025 mm are clearly detectable if they extend around the jacket circumference at least 90° and along the capsule longitudinal axis for at least 1 mm.

The prototype equipment has so far provided a satisfactory test for voids in the Mark-A capsules. The Mark B-19, B-37, and B-61 types of irradiation capsules have a spiral spacer wire attached to the outside of the jacket. The spacer wire will seriously complicate the application of the electromagnetic test at best, and may make it impractical. When samples of these types of capsules have been obtained, determination of the seriousness of the spacer wire problem will be possible.

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

Last Reported: ANL-7399, pp. 82-84 (Nov 1967).

(i) Operational Analysis. All computer programs for processing FCF production and surveillance data have been completed. Data are now being placed on magnetic computer tape. The first data retrieved from the tape files will be a summary of fuel-pin processing for the month of December 1967. The data will be computer analyzed to determine fuel-pin and fuel-element acceptance averages and production bottlenecks.

Results from the data system will be included in future monthly reports to provide a better description of the production capabilities and achievements of the FCF. This was not possible in the past because of the lack of personnel effort required to hand analyze the FCF production data.

(ii) Test and Analytical Methods. The recent receipt of two suspect driver-fuel subassemblies from EBR-II has emphasized the need for development of satisfactory test methods for determining and identifying defective irradiated fuel elements.

The following are the preliminary test methods which have been used to date and which will be improved upon in the future as a result of operating experience:

(1) Sodium is washed from subassemblies in the Inter-building Coffin (IBC), utilizing a special "Suspect Subassembly Procedure." This procedure includes sampling of the IBC system for radioactive gases and wash-water activity during the washing operation, and monitoring of the wash-water effluent for sodium in order to assure efficient sodium removal from the subassembly.

(2) Each element of the subassembly is visually examined during subassembly-dismantling operations. Elements identified as suspect during these "through-the-window" inspections will also be visually and photographically examined by use of the through-the-wall periscope.

(3) The upper closure welds of all elements of the subassemblies are subjected to the standard FCF pressure-decay leak-detection test.

(4) Each element is subjected to the standard FCF eddy-current testing of sodium bond and level at cell temperature. This testing should indicate features such as gross sodium-level changes, deficiencies in the sodium bonds, and possible fuel relocation, which could be related to leakage or failure of fuel elements.

(5) Depending on the results obtained in Step (4) above, suspect elements may be subjected to sodium-bond and -level tests at elevated temperatures to aid in interpretation of the previously conducted cell-temperature tests. Additionally, by causing the bonding sodium to become molten at the test temperature and by possibly causing defect regions in the cladding to "open up" as a result of the temperature and expansion effects, visual evidence of sodium leaks through the element cladding could be obtained if such defects are present in the sodium-bond region of the elements.

(6) Elements remaining suspect after Step (5) may be subjected to a modified FCF sodium-rebonding treatment at 500°C. Here again, should severe cladding defects be present in the sodium-bond region, visual evidence of sodium leakage through the jacket may be obtained.

Failure of Steps (1) through (6) to confirm cladding leaks or failures in the sodium-bond region or in the closure-weld region of the elements will require that further leak-detection methods be applied.

Methods are now being formulated and equipment is being assembled for an attempt to monitor the suspect elements for evidence of release of radioactive gases (^{40}Ar , ^{133}Xe , and ^{135}Xe) during postirradiation heat treatments. The heat treatments will be conducted in vacuum chambers which will be monitored for radioactive gases which would be evolved by a leaking fuel element.

Another method being explored is the possibility of re-irradiation of suspect fuel elements in TREAT. Elements would be given low-power irradiation while enclosed in a special chamber. Following this, the chamber containing the fuel elements would be flushed with inert gas and sampled for evidence of evolution of radioactive xenon (^{133}Xe and ^{135}Xe) from leaking elements.

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

Last Reported: ANL-7399, p. 85 (Nov 1967).

(i) Auxiliary Equipment. The loading and unloading of the transfer cask for the interim subassembly-storage area will require an in-cask subassembly grapple. Air cooling of the contained subassembly is proposed. Mechanical and pneumatic tests of a model of one proposed grapple have been performed. Although test results disclosed deficiencies in the proposed design, the overall results are considered promising. Difficulties were encountered when the grapple, under load conditions, twisted in one direction, and then after release of the load twisted in the opposite direction. This would cause the hoist cable and the grapple triggering cable to entangle and interfere with the triggering mechanism for release of the load from the grapple.

To overcome the twisting problem, the cable system for the grapple is being redesigned. Fabrication of a new model is planned, utilizing many of the components from the previous model. Testing will be resumed following construction of the new model.

The pneumatic tests revealed an air-leakage rate of 2 to 4 cfm in the small annular clearances of the grapple system. This leakage is not considered excessive (it may be compared with the 10 cfm of air required to cool the subassembly). The leakage measurements are consistent with flow predictions based on laminar flow in small annular clearances.

The transfer port and view window for the subcell passageway between the Argon Cell and Air Cell have been completed and will be checked out by FCF personnel. This port will allow transfer of medium-sized items in and out of the Argon Cell without going through the Air Cell. Master-slaves will be installed at this station to assist with transfers and to make minor repairs to equipment on the transfer platen.

Preparations continue to be made for further testing of the top-loading and -unloading cask which will be used to transfer irradiated subassemblies to the Idaho Chemical Processing Plant (ICPP). This cask employs water to cool the contained subassembly convectively. Plans are in progress for using an irradiated subassembly to obtain temperatures of the circulating coolant. Safe conduct of this experiment has required selecting a subassembly with enough fission product decay heat to promote natural circulation of the water coolant but not enough to require forced-air cooling, should handling problems arise during insertion of the subassembly in the water-filled cask. A subassembly believed to meet this requirement is available and has been set aside for this experiment. Just prior to introducing the subassembly in the cask, air cooling of the subassembly is discontinued to prevent drawing water into the grapple system. The ICPP has fabricated equipment needed for unloading the cask at the water storage basin. Testing of unloading procedures at the ICPP will be conducted with a dummy subassembly.

q. Superheater and EM Pump Study and Test (R. A. Jaross)

Last Reported: ANL-7399, pp. 85-86 (Nov 1967).

(i) Superheater Vibration Study. Baffles are expected to cause the shell-side flow pattern in the EBR-II superheater to exhibit a relatively strong crossflow component. A comparison of the Strouhal numbers for the tubes and the vortex shedding has been used to evaluate qualitatively the potential for exciting a resonance condition in the tubes.

When fluid flows across a blunt body, vortices are shed alternately from each side of the body, inducing a periodic force perpendicular to the flow. If the vortex-shedding frequency is close to a natural frequency of the body, a resonance condition can occur. Therefore, the fundamental natural frequency of the structure must be greater than the maximum expected vortex-shedding frequency.

The vortex-shedding frequency is a function of the geometry of the body, transverse and longitudinal spacing between adjacent bodies, and the flow velocity; it is represented by the Strouhal number, a nondimensional number which is a function of the Reynolds number, given by

$$N_s = f_v d / U = N_s(N_R),$$

where f_v is the vortex-shedding frequency, d is a characteristic length of the structure, and U is the crossflow velocity. To characterize the potential for a structural member to be excited by a vortex-shedding phenomenon, we define a Strouhal number for the structure as

$$(N_s)_{str} = f_s d / U,$$

where f_s is the fundamental natural frequency of the structure.

Considerable work has been done on vortex shedding from a single cylinder, but the mechanism is much more complicated for a series of parallel cylinders. Chen* has measured the vortex-shedding frequency in staggered tube banks and shows that in the range of Reynolds numbers from 1.5×10^4 to 6×10^4 , the Strouhal number of the vortex frequencies is always constant for each arrangement of tubes; the results of the measurement are presented as a family of curves.

Heat-exchanger tubes are often mounted on a triangular pitch, for which two extreme cases are defined in Fig. II.E.12, where subscripts 1 and 2 represent the two cases, T is the spacing transverse to the

*Chen, Y. N., Flow-induced Vibration and Noise in Tube-Bank Heat Exchangers due to von Karman Streets, Paper presented at The ASME Vibration Conference, Boston, Mass., March 1967, ASME Paper No. 67-VIBR-48.

flow direction, and L is the longitudinal spacing. Geometrical considerations give $T_1 = 0.577L_1$ and $T_2 = 1.732L_2$. The curves in Fig. II.E.12 for $X_{t1} = 0.577X_{l1}$ and $X_{t2} = 1.732X_{l2}$

Fig. II.E.12. Strouhal Number vs. Spacing Ratio for a Tube Bank Mounted in a Triangular Pitch. U = average velocity through minimum cross section.

were sketched from Chen's results as an aid in applying them to heat-exchanger tubes mounted on a triangular pitch. These curves show the effect of the flow-field orientation on the Strouhal number or vortex-shedding frequency.

To assess the potential for the EBR-II superheater tubes to vibrate in resonance with the vortex-shedding phenomenon, we compare the Strouhal numbers for the tubes with the one for the flow. As a rule-of-thumb in the design of heat exchangers, the Strouhal numbers for the tubes should be at least 25% higher than that for vortex shedding.

Calculation of a Strouhal number for the structure requires an

average value for the crossflow velocity, which can be obtained by calculating the crossflow area as the product of the number of tubes in a longitudinally sliced section, the minimum clearance between adjacent tubes (pitch minus tube diameter), and the length between baffles (baffle spacing), and then dividing this area into the average volume rate of flow. (An average crossflow velocity calculated in this manner for the Advanced Test Reactor primary heat exchanger compared favorably with experimentally measured velocities from an independent study.)

For the EBR-II superheater, a minimum crossflow area of 1.22 ft^2 was calculated. With sodium flowing at $1,250,000 \text{ lb/hr}$ at full reactor power, the average velocity was then $\bar{U}_c = 5.26 \text{ ft/sec}$. Now the Strouhal numbers for the two tube-support configurations can be computed, based on percent of full-power flow. The Strouhal numbers for the superheater tubes are summarized in Table II.E.15. The Strouhal number for the flow depends on the orientation of the flow fields, which is as shown in Fig. II.E.12. A baffle plate for the EBR-II superheater (and evaporator) is shown in Fig. II.E.13. The flow pattern through the superheater is oriented in the "2" direction, as defined in Fig. II.E.12, with respect to the tube configuration illustrated in Fig. II.E.13.

TABLE II.E.15. Strouhal Numbers of Superheater Tubes

Percent of Full-power Flow ^a	Flow Configuration No. 1	Flow Configuration No. 2
60	0.62	0.85
74(45 MWt)	0.50	0.69
80	0.47	0.64
100(62.5 MWt)	0.38	0.51

^aFull-power flow = 1,250,000 lb/hr; $\bar{U} = 5.26$ ft/sec.

In the EBR-II superheater, the 1.438-in.-dia tubes are mounted on a 1.938-in. triangular pitch, which gives $X_{t_2} = T_2/d = 2.34$.

Fig. II.E.13. Baffle Plate for EBR-II Superheater and Evaporator

From Fig. II.E.12 we read 0.66 as the Strouhal number for vortex shedding, which can be assumed to remain constant over the range of Reynolds numbers of interest.

Comparison of the Strouhal numbers for the tubes, from Table II.E.15, with that for the flow shows that in the worst case, viz., 100% full-power flow (62.5 MWt) and Flow Configuration No. 1 ($f_s = 16.4$ cps), the Strouhal number for the tubes is less than that for the vortex shedding.

At current operating power (45 MWt), the flow through the tubes is 74% of full-power flow. Because the Strouhal numbers for the tubes at this flow are close to that for vortex shedding, a more detailed analysis is needed to define better the potential for resonant vibrations.

In general, Strouhal-number comparisons are good only for qualitative evaluation of the potential for crossflow-induced resonant vibration in heat exchangers. The reasons are: (1) the Strouhal numbers for the flow are usually based on the frequencies of vortex shedding from stationary cylinders or tube banks; (2) the Strouhal numbers for the flow assume uniform crossflow, whereas in the real case (that is, within a heat exchanger) a nonuniform velocity distribution is expected; (3) in the real case, the flow pattern is not ideally crossflow, but contains parallel components that may tend to damp out vortex formation, and (4) it is often difficult to predict analytically the natural frequencies of tube vibrations, and hence Strouhal numbers for the tubes, because of mounting methods and restrictions at baffle locations.

Further effort must include a comparison of vortex-shedding phase relations in practice with those required to feed energy into

the tubes, particularly in view of the nonuniform flow field in the heat exchanger as compared with the uniform flow fields intentionally used in experimental studies of this phenomenon (in wind tunnels, for example). This nonuniform flow could significantly influence the potential for producing sustained vibrations.

The results in Fig. II.E.12 are of further interest for their possible aid in minimizing the potential for crossflow-induced vibration in future baffle-plate designs. For example, if the EBR-II baffle plate (see Fig. II.E.13) were cut along the line A-A, the flow would be oriented in the "1" direction (as defined in Fig. II.E.12 and $X_{t_1} = 1.35$). On the curve of Fig. II.E.12, the Strouhal number for the flow would read 0.18. A comparison of this number with those for the tubes (see Table II.E.15) reveals that in the worst case the Strouhal number for the tubes would be 2.1 times greater than that for the shedding of vortices--sufficiently greater to expect the tubes not to vibrate in resonance with the vortex shedding. Nevertheless, we must be aware that the low-frequency periodic force arising from vortex shedding is still present to "drive" the tubes in a forced-vibration response.

r. Reactor Instruments, Nuclear Instrument Test Facility Study (B. C. Cerutti)

Last Reported: ANL-7399, pp. 86-87 (Nov 1967).

(i) "01" Thimble Modification to Provide a Facility for Tests up to 650/700°F, Phase I. A high-temperature fission counter, Type WX-4036 (experimental unit), for the nuclear instrument test facility, Phase I, and a special carriage for attaching this instrument to the existing "01" thimble insert, are on hand. A gamma chamber for mapping the flux in the "01" thimble is on order, with delivery expected within a few weeks.

A drawing showing required electrical changes to the shut-down circuitry for the nuclear instrument test facility is being prepared.

(ii) "01" Thimble Modifications to Provide a Facility for Tests from 650/700 to 1200°F, Phase II. Detailed design sketches for the nuclear instrument test facility, Phase II, were completed for the high-temperature thimble, thimble extension, and some details of the thimble insert. These designs are based on the assumption that a supplementary gamma source will not be required.

A sketch of a proposed heater and a bid were received from the Falcon Equipment Company. The sketch and bids are being evaluated.

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

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

Last Reported: ANL-7399, p. 87 (Nov 1967).

The ultimate goal of the tagged-element study is to develop a reliable technique for the identification of tagged elements in liquid sodium. Of equal importance is that the tag element be readily available for dispersion into the main sodium stream should a fuel failure occur.

Recently our investigations of ceramic fuels indicated that quantities of fission product antimony have migrated out of the fuel pin to its surface. In several of the pins examined, the antimony was found to segregate in small pockets on the surface of the pin. These findings, along with the fact that antimony is soluble in sodium at nominal temperatures (e.g., ~10 a/o at 600°C), suggest that antimony may be a useful tag element. By placing a suitable isotope of antimony at the cladding-fuel matrix interface the antimony tag would be available for dispersion into the mainstream of the sodium coolant upon breaching of the cladding. The detection of the tag in the sodium coolant should not be difficult.

Information has been received that the Associated Electronics Industries MS-7 spark-source mass spectrometer has been shipped from England and should arrive shortly. Construction of the laboratory facility for the tagged element study is proceeding on schedule.

(ii) Tag Confirmation Study (F. A. Cafasso)

Last Reported: ANL-7399, p. 87 (Nov 1967).

The feasibility of individually tagging fast reactor fuel elements with isotopic tracers is under investigation. The purpose of such tagging would be to determine, by monitoring the release of the tag into the sodium coolant, which fuel element in an operating reactor has ruptured. As part of this program, the behavior of potential tags in a circulating sodium stream must be studied in order to determine whether their concentrations are stable or are reduced below limits of detection owing to absorption on surfaces and cold-trapping. A small-scale experimental loop will be used for this purpose. The engineering design of this loop is 30% complete, construction of vacuum-line and gas-purification accessories has begun, and bids have been solicited for suitable electromagnetic pumps, and magnetically operated, bellows-sealed stainless steel valves.

t. EBR-II Materials--Coolant Compatibility (J. E. Draley)

Last Reported: ANL-7399, pp. 88-92 (Nov 1967).

(i) Metallographic Examination. Optical metallographic examination has been carried out on about half of the hardness samples from the Materials Surveillance Subassembly Number 1. They were exposed to primary sodium in the blanket region of EBR-II while loaded in perforated stainless steel tubes. Control specimens of the same materials, stored in air at room temperature, were also examined for comparison.

Observations are summarized in Table II.E.16. Only the beryllium copper showed evidence of significant surface dissolution. The selective nature of the attack is shown in Fig. II.E.14. This material apparently overaged from the thermal exposure. Precipitation and agglomeration also occurred in Type 420 stainless steel.

TABLE II.E.16. Effect of Irradiation, in Sodium at 700°F, on Micrographic Appearance of Reactor Materials

Material	Fluence (fast) ($n/cm^2 \times 10^{-19}$)	Observations
304 SS	1.3	Little or no difference between control and irradiated specimens; same surface transformation (striations in micrograph) apparent in both.
Borated 304 SS	1.0	No change.
347 SS	0.8	No change apparent, although slight difference in metallographic etching sensitivity noted.
416 SS	0.4	No change.
420 SS	1.4	Grain boundary precipitation and precipitate agglomeration after irradiation.
17-4 PH	1.4	No change.
T-1 Tool Steel	1.4	No change.
Stellite	0.4	Little or no change. Some transformation and accentuated grain boundary etching near surface, before and after exposure.
Inconel-X	1.4	Some change in sensitivity to etching.
Tantalum	1.4	No change.
Aluminum Bronze	1.1	Dispersion of γ_2 -phase during exposure, probably due to peritectoid reaction at reactor blanket temperature.
Beryllium Copper	1.4	Irregular surface dissolution. Considerable fine precipitation; apparently overaging, since time and temperature far exceed standard aging conditions.

There has been no definite evidence of material damage to any of the specimens from contact with the sodium (excepting beryllium-copper) or from the irradiation, as distinct from effects due to time at the elevated temperature.

(ii) Diffusion Studies
(F. A. Cafasso)

The finding of copper, tin, bismuth, and lead in the sodium coolant of EBR-II has focused attention on a need for information concerning the interaction of these contaminants with Type 304 stainless steel. It is the objective of this program to satisfy this need by kinetic and equilibrium studies pertinent to elucidating the extent of these interactions.

The factor of prime importance in defining the scope of these studies is the rate at which each contaminant diffuses into stainless steel at reactor temperatures (375-650°C). Diffusion coefficients for the four elements (Cu, Sn, Bi, and Pb) in stainless steel, however, have not been measured. Judging from the coefficients for Cu and Sn in γ -Fe (a good stand-in for stainless steel), diffusion in stainless steel in the temperature range of interest may be slow, so slow in fact, that equilibrium measurements may not only be impractical but of doubtful utility. As a first step in this program, effort will be made to obtain experimentally the

Fig. II.E.14

Copper-Beryllium Sample H-4, As Polished; Fluence 1.3×10^{19} n/cm². The light area is the Cu-Be. This photograph shows an edge area which is typical of about half of the circumference.

requisite kinetic information. Appropriate sodium solutions of each of the contaminants will be exposed (at temperatures ranging from 500-650°C) to stainless steel of large surface area for extended periods to determine changes in the levels of each contaminant with time. The results of these experiments will serve as a basis for decisions regarding the nature, direction, and temperature range of future studies in this area.

u. Hot Fuel Examination Facility--Engineering and Design
(N. J. Swanson)

The updated reference design for HFEF (revised with respect to that used in the Preliminary Cost Estimate) was established on December 14, and was presented to the architect-engineering firm (Norman Engineering Co.) on December 20-21, 1967. This reference design provides better utilization of the basement regions and incorporates the ability to receive and handle, in a vertical position, experiments up to 20 ft in length. This design will form the basis from which a cost estimate will be developed for presentation in the forthcoming Feasibility and Cost Study report, scheduled for completion and release on February 16, 1968.

Conceptual engineering work is now proceeding on the various remote material handling systems, shielding, shielding penetrations, viewing devices, inert-gas system, and other systems and devices for which ANL has retained prime design responsibility. This work is being performed in Idaho and in the Remote Control Engineering Division at Argonne, Illinois.

It appears that the most convenient system for critical-path scheduling and cost monitoring for ANL-procured items will be a PERT system utilizing the 32-case 1604 computer operated by Phillips Petroleum Company at Central Facilities. Discussions were held between project management and personnel who are familiar with the computer programming both in Idaho and Illinois. The goal is to have an operable critical path system by the end of January.

2. Outside Fuel Procurement (C. E. Stevenson)

Last Reported: ANL-7399, pp. 92-93 (Nov 1967).

a. Mark-IA-IB Fuel Procurement

(i) Fuel Elements. Aerojet-General Corp., with whom a contract has been negotiated for the fabrication of fuel elements, continued with procurement of materials and installation and startup of equipment. The ability to detect wall defects by the pulsed eddy-current inspection equipment for fuel-element tubing has been established, and defect standards are being submitted for evaluation by ANL. Delivery of a second 1000-ft qualification lot of tubing by the vendor (the Matthey Bishop Company) for Aerojet's evaluation is expected before the end of the month.

A lot of 50 fuel-element tips was received by Aerojet from the fabricator (Pressed Metals) for test purposes. These did not fully meet specifications, but an acceptable product is expected to be furnished soon after minor process modifications have been completed.

The alloy-preparation furnace has been installed and put in operation by Aerojet. Three melting and ingot-pouring runs have been completed satisfactorily using depleted uranium (molybdenum was also blended in the last melt). The injection-casting furnace has also been installed, and testing has begun. The inert-atmosphere glovebox for sodium loading and element welding has been received and installation is underway.

Analyses by Aerojet of samples from the first lot of ten 4.5-kg buttons of 52% enriched uranium obtained from Nuclear Fuel Services indicated a higher nickel content than the 200 ppm permitted by the alloy specification. Over 250 ppm was found in several samples by both emission spectrographic and atomic absorption methods of analysis. Confirmation of high nickel content of one sample was found in analyses by Argonne, and samples have been submitted for referee analyses. High nickel analyses have also been found in samples of a second lot of uranium buttons which was received more recently.

In support of a proposal to define the maximum hardness specification for the fuel-element tubing by a microhardness test requirement in place of the presently specified Rockwell B maximum, measurements to correlate Rockwell B and Vickers hardness were recently undertaken at Argonne. The data are being obtained on plate samples of Type 304 stainless steel, initially annealed at $1975 \pm 25^\circ\text{F}$, which were subjected to varying degrees of cold reduction in 2-3% increments up to 15%. The Vickers data are expected to be available for next month's report; the following data have already been obtained on Rockwell hardness:

<u>Sample</u>	<u>Approx % Reduction</u>	<u>Rockwell Hardness</u>	<u>Sample</u>	<u>Approx % Reduction</u>	<u>Rockwell Hardness</u>
1	0	B70.6	5	10	B96.6, C20
2	2	B84	6	12	C21.6
3	5	B92	7	15	C23.7
4	7	B94			

(ii) Subassembly Hardware. Aerojet-General Corp. is now also under contract to Argonne for the fabrication of 505 core, blanket, control, and safety hardware preassembly components (upper and lower units), and of 100 additional lower adapters. Aerojet reports making progress on planning and procurement for this contract. Quotations have been received for all stainless steel bar stock, tubing, and extrusions, and bids are being reviewed. Parts quotations have been requested from proposed subcontractors. Machinery relocation is being accomplished in the San Ramon plant. Manufacturing shop orders, which describe the proposed production operations, have been written and are being reviewed. Tooling criteria have been established and a "required tooling" chart has been developed.

3. Operations--Reactor Plant (G. K. Whitham)

Last Reported: ANL-7399, pp. 94-98 (Nov 1967).

a. Operations. Reactor operation in Run 26C amounted to 435 MWdt during December and to 619 MWdt since the start of the run on November 22. Two types of problems were encountered during this run:

(i) Small fission product releases occurred in the primary tank (see Sect. II.E.1.a). The first of these, at the beginning of the run, was soon identified with two newly loaded driver fuel subassemblies and replacement of these solved the problem. Another kind of release, consisting of a very slow xenon buildup, was still under investigation at the end of December.

(ii) Power-coefficient measurements continued to indicate a near-zero coefficient between 15 and 25 MWt. At the end of the run, with control rods at a high bank position, the power coefficient at 25 MWt reached the operating limit of 0.5 lh/MW.

The details of Run 26C operations follow:

Fuel handling before the run included the addition of experimental subassemblies XO27, XO28, XO31, and XO32. Subassemblies around the oscillator rod were relocated in an attempt to obtain a more uniform radial temperature distribution in the oscillator-rod thimble, and two standard driver subassemblies were replaced by high-silicon driver subassemblies C-2111 and C-2112 (see Sect. II.E.1.e).

Reactor startup on November 22 was followed by period calibration of No. 5 and No. 12 control rods, and intercomparison of all other rods. Power-coefficient measurements and oscillator-rod torque data were obtained at 5-MW increments during the approach to power. At approximately 25 MW the signal increased from the Fission Gas Monitor (FGM), indicating a possible small release of fission gas. Power was initially reduced to 17.5 MW, a primary-tank blanket-gas sample was obtained, and then power was further reduced to 50 kW. When the FGM signal continued to increase, the reactor was shut down. Analysis of the gas samples confirmed that a fission gas release had occurred, although it was of much lower magnitude than the release which occurred in May 1967. Building activity levels and the FERD system signals remained unchanged.

To confirm the release and to collect more definitive information which could be used for failure diagnosis, the reactor was started up on November 24. Power was raised in 5-MW increments to 25 MW, with a 1-hr period of operation at each increment. Soon after a power level of 25 MW was reached, the signal from the FGM increased from a background level of 7% to about 30% of full scale. Results of the analysis of a special gas sample taken at approximately this time indicated that a second release of fission product gas had occurred. Operation of the reactor continued under gas-release conditions at a power level of 25 MW for 8 hr, after which the reactor was shut down. During the 8 hr, the FGM signal level returned to approximately twice the normal operating background value, with no other detectable releases.

Pulse-height analyses of cover-gas samples indicated the probability that the failed pin was part of a driver subassembly with very low burnup; therefore, the two new driver subassemblies (C-2111 and C-2112) that had been inserted in the core at the end of Run 26B were replaced. Incremental increase of power to 45 MW and subsequent operation at that power level produced no further detectable gas releases.

During torque measurements of the oscillator rod, rubbing occurred at 30 MW. The measurements were terminated after 35 MW because of torque values above 30 in.-lb. The torque data showed some improvement over Run 26B because of the changed loading around the oscillator rod, and showed a slight difference in the angular orientation at which rubbing occurred. Constant- ΔT critical positions were obtained at 41.5 MW, 100% flow, and at 22.5 MW, 54% flow on November 29.

Further power-coefficient measurements on November 30 indicated a possibly positive point-differential power coefficient between 10 and 15 MW with control rods banked at 12 in. After lowering power from 15 to 10 MW and banking rods at 11.5 in., the power-reactivity decrement (PRD) between 10 and 15 MW was measured again and the

results agreed with those previously obtained. The reactor response was quite stable, with no control problems. Control rods were rebanked to 12 in. and power coefficient measurements continued to 45 MW. After reactivity was allowed to stabilize at 45 MW, the PRD agreed with other Run-26 measurements.

Normal operation continued until December 7, when rod-drop tests were performed in which control rod No. 5 was dropped from 5.85 in. at power levels of 0.5 and 45 MW. A ramp withdrawal of reactivity of approximately 55 lh was involved. In the 45-MWt test, a minimum power level of 15 MWt was reached before power was increased by the insertion of control rod. The data are being analyzed.

Operation at 45 MW continued until December 10, when reduced-flow experiments were carried out at the following reactor power levels and primary system flows:

Reactor Power (MWt)	Primary Flow (%)	Reactor Power (MWt)	Primary Flow (%)
41.5	100	22.5	75
30.0	72	22.5	60
22.5	100	22.5	54
22.5	90	15.0	36

Zero-power (500-kW) transfer-function measurements were conducted in an effort to establish a final basis for the normalization of 30- and 41.5-MW data. The results indicate that such a normalization may now be performed.

The final power-coefficient measurement for Run 26C was initiated on December 11. This test was terminated at about 2300 hr at a power level of 25 MW when the average power coefficient reached the operating limit of 0.5 lh/MW.

Heaters and a thermocouple in the large and small seal troughs were replaced. Temperature control for the half-molten condition of the seals was changed from thermocouples in the trough walls to thermocouples in the dip ring. Both seal troughs were thoroughly cleaned, after which 240 lb of tin-bismuth seal alloy were added to the troughs.

An evaluation study of the optimum reactor run length for 1.2, 1.4, 1.6, 1.8, and 2.0 a/o fuel burnup has been completed, and a report is being written. The optimum run length for the present burnup limit of 1.2 a/o is 1125 MWd. Average subassembly utilization of 91% is realized for this run length.

b. Operational Support and Maintenance

(i) Plant Maintenance

(a) Turbine-driven Condensate Pump. During investigation of the cause for loss of the inboard pump bearing on the turbine-driven condensate pump, the alignment of the turbine to the pump was found to be out of tolerance. The apparent cause of the misalignment was excessive piping loads on the turbine exhaust nozzle. The pipe hangers were modified to bring the nozzle loading within the manufacturer's specifications.

One week after the pump was returned to service, the inboard turbine bearing failed. The alignment of the turbine to the pump was rechecked and found to be within tolerance. The reason for the loss of the turbine bearings was not determined, but a new bearing was installed and initial operation of the unit has been satisfactory.

(b) Acid Injection into Condenser Cooling Water. A temporary injection nozzle for sulphuric acid was installed in the cooling water return line to the cooling tower. This will eliminate the need for handling carboys of acid at the cooling tower and will provide a means for controlling pH until the acid-injection system is moved to the tower.

(c) Oil Storage Tank for Main Turbine. Electric heaters were installed on the bottom of the turbine-oil storage tank. The heaters were installed to provide a means for heating the oil to drive moisture out.

(d) Gland Exhauster Pump for Main Turbine. The gland exhauster pump for the main turbine was disassembled for a general inspection and was found to be in excellent condition. This indicates that the use of raw water for pump cooling instead of plant cooling water has been successful. Wood fibers from the cooling tower had previously caused trouble by clogging the strainers in the pump cooling system.

(e) Steam-driven Auxiliary Oil Pump for Main Turbine. The auxiliary oil pump was disassembled for a semiannual inspection and was found to be in good condition.

(f) Turbine-driven Feedwater Pump. Oil samples taken from the lubrication system of the turbine-driven feedwater pump indicated that the viscosity of the oil was higher than normal. The oil was changed, and the oil coolers were inspected and cleaned.

(g) Primary Purification System. Work continued on installation and alignment of nozzle couplings on the spare cold traps, one of which will be installed to replace the existing cold trap.

(h) Air Cooling Blower for Primary Sodium Pump No. 2.

The motor on the air cooling blower for primary sodium pump No. 2 was replaced with a spare after the blower developed significant vibration, which indicated a potential bearing failure.

(i) Leak-rate Testing of a Reactor Building Penetration.

The penetration for the manual building-pressure equalization valve was satisfactorily leak tested.

(j) Reactor-building Argon-supply System. Prefabrication of the piping for installation of two argon-gas totalizers in the reactor-building supply system was completed.

(k) Reactor Vessel Cover Locks. New oil seals were installed in the upper drive assemblies on cover locks No. 1 and No. 2.

c. Operator Training. Four technicians completed qualification training for a plant system. Thirty-eight lesson-plan examinations were graded and returned to the shift crews for review. Six written examinations for final qualification were successfully completed by crew personnel.

Updating and revision of writeups, study guides, open book assignments, and examinations for the Power Plant Qualification Area continued.

d. Operating Manual. Revisions to the Plant Cooling Water and Normal Operating Procedures sections of the Operating Manual were completed, and a new section covering operation of the Fission Gas Monitor was drafted. Work continued on revisions to the Boiler Feedwater Treatment, Plant Cooling Water Treatment, Water Sampling Stations, and Sodium Sampling sections of the Manual.

Preparation of operating procedures to be posted near various items of equipment is in progress. Equipment control centers and systems panels are being checked so that any changes may be incorporated in the Manual.

A special operating procedure was issued for the plugging meter in the FERD system. Administrative procedures were issued for scram report evaluation and incident reporting and follow-up. Emergency procedures for fuel element rupture were revised.

New data sheets for recording initial conditions at the start of each reactor run and for an hourly power-plant equipment check were drafted and submitted for review.

e. Planning and Scheduling. The projected long-range operations program for EBR-II was completed.

4. Operations--Fuel Cycle Facility (M. J. Feldman)

a. Hot-line Operation. Routine operations were conducted, with emphasis on experimental and surveillance programs. Table II.E.17 summarizes the production activities.

TABLE II.E.17. Production Summary for Hot Line

	12/1/67 through 12/31/67	Total This Year
1. Subassemblies Received:		
Core, Control, Safety	10	74
Other	7	134
2. Subassemblies Dismantled (for processing)	6	78
3. Subassemblies Dismantled (for examination, etc.)	1	41
4. Subassemblies Fabricated	7	157
5. Subassemblies Transferred to Reactor	11	95
Subassemblies Stored in L&O Vault and Interbuilding Corridor	0	34
6. Elements Decanned:		
From Irradiated Subassemblies	486	6,498
Rejects	0	876
Other	163	163
Total Decanned	649	7,537
<u>Melt Refining</u>		
	<u>Irradiated Fuel</u>	<u>Recycle Material</u>
7. Number of Runs	4	4
8. Average Pour Yield, %	93.1	93.5
1967 to date (Total)	52	78
<u>Processing</u>		
9. Injection-casting Runs (Total Number)	14	175
10. Elements Processed:		
Accepted	681	14,576
Rejected	229	2,056
11. Elements Welded	763	23,544
Rewelded	0	17
12. Elements Leak-tested:		
Accepted	636	13,777
Rejected	27	320
13. Elements Bonded (including recycle)	1,145	17,793
14. Elements Bond-tested:		
Accepted	701	14,062
Rejected	68	891
15. Elements to Surveillance	105	864
Number of Subassemblies	3	59
<u>Waste Shipments</u>		
16. Cans to Burial Ground	5	97
17. Oxide and Glass Scrap to ICPP	1	53

Fabrication of two Mark-IB fueled subassemblies was completed. Fabrication of 70% enriched fuel elements has started in the hot line, using unirradiated fuel rods previously cast in Illinois. Remelting and recasting of these rods was required to adjust the silicon content to the desired level.

b. Cold-line Operations. A summary of the month's production data is given in Table II.E.18.

TABLE II.E.18. Production Summary for Cold Line

	12/1/67 through 12/31/67	Total This Year
1. Alloy-preparation Runs		
New Fuel	5	16
Remelts	4	9
Total	9	25
2. Injection-casting Runs	10	23
3. Pins Processed		
Accepted	770	1,707
Rejected	116	226
4. Elements Welded	758	1,617
Elements Rewelded	0	0
5. Elements Leak Tested		
Accepted	751	1,603
Rejected	7	14
6. Elements Bond Tested		
Accepted	659	1,364
Rejected	37	144
7. Subassemblies Fabricated	7	15

Another depleted-uranium ingot containing 2 w/o silicon was made for use as a master alloy in the silicon-addition program. Several different mold coatings were used for injection casting this material, including zirconium oxide, stabilized zirconium oxide, yttrium oxide, and magnesium zirconate. A few of the molds were air dried instead of oven dried. Although this casting was more successful than the previous one, the pins for the most part were very short and there was still a large amount of glass breakage. Enough material was cast for addition of silicon to a few alloying runs. The best pins were obtained from the oven-dried molds that were coated with zirconium oxide; this is the coating used in the regular cold-line injection-casting runs, and the coating used for the

previous run with depleted uranium-2% silicon. Since mold coating does not appear to be the problem in injection casting this alloy, further investigations will have to be made.

All the melts made in cold-line alloy preparation during the past month have had metallic silicon added to bring the total silicon content up to about 375 ppm. Analyses of injection-cast pins made from those ingots has indicated that this method of addition may be satisfactory.

c. Maintenance and Repair. A failure recurred in the electrical system of the bridge drive for the air-cell crane. This is the first instance of this type of failure since the bridge drive was replaced about one month ago.

The argon-compressor system (used to activate various pneumatic cylinders on argon-cell equipment) was modified so that it operates on intermittent cycle instead of continuously. This should greatly increase the useful life for the carbon-Teflon rings.

A study is being made of the argon-cell pressure-control system to make it more reliable with less instrument maintenance.

d. Operator Training. A lecture-demonstration course on unusual fires, i.e., sodium, alcohol, or metal, was given for all FCF operation personnel.

e. Operating Manuals and Reports. Seven sections were issued as accepted portions of the FCF Operating Manual: three were new, two were revisions, and two were made up of both new and revised material. Two new sections and one revised section are in various stages of preparation.

Criticality procedures for processing of 70% enriched fissium alloy were submitted and approved.

f. Analyses

(i) Chemical Analyses. The numbers of fuel-production analyses on hot- and cold-line samples, together with the average values and range, are given in Table II.E.19.

Analyses for surveillance of irradiated pins from the reactor are given in Table II.E.20.

TABLE II.E.19. Production Fuel

Analyzed for	Number	Average Value	Range
U (total and isotopic)	32	94.73 w/o	94.16-95.85
²³⁵ U		52.17 w/o	51.65-52.37
Pu (total and isotopic)	12	910 ppm	125-1530
Mo	33	2.70 w/o ^a	2.41-3.22
Ru	11	1.97 w/o	1.84-2.07
Pd	9	0.188 w/o	0.165-0.218
Rh	17	0.284 w/o	0.266-0.316
Zr	8	0.044 w/o	0.030-0.067
Nb	4	0.018 w/o	0.012-0.025
Fe	65	370 ppm	25-770
Al	21	160 ppm	55-360
Si	62	325 ppm	80-600
C	11	290 ppm	190-380
Total Analyses: 285			

^aThis apparent high average (2.46 w/o nominal) is being investigated.

TABLE II.E.20. Surveillance Pins

Analyzed for	Number	Average Value	Range
U (total and isotopic)	6	93.83 w/o	93.54-94.02
²³⁵ U		51.35 w/o	51.30-51.39
Pu (total)	8	1785 ppm	1557-1842
²³⁹ Pu		99.70 w/o	
²⁴⁰ Pu		0.30 w/o	
Mo	4	2.54 w/o	2.48-2.62
Fe	9	540 ppm	240-880
Al	6	150 ppm	70-180
C	16	260 ppm	160-580
Tc	7	450 ppm	370-490
Total Analyses: 56			

(ii) Postirradiation Analysis of EBR-II Fuel. Fuel-element surveillance data are now being placed on magnetic tape. These data will be computer analyzed and automatically plotted, resulting in a substantial saving in personnel effort.

Subassembly C-2113, which is one of the two subassemblies suspected of containing a defective fuel element (see Sect. II.E.1.a), was received from the reactor. The fuel elements from C-2113 were subjected

to visual checks and standard leak-detection and bond tests (at room temperature); no defective elements were identified by these examinations.

The Idaho Chemical Processing Plant (ICPP) is continuing its two testing programs concerned with the underwater storage at ICPP of EBR-II fuel elements. No additional results are available from either program. One program is intended to determine whether the stainless steel clad of the fuel, as a result of exposure to in-reactor sodium, would become defective when stored in the water environment of the storage basin. The other program is testing the leachability of a 1-in. piece of declad EBR-II fuel which is exposed to ICPP storage-basin water.

AEC (Idaho Operations Office) has authorized Idaho Nuclear Corporation to receive, store and process irradiated EBR-II fuel elements.

PUBLICATIONS

LIQUID-METAL FAST BREEDER REACTORS--CIVILIAN

Recent Theoretical Work in the U.S. on Fast Reactor Physics

H. Hummel, P. Greebler, W. B. Loewenstein, J. B. Nims, A. B. Reynolds,
and T. A. Pitterle

Fast Breeder Reactors, Proc. Conf., London, May 17-19, 1966.
Pergamon Press, Oxford, 1967, pp. 493-511

Fluidized-Bed Technique for Preparation of Uranium-Plutonium Monocarbide

E. J. Petkus, P. A. Nelson, S. Vogler, and P. W. Krause

Chem. Eng. Progr. Ser. 63(80), 111-120 (1967)

The Fabrication of Plutonium from Highly Irradiated Reactor Fuel

A. B. Shuck

Plutonium as a Research Fuel, Proc. IAEA Symp., Brussels,
March 13-17, 1967: Intern. Atomic Energy Agency, Vienna, 1967,
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The Codes NURF, COMBO, and TWOSORCE for Processing Foil Counting Data

George S. Stanford

ANL-7356 (Aug 1967)

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development-- Research and Development

1. Theoretical Reactor Physics

a. Cross-section Data Evaluation (C. N. Kelber)

Last Reported: ANL-7399, p. 105 (Nov 1967).

The status of the theoretical interpretation of slow-neutron cross sections for ^{239}Pu has been reviewed. As suggested by the recent compilation and evaluation of Greebler et al.,* sufficient new material is available to warrant a more careful and systematic calculational effort.

Particular attention has been focused upon the resolved resonance region which extends up to 300 eV. The methods of cross-section analysis developed and extensively used by F. T. Adler and D. B. Adler** appear to provide a suitable starting point for the determination of resonance parameters over the whole resolved resonance region.

With the above objective in mind, possible modifications of the method and the relevant computer programs are being studied. An MC² library tape containing 33 materials from the revised ENDF/B distribution has been prepared. Nearly all of the materials of interest in reactor applications are on the tape. Elastic-scattering Legendre coefficients are included for most of the materials having masses less than 100 amu. A few MC² test problems have been run using the tape. Many more problems will be run in the near future in order to test the data as completely as possible.

The EDIT program, which can be used to print portions of an ENDF/B tape in formats which include identifying Hollerith information, has been made operational on the CDC-3600. This program was written at Oak Ridge and distributed as an ENDF/B code.

b. General Reactor Theory Studies (C. N. Kelber)

(i) Coherence Effects in Superprompt Critical Excursions.

The effects of the time distribution of reactivity additions in superprompt critical excursions are frequently ignored in initial estimates of excursions in fast reactors. Instead, the total reactivity addition is usually assumed to

*Greebler, P., Aline, P., and Hutchins, B., Evaluation and Compilation of Pu-239 Cross Section Data for the ENDF/B Files, GEAP-5272 (Dec 1966).

**Adler, D. B., and Adler, F. T., Multilevel Analysis of the U²³⁵ Total and Fission Cross Sections in the Energy Region below 37 eV, BNL-50045 (T-455).

be inserted at one time, or as a single ramp. As shown by Gopinath and Dickerman,* such estimates overlook a realistic spread in time of reactivity excursions expected in real systems. There are two questions to be resolved:

1. What is the estimated reduction in total energy release from a superprompt critical excursion when the reactivity excess beyond prompt critical is introduced over a period of time instead of at a single instant?

2. How sensitive is the estimate of energy release to variations in the times at which the reactivity increments are made?

To answer these questions we can use a highly idealized model which has the crucial features of excursion calculations. For example, we have used the Nordheim-Fuchs model and assume that reactivity over prompt critical is added in N steps of amount ρ_0 , which is arbitrary, each. The mean time between steps is also arbitrary, and the actual time between steps is assumed to be normally distributed about this mean with a variance which is also arbitrary. Some care must be taken with this model since with a large ratio of variance to mean it is easy to encounter a negative time between steps.

Under these assumptions, the Nordheim-Fuchs equations during the n^{th} time step, in dimensionless form, are

$$\frac{dP_n}{d\tau} = (n - W_n) P_n \quad (1)$$

and

$$\frac{dW_n}{d\tau} = P_n; \quad (2)$$

at

$$\tau = 0: W_n = W_n(0) = W_{n-1}(\tau_{n-1}); \quad (3)$$

$$P_n = P_n(0) = P_{n-1}(\tau_{n-1}). \quad (4)$$

Also,

$$W_1(0) = 0; \quad P_1(0) = p, \ll 1.$$

* Gopinath, D. V., and Dickerman, C. E., Calculations of Coherence of Failure for Hypothetical Meltdown Accidents in an EBR-II-like Reactor, ANL-6844, (March 1964).

The subscript n denotes the solution during the n^{th} step; τ_{n-1} is the duration of the $(n-1)^{\text{st}}$ time step. The dimensionless variables are:

$$W = E/(\rho_0/\alpha);$$

$$P = P(\alpha\ell/\rho_0^2);$$

$$(\rho/\rho_0) = [n - (\alpha E/\rho_0)] \quad (\text{during the } n^{\text{th}} \text{ time step});$$

$$\tau = t\rho_0/\ell,$$

where

E = energy content of the reactor (MW-sec);

P = reactor power (MW);

α = reactor power coefficient (\$/MW-sec);

ℓ = reactor lifetime (sec/\$);

ρ_0 = reactivity increment beyond prompt critical (\$);

t = real time (sec).

These equations have well-known solutions:

$$W_n = n + \sqrt{2C_{1,n} + n^2} \tanh \left\{ \sqrt{2C_{1,n} + n^2} \left(\frac{\tau}{2} + C_{2,n} \right) \right\}$$

and

$$P_n = \frac{1}{2}(2C_{1,n} + n^2) \operatorname{sech}^2 \left\{ \sqrt{2C_{1,n} + n^2} \left(\frac{\tau}{2} + C_{2,n} \right) \right\},$$

where

$$C_{1,n} = \frac{1}{2}(W_n(0) - n)^2 + P_n(0) - \frac{1}{2}n^2;$$

$$C_{2,n} = \frac{1}{\sqrt{2C_{1,n} + n^2}} \tanh^{-1} \left(\frac{W_n(0) - n}{\sqrt{2C_{1,n} + n^2}} \right).$$

As an example consider the case $N = 2$, and examine three cases: $\tau_1 \ll 1$, $\frac{\tau_1}{2} + C_{2,1} \approx 0$, and $\tau_1 \gg 2C_{2,1}$. Recall that $W_1(0) = 0$ and $P_1(0) = p \ll 1$. The solution constants are: $C_{1,1} = p$; $C_{2,1} = -2/p^2$. Then, for

the case $\tau_1 = \delta \ll 1$, we have $W_1(\tau_1) = W_2(0) \approx 2\delta$ and $P_1(\tau_1) = p(1 + (2\delta/\sqrt{1+2p}))$. Using these initial conditions, we find

$$C_{1,2} \doteq p - 4\delta + (2p\delta/\sqrt{1+2p}),$$

whence for the final energy $(2C_{1,2} + 4)^{1/2} + 2 = 4 - 2\delta + \frac{p}{2}$. If all reactivity were introduced at once the final energy would, of course, be $4 + \frac{p}{2}$. Hence, the division into two steps decreases the final energy release by an amount corresponding to the small amount of reactivity compensation, 2δ , that occurred in the first step. If

$$(\tau/2) + C_{2,1} \doteq 0,$$

then

$$W_1(\tau) = W_2(0) = 1$$

and

$$P_1(\tau) = (1+2p)/2.$$

Thus, $C_{1,2} = 2p - 1/2$, and the final energy release is $2 + \sqrt{3+4p}$, to be compared with $2 + \sqrt{4+2p}$. The final energy release is smaller in this case by the factor $(1 + \sqrt{3/4})/2$ than the energy release obtained in a single step.

Finally, in the case $\tau/2 \gg |C_{2,1}|$, the final energy release is just $2 + \sqrt{p}$, corresponding to the complete reactivity compensation obtained during the first step. The reduction in energy is by a factor of $1/2$, corresponding to the fact that the reactivity after the first step is below prompt critical by just the amount added at the second step, so that the system does not undergo any further excursion.

Thus at this point we would conclude that a lack of coherence in reactivity calculations will reduce the total energy release, but the ratio of estimated energy releases will in general be close to one and certainly greater than $1/N$.

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

Last Reported: ANL-7399, pp. 105-107 (Nov 1967).

(i) Phase-1 Fuel-cycle Package--Search Options Available.

Two basic quantities may be adjusted to achieve required conditions during a problem: the enrichment of freshly loaded fuel at the beginning of a burn step, and the length of the burn step. Control materials may be adjusted

during a burn step to maintain criticality. Two search-procedure options have been defined which, in combination with variations in convergence criteria, allow several types of problems to be run. These options and some of the problem alternatives are described below. Other options may be added in later versions of the package.

1. Given the burnup limit for specific materials and the unpoisoned static k_{eff} (no control) at a specified fraction of the burn time, find the burn time which satisfies the burnup limit as well as the fuel enrichment(s) for which the given k_{eff} is achieved.

The burnup limit must be given for only one material or for the average over a group of materials. A number of different materials or groups may be assigned limits; in this case the burn time will be determined by that material or group which first reaches its burn limit, within a given ϵ .

Adjustment of the fuel charge enrichment is made by a single factor which multiplies the initial enrichment of each fresh fuel charge. A material-dependent modifier may be input, which permits the relative enrichments between materials to be varied as the enrichment factor is changed. This search is terminated when a second value of ϵ is satisfied.

When both the enrichment and burnup convergence criteria are operative in an equilibrium problem, this search determines the infinite-time operating conditions for a reactor running on a fixed fuel-management pattern. In a nonequilibrium problem, the fresh fuel enrichment and burn time will be found for each separate burn step. Fuel-management patterns, burnup limits, control fractions, etc., may be modified at the end of given burn steps.

When only the enrichment convergence is operative the search may be used for approach-to-equilibrium problems with fixed burn time or for nonequilibrium problems in which fuel burnup is not limiting on burn time. When only the burnup convergence is operative, it may be used for nonequilibrium, burnup-limited problems.

2. Given the unpoisoned static k_{eff} at the beginning and end of the burn step, find the fuel charge enrichment for which the initial k_{eff} is satisfied and the burn time at which the final k_{eff} is reached. This search is intended primarily for nonequilibrium, Δk -limited problems.

All the above search alternatives may be run with or without movable control materials. A controlled problem performs control-density searches at each of the user-specified substeps within the burn step, such that the reactor is critical at these time points.

2. Nuclear Data

a. Cross-section Measurements (C. E. Crouthamel and N. Dudey)

(i) Monoenergetic Neutron Cross-section Measurements

Last Reported: ANL-7267, pp. 27-28 (Oct 1966).

Neutron radiative capture cross sections are being measured as a function of neutron energy between about 100 keV and 2 MeV. Monoenergetic neutrons are produced by bombarding lithium targets with protons from the Van de Graaff accelerator. Capture cross sections are needed to evaluate the behavior of materials proposed for structural or control use in a fast reactor. The data are also useful in testing theories of nuclear reactions.

Two series of Van de Graaff irradiations totaling 80 hr of bombardment time have been concluded. Irradiations at neutron energies between 100 keV and 1.5 MeV were carried out with 15 Y_2O_3 pellets, 17 vanadium metal foils, 15 RbNO_3 pellets, and 8 $\text{Sr}(\text{NO}_3)_2$ pellets. The neutron flux in each irradiation was monitored with a ^{235}U fission counter; in addition, gold monitor foils were used in front and back of the yttrium and strontium targets.

Each irradiated sample was counted by detection of its characteristic radiation by means of a $\text{Ge}(\text{Li})$ detector and a $\text{NaI}(\text{Tl})$ γ - γ coincidence spectrometer. An intensive program has been completed for accurate calibration of all gamma-ray detectors. Calibrated standards have been obtained from three independent sources, and where possible have been confirmed by γ - γ coincidence counting methods. Both point sources and sources prepared by uniformly distributing standardized solutions throughout a pellet have been prepared and used to calibrate detector efficiencies. A 4π β - γ coincidence system is being assembled to assist in the calibration of counters.

Neutron energies, energy spreads, and fluxes have been determined for all irradiated targets. Preliminary cross sections for the $^{89}\text{Y}(n, \gamma)^{90\text{m}}\text{Y}$ reaction have been calculated and are presented in Table III.A.1. Data collection has been completed for the reactions $^{51}\text{V}(n, \gamma)^{52}\text{V}$ and $^{85}\text{Rb}(n, \gamma)^{86\text{m}}\text{Rb}$, but calculations of cross sections are still in progress. A computer program for cross-section calculations is being written, and final cross-section calculations are awaiting its completion. One attempt to measure the $^{87}\text{Rb}(n, \gamma)^{88}\text{Rb}$ cross section by gamma counting was unsuccessful because of insufficient activity. A method of calibration utilizing 4π β counting and γ - γ spectrometry is being investigated.

TABLE III.A.1. Capture Cross Section for the
 $^{89}\text{Y}(n,\gamma)^{90\text{m}}\text{Y}$ Reaction

Neutron Energy (MeV)	Cross Section (mb)	Neutron Energy (MeV)	Cross Section (mb)
0.262	0.293	0.852	0.463
0.318	0.247	0.895	0.490
0.433	0.340	0.985	0.310
0.478	0.326	1.069	0.286
0.513	0.371	1.191	0.267
0.642	0.446	1.240	0.277
0.688	0.431	1.461	0.256
0.791	0.505		

(ii) Integral Cross-section Measurements

Last Reported: ANL-7105, p. 40 (Sept 1965).

In evaluating the long-term behavior of reactors it is necessary to know fission and capture cross sections of the uranium and plutonium isotopes which build up through successive captures in the primary fuel isotopes. One method of obtaining these data is by measurement of the capture-to-fission ratios of an isotope as a function of position in a fast reactor. A program is currently being carried out to determine these ratios in the core and blankets of EBR-II. The behavior of structural materials is also of interest, and will be examined by determination of long-lived (n,p) and (n, α) reaction products. The first irradiation in EBR-II has been completed, and the 65 samples of ^{239}Pu , ^{240}Pu , ^{242}Pu , ^{233}U , ^{235}U , and ^{238}U have been shipped from Idaho to Illinois. Scheduling has been arranged for decladding in the senior cave facility of the Remote Control Division.

Analytical separation schemes for (n, γ), (n,f), and, where possible, (n,2n) products from these isotopes have been developed. Analytical schemes are being devised for separation of long-lived (n,p) and (n, α) products in the stainless steel spacers used in the subassemblies. Calculations have been completed for expected yields of these reactions and for the amounts of hydrogen and helium that will be produced by these reactions.

Calibrations of a high-resolution Ge(Li) detector for gross analysis of fission products has been completed. A computer code for analysis of these very complex gamma-ray spectra is being prepared by modification of a code obtained from Lawrence Radiation Laboratory.

(iii) Low-mass Fast-neutron Fission Yields. A program has been initiated to study the production of low-mass charged particles in fission. The particles of special interest to the fast reactor programs are tritons, protons, and alpha particles. Tritium represents a bioradiological disposal problem which must be handled by the fuel-reprocessing scheme. Hydrogen and helium isotopes are of concern because of their interaction with the fuel matrix and fuel cladding materials. The emission of low-mass particles in fission is also of fundamental interest to nuclear theory. The light particles are thought to be released in the space between the two separating fragments very nearly at the time of the actual scission of the fissioning nucleus. A review of all available experimental and theoretical information concerning low-mass fission yields has been conducted, and this survey has been used to extrapolate tritium, helium, and hydrogen yields for nuclei and neutron energies encountered in fast reactors. Experimental measurements of these yields will be used to determine the validity of the calculations and, in addition, can be expected to provide information concerning the configuration of the fissioning nucleus at scission, properties of the neck at fission, and nuclear charge and mass distributions.

The method for the determination of fast-fission yields of the low-mass particles will involve bombardment of fissionable targets with monoenergetic neutrons from the Argonne 4-MeV Van de Graaff accelerator. The energy and angular distributions of the emitted particles will be measured, and the mass and charge determined by means of an electronic particle-identification technique employing a two-detector telescope and associated electronics. The yield of each particle X will be determined by requiring a coincidence between X and a fission fragment. Spontaneously fissioning ^{252}Cf will be used as the reference and calibration standard.

A particle-identification unit and calibration pulser have been received from ORTEC, Inc. Equipment purchases and design of the electronics configuration for the complete system are being discussed with members of the Electronics Division.

Price quotations for the additional electronic components required to supplement the particle identifier have been received, and the equipment will be ordered when funds become available. Close and continual cooperation with the Electronics Division is expected to be maintained throughout the assembly and checkout of the system.

b. Cross-section Measurements (A. B. Smith)

(i) Fast-neutron Cross Sections

Last Reported: ANL-7399, pp. 111-112 (Nov 1967).

(a) Elastic Neutron Scattering and Polarization Measurement. The program of neutron scattering and polarization measurements

is continuing in line with the original aim of determining accurate parameters for an optical model so that cross sections for unmeasured nuclides can be obtained by interpolation between measured values. At present, data for Al, S, Ni, Lu, Se, Y, Mo, and Sn are partly or completely processed. These data are corrected for effects due to multiple scattering, sample attenuation, and finite detector solid angle. Measurements were taken at 8 angles from 20 to 155° and at incident neutron energy intervals of ~10 keV. The neutron-energy resolution was ~20 keV, which is narrow enough so that structure is easily evident in the cross section and polarization of some elements. The experimental results now cover the energy range from ~400 to ~1100 keV. We plan to extend this to ~1500 keV for the elements mentioned above. In addition, we plan complete measurements for a number of other elements in order adequately to investigate the influence of the S-wave and P-wave strength functions and to cover regions of spherical and deformed nuclei. Optical-model calculations are in progress to determine the best parameter fit to the data, the objective being to achieve a simultaneous fit to the scattering cross section and polarization. For this purpose, the data are averaged over 200-keV intervals to smooth out the local cross-section fluctuation (which are not included in the optical model) and to improve the statistical nature of the data for a more accurate comparison.

(b) Nuclear Reaction Theory. The level matrix inversion program for statistical reaction model calculations has been modified and adapted for the IBM-360 computer. Further modification for increasing the maximum size of the matrix and, hence, of the statistical sample are underway. Papers on the distributions and correlations of resonance pole parameters and on the role of classical states in the measurement process are in preparation.

(ii) Fission Process Data

Last Reported: ANL-7391, p. 124 (Oct 1967).

(a) Correlation of Mass, Energy, and Angle in Neutron-induced Fission

1. ^{235}U . Since the ground state of ^{235}U has negative parity, thermal-neutron-induced fission must proceed through a saddle-point state of negative parity. However, optical-model transmission coefficients indicate that for neutron energies greater than 100 keV fission must proceed primarily through levels of positive parity. Experiments have been carried out to see if this change of parity has any effect on the mass-energy-angle distribution of fragments in fission. Measurements have been made for thermal, 150-keV, and 500-keV neutron-induced fission. The data are now being processed.

2. ^{238}U . Measurements of fragment distributions for the 5.6-MeV neutron-induced fission of ^{238}U have been completed. About 15,000 fission events have been analyzed at each of three angles. The gross angular distribution is given by the following table:

Average Angle of Fragment Axis to Incident Neutron Direction, degrees	8.2	45.4	90.4
$P(\theta)/P(90)$ (Intensity Relative to that at 90°)	1.203	1.099	1.000

Data processing is continuing.

3. ^{239}Pu . Fission-fragment angular distributions were measured for the fission of ^{239}Pu at incident neutron energies of 475, 600, 800, 900, and 1200 keV. The fragments were detected by surface-barrier detectors at six angles; 3.7, 24.0, 37.2, 54.7, 72.3, and 90.0° . The statistical accuracy is $\sim 1.5\%$. The data are now being processed.

(iii) Instrumentation and Digital Techniques (A. B. Smith)

Last Reported: ANL-7399, p. 113 (Nov 1967).

(a) Digital Instrumentation. The digital circuitry has been reorganized and tested. A new interface for controlling both digital-to-analog converters was debugged and placed into operation. Both computers can now be selected and controlled independently of the regular interface which is used for the analog-to-digital converters.

One digital-to-analog converter will still be used to control the energy of the Van de Graaff; however, the circuitry has been modified to increase the precision of the control by utilizing the full range of the converter. Previously, 4096 energy steps, essentially from zero proton energy to the maximum of about 3.5 MeV, or slightly less than 1-keV increments, were available. The new circuitry permits proton energy steps smaller than 0.5 kV from an initial proton energy of about 1.7 MeV (somewhat below the $\text{Li}(p,n)$ reaction threshold of 1.883 MeV). This modification should prove more efficient since it places the full operating range of the computer control in the usable range of the Van de Graaff. The above circuit changes will require some modification in software. The new interface also provides 6 additional select codes with corresponding voltage-step outputs for computer control of external instrumentation (such as the auxiliary display mentioned in the last monthly report).

B. Reactor Fuels and Materials Development

1. Fuels and Claddings--Behavior of Reactor Materials

a. Irradiation Behavior of Advanced Ceramic Materials (L. C. Michels)

Last Reported: ANL-7399, pp. 113-114 (Nov 1967).

(i) Irradiation of High-temperature Materials. A series of irradiations of various ceramic fuels of interest for LMFBR's is in progress. The materials include UC-20 w/o PuC and US. All materials are clad in Nb-1 w/o Zr. They are being irradiated in instrumented, temperature-controlled capsules in MTR. A summary of the irradiations in progress is shown in Table III.B.1.

TABLE III.B.1. Status of Ceramic-fuel^a Irradiations in Progress in MTR

Capsule No.	Specimen No.	Design Parameters		Operating Conditions		
		Fuel Composition (w/o)	Effective Density (%)	Max Cladding Temp (°C)	Estimated Burnup to Date	
					a/o (U+Pu)	fiss/cc x 10 ⁻²⁰ ^b
56-11	MV-2	UC-20 PuC	79	470	9.1	24
56-8	MV-3	UC-20 PuC	81	715	9.5	26
56-8	MV-5	UC-20 PuC	80	705	9.2	25
56-11	MV-6	UC-20 PuC	80	480	9.6	25
56-8	S-7	US	80	535	7.7	15
56-8	S-8	US	89	725	9.8	20
56-8	S-9	US	76 ^c	750	9.8	19
56-8	S-10	US	91	690	9.8	22
56-11	S-15	US	82	380	6.6	13
56-11	S-16	US	90	510	8.9	20
56-11	S-17	US	88	500	7.5	17
56-11	S-18	US	77 ^c	610	9.4	18

^aAll cladding of fuel is Nb-1 w/o Zr alloy of 0.281-in. OD and 0.012-in. thickness.

^bBased on effective density.

^cPartially annular pellets with center thermocouple.

Postirradiation examination of fuel specimens irradiated in capsules ANL-56-12 and ANL-56-13 is continuing. Measurements to date indicate that no changes occurred in specimen diameter or volume exceeding the limits of error of the measurements during irradiation.

2. Fuels and Claddings--Chemistry of Irradiated Materials

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

Last Reported: ANL-7399, pp. 119-121 (Nov 1967).

(i) Shielded Fuel-evaluation Facility. Construction and testing of equipment for a shielded fuel-evaluation facility for the study of irradiated fuels is continuing.

Previous tests of a Leitz metallograph (Model MM5 RT), which will be installed in an inert-atmosphere enclosure, showed minor leaks in various fittings. Representatives of the German manufacturer were unable to eliminate these leaks completely; the manufacturer is designing and building a different type of seal to correct this difficulty.

Design work has been completed on the shielding for the metallograph and is proceeding on the enclosure for this instrument. Footings have been poured for the shielding.

Installation of the jib crane was scheduled for the latter part of December, and installation of manipulators and periscope will begin when the work on the jib crane is completed. Installation of the pneumatic tube is awaiting the receipt of the motor-drive units.

When this work is completed, the interior of the cell will be painted. Installation of the wet-chemistry box and the metallograph will complete the placement of the major equipment items.

(ii) Laser Sampling System. Attempts are being made to determine the effect of different combinations of optical components in a microscope-laser system on the size and shape of craters formed in UO_2 . As was reported in ANL-7399, p. 120, some deep craters of unusual shape were found in UO_2 . It has now been determined that the shape and size of the craters are highly dependent on the optical system and that the shapes reported previously were atypical. The "normal" shape in UO_2 seems to be approximately conical.

(iii) Fuel Specimen Preparation and Sampling. Conditions for cathodic etching of sintered UO_2 by ion bombardment have been tentatively established as an argon pressure of 2.5×10^{-5} Torr at 2.8 kV with a current of 0.3 to 0.5 mA. The size of the area etched is dependent on the argon pressure as well as the size of the opening in the Teflon mask that is used to protect the plastic mounting material.

Several samples of Type 304 stainless steel rod were mounted in copper-Bakelite for use in establishing conditions for cathodic etching of stainless steel. Heat generated in cutting these samples apparently affected the structure of the samples at the cut surface. No grain structure was developed by the etching even though attack of the surface was evident. New samples are being prepared.

b. Postirradiation Studies of Reactor Fuels (M. D. Adams)

Last Reported: ANL-7399, pp. 121-123 (Nov 1967).

(i) Laser Sampling of Irradiated UO_2 Fuel Pin. Twenty-five laser-evaporated samples were taken across the radius of a 4-mm-dia irradiated UO_2 fuel pin (6.8 a/o burnup, 13% enriched). Results of the

examination of another cross section of this fuel pin were reported previously (see Progress Report for August 1967, ANL-7371, pp. 87-88). Gamma spectrometry of the current samples was performed with a NaI(Tl) detector. Although the results have not been completely analyzed, the gamma activities found corresponded, in general, to those of the first cross section that was examined. However, in specimens taken from the interface between the UO_2 fuel and the Type 304 stainless steel cladding, a substantial amount of the activity was from ^{125}Sb . The presence of antimony in this region has not been anticipated, nor was it found by electron microprobe analysis of the same specimen.

These samples have been dissolved and a mass-spectrometric analysis will be attempted even though the weight of each sample is only a few nanograms.

3. Fuels and Claddings--Thermodynamics of Fuel Materials

a. Vapor Species Partial Pressures in the Ternary U-Pu-C System (P. E. Blackburn and J. W. Reishus)

Last Reported: ANL-7382, pp. 103-104 (Sept 1967).

The objective of this study is to measure the partial pressures of vapor species over Pu-C and U-Pu-C compositions. A Bendix time-of-flight mass spectrometer has been coupled to a glovebox for this purpose. Initial work will be with the Pu-C system. In this system there are, at selected temperatures, four thermodynamically invariant two-phase fields consisting of solids only: $\text{PuC-Pu}_2\text{C}_3$, $\text{Pu}_2\text{C}_3\text{-C}$, $\text{Pu}_2\text{C}_3\text{-PuC}_2$, and $\text{PuC}_2\text{-C}$. Olsen and Mulford* at Los Alamos have conducted mass-spectrometric studies of the vapors over the $\text{PuC-Pu}_2\text{C}_3$ and $\text{Pu}_2\text{C}_3\text{-C}$ mixtures. Plutonium was the predominant species observed over both mixtures, and about 0.1% of the vapor over the $\text{Pu}_2\text{C}_3\text{-C}$ mixture was the species $\text{PuC}_2(\text{g})$. Our research efforts will be aimed primarily at the $\text{Pu}_2\text{C}_3\text{-PuC}_2$ and $\text{PuC}_2\text{-C}$ fields; special attention will be devoted to looking for $\text{PuC}_2(\text{g})$ and $\text{PuC}(\text{g})$. Pu-C samples of various compositions from $\text{C/Pu} = 1\text{-}2$ are being prepared, and mass spectrometric studies will be started as soon as the samples are characterized.

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

a. Development of Nondestructive Testing Techniques (H. Berger)

Last Reported: ANL-7399, pp. 123-124 (Nov 1967).

(i) Ultrasonic Measurement of Elastic Moduli. The elastic modulus of Type 316 stainless steel** has been calculated from room temperature to 1200°C .

*Olsen, W. M., and Mulford, R. N. R., LA-DC-8012 (1967).

**From Chicago Steel Service Company, heat No. 0807838.

Two samples of the stainless steel were fabricated and tested. The elastic moduli calculated from this test differed from that of Type 304 stainless steel by an amount within the limits of experimental error. A report on the results is to be written.

A tungsten sample is now being installed in the furnace. Decontamination of all furnace parts is necessary before the furnace can be taken to temperatures over 1400°C.

(ii) Neutron Studies. Work continues with the Am-Cm-Be neutron source, which yields about 10^7 n/sec. One purpose of this work is to check experimentally the predictions made by theoretical work at ANL-Idaho (D. C. Cutforth). A BeO stack, 1 m high and approximately of 38-cm horizontal radius, has been constructed in the M3 hot cell. The source is positioned at the center in a vertical inlet, 2.5 cm square. Initial measurements of thermal-neutron flux indicate a peak of approximately 0.9×10^5 n/cm²-sec (compared with 1.25×10^5 in H₂O) and a cadmium ratio for gold foil of 2.7. This preliminary result contradicts the theoretical predictions, based on computer calculations, that BeO would be superior to H₂O for this purpose. However, strange flux distributions are being found and the measurements are being repeated. It is also recognized that the computer result makes use of an infinite BeO moderator thickness (50 cm).

Work is in progress to develop a standard test system for comparing the neutron-radiographic capabilities of different facilities. The comparison also will be, at the same time, a measure of: beam collimation, neutron spectrum, gamma-ray contamination, and detector characteristics. After development at Argonne, with utilization of the Juggernaut reactor and small-source facilities, a test system is to be circulated to other laboratories for comparison and for correlation.

(iii) Infrared and Thermal Techniques. A pulsed ruby laser, which is to replace the xenon flash tube as the source of thermal energy for the pulse-method diffusivity and conductivity measurements, has been installed and aligned. Preliminary room-temperature experiments indicate that there are a number of difficulties inherent with the use of a laser.

Lasing action is not reproducible at low voltages; at high voltages the front surfaces of some materials undergo excessive temperature changes. In such cases, filters will be used to decrease the energy to acceptable levels. There were severe heat losses from a 0.366-cm-dia specimen when the diameter of the laser beam was reduced to 0.366 cm, or when the beam was a full 0.951-cm diameter. Heat was at first lost through the Lavite specimen holder until a lens was introduced into the system so as to enlarge the laser beam to 2.5-cm diameter. Complete irradiation of the front of the holder with thermal energy eliminated the heat loss problem.

(iv) Passive Ultrasonic Techniques. The fabrication of Type 304 stainless steel housings for ultrasonic acoustic emission tests in sodium is complete. Methods of bonding piezoelectric elements with gold electrodes to the housings and to porous tungsten backings are being investigated.

C. Engineering Development--Research and Development

1. Development of Master-Slave Manipulator Systems (R. C. Goertz)

a. Development of Manipulator Systems

Last Reported: ANL-7399, pp. 126-127 (Nov 1967).

(i) Electric Master-Slave Manipulator, Mark E4A. The filtered-gas-cooling system for the slave arm is being fabricated and parts should be completed during January.

Tests were conducted with a single motor using argon as the cooling medium, similar to those (see Progress Report for October 1967, ANL-7391, p. 140) with air as the cooling medium. The motor (Diehl 25-frame) was excited with 150 V, 60 Hz, on each field, and the stator copper temperature was held so as not to exceed 170°C. Without fins, 34 cfm of argon was required; with fins, 11 cfm was required. The corresponding previous results with air were 24 and 8 cfm. It may be noted that for both cases the volume required for the unfinned motor is approximately three times that of the finned motor. Also, the additional volume of gas required when using argon is increased by 40%.

Cyclic testing of the brake system was continued. There has been a total of 2.3 million cycles on the brake system. Modifications previously reported have resulted in 1.5 million cycles to the new pawl with no significant wear and 1.8 million cycles to the larger-diameter pin installed. The solenoid had to be replaced at 1.9 million cycles because of reduction in the magnetic air gap resulting in the plunger sticking some of the time. The reduction in the magnetic gap is due to mushrooming of the pole faces. Presently the required minimum voltage to operate the solenoid is 78 V, whereas we operate the system at 115 V. A different coil for the solenoid can be specified which would still allow operation at 115 V while having a higher minimum voltage and therefore a reduction in force. This reduction in pull-in force should minimize the mushrooming of the pole faces.

(ii) Components and Tools

(a) Low-inertia DC Motors for Manipulators. Different diameters, lengths, etc., are being studied in conjunction with various gearing and drum systems for a 100- to 150-lb manipulator.

The dc motor system seems to have many advantages over the ac motor system. We are concentrating our effort on the dc approach. All material, except the permanent magnets, is available to

build an experimental motor. Electric fields will be used for the first group of tests. Parts of the first motor will be fabricated during the month of January.

b. Study of Remote Handling for LMFBR Facilities

(i) Objective. A program is underway to investigate and review the needs for further development of general-purpose remote manipulation and viewing equipment.

(ii) Scope. The concepts, plans, and operating philosophies of the remote-handling and repair systems of existing and planned LMFBR facilities are to be analyzed. What equipment should be developed and used to permit the operation of these plants at the highest plant factor, efficiency, and reliability, consistent with the lowest risk to personnel safety and program schedule, is to be determined. The findings of the study and recommendations for long-range development of remote-handling systems for use in these and future facilities will be reported to the AEC.

(iii) Progress. Members of the study group met with representatives of the design group of the FFTF at Pacific Northwest Laboratory. The study group was briefed on the fuel-handling, examination, and remote-repair facilities planned for this reactor. The talks also covered operating philosophies and particularly an exchange of philosophies on methods of remote repair.

Our immediate observations are that there are quite a number of places where good electric master-slave manipulators and slave TV would be very helpful to the design and eventual operation. The Mark E4A would be very useful, but several of the people working on remote-handling problems were not fully aware of the present state of development of the Mark E4A.

It is quite clear, however, that FFTF is in need of electric master-slaves with high load capacities and somewhat higher reliability. Integral slave TV is also needed. These observations have confirmed and stimulated our program on these systems, especially as regards the higher loads and very high reliability aspects.

There seemed to be considerable discrepancy in concepts of the reliability of specific-purpose handling equipment, such as fuel-unloading machinery, versus that of well-developed general-purpose manipulation. A brief seminar was given by the study group to acquaint members of the design group with the current state of development of the Mark-E4 electronic master-slave manipulator.

Contact was also made to arrange a similar meeting with members of the design group for the SEFOR reactor at the General Electric Advanced Products Division. They indicated that the meeting would have to be deferred until late January 1968 because of vacation schedules and other commitments.

2. Heat Transfer, Fluid Flow, and Mechanics of Materials

a. High-temperature Boiling Sodium Experiments (J. V. Tokar)

Last Reported: ANL-7399, p. 128 (Nov 1967).

(i) Niobium-1% Zirconium Loop. Modifications and repairs of the power feed-throughs, shutter-drive liquid-nitrogen system, and the data-acquisition system are nearly complete.

In anticipation of replacement of the thermal-radiation boiler with an electron-bombardment heater within the next 6 months, Nb-1% Zr welding experiments are being made. The first attempts have shown that the field welder produces good butt welds with 1/2-in. tubing. Commercially pure argon gas used for purge during welding caused weld contamination as follows: 410-530 ppm O₂, 10-16 ppm H₂, and 110-120 ppm N₂. These values are not appreciably higher than the original material specifications for the Nb-1% Zr tubing used in the welding experiments. In further tests processed argon from the purifying train is connected to the Nb-1% Zr loop. This is expected to reduce contamination to acceptable levels.

b. Heat Transfer to Liquid-metal Heat Exchangers (R. P. Stein)

Last Reported: ANL-7399, pp. 128-129 (Nov 1967).

Construction continues on the loop with which experiments for simple simulations of liquid-metal-heated steam generators will be run.

Several empirical correlations based on actual experimental data have been found for heat transfer during nucleate boiling of Freon 11, the fluid chosen to simulate water in the experimental program. These correlations are of the same form as those currently used in heat-transfer-design procedures for actual LMFBF-type sodium-heated steam generators. The availability of these correlations will allow comparison of heat-transfer prediction of the design methods with experimental data obtained from the mercury-freon simulation.

An interim report* is being expanded to include a more complete survey of LMFBF-type steam-generator heat-transfer design methods.

*Lo, R. K., A Review of Methods Used to Predict the Heat Transfer Area of Liquid-metal Heated Steam Generators, RE-67-37 (July 1967).

c. Fog-flow Heat Transfer and Fluid Flow (R. P. Stein)

Last Reported: ANL-7391, p. 142 (Oct 1967).

On the basis of the literature survey, a research program has been defined. The initial study will test the utility of a Fick's-law formulation for describing the turbulent diffusion of fine liquid droplets suspended in a gaseous duct flow.

An apparatus has been designed to establish a fully developed turbulent gas flow in which the droplet experiments are to be carried out and to enable detailed studies of droplet flow over a considerable length of duct.

The test section will consist of an 8-in.-dia, 40-ft-long column of cast acrylic-resin tubing with allowance for air filtration, flow straightening, development of fully developed turbulence, droplet injection, and droplet-flow measurements. Plans call for the measurement of gas-phase and droplet-flow properties.

(i) Gas-phase Properties. A hot-wire anemometry system will be used to establish the (time) mean droplet-free gas-flow distribution and some statistical properties of the turbulence.

(ii) Droplet-flow Properties. A hot-wire anemometry system will measure profiles of droplet number flux (droplets/cm²-sec) by counting the number of droplets impinging on the hot-wire sensor. An isokinetic sampling probe will measure profiles of liquid-mass flow (g/cm²-sec). A fiber-optics technique, involving attenuation of light as droplets pass through the sensing region, will measure local liquid-concentration profiles.

These measurements will be interpreted for information on (a) the local effective convective-liquid velocity and (b) mean droplet diameter. Initially, the liquid-concentration data will be compared with predictions based on a simple Fick's law to determine the range of operating conditions for which the assumption of a constant effective eddy diffusivity is reasonable. Effective values of the diffusivity will be determined.

The theoretical aspects of this problem are also being considered, including the relation between the statistical properties of the gas phase and those of the droplets. Experiments related to droplet deposition at the duct walls are also being considered.

3. Liquid-metal Linear Generator

a. Effect of Interfacial Transport Process on Condensation (J. H. Linehan)

Last Reported: ANL-7371, p. 92 (Aug 1967).

Data for both the air-water and steam-water experiments have been reduced. For the air-water experiments, the models developed to describe the various conditions of gas-liquid flow (turbulent-turbulent, etc.) have indicated a strong effect of the interfacial condition. For example, it has been accepted in the literature that the transition or critical Reynolds number in film flow ($Re = \text{flow rate per unit length of wetted perimeter divided by the dynamic viscosity}$) is between 250 and 500. The air-water experiments indicate that, in this range, the air velocity determines whether the behavior in the film is laminar or turbulent. For low gas velocities, the surface interfacial disturbances are minimized and the film displays laminar characteristics; the converse is noted at higher gas velocity. In general, the models successfully describe the variation of film thickness with axial length from the injector. (The models are limited to nonentraining flows.)

Steam-water experiments were performed in an adiabatic test section, i.e., the condensation results from the initial subcooling of the injected water film. The main interest has been in the turbulent-turbulent flow regime. The existing correlations for condensing heat transfer underestimate the experimentally measured condensation rate by a factor of 3 to 5. This is attributable to the fact that the correlations emphasize the wall region in describing the resistance to heat transfer. But in the present experiments, no heat is transferred from the condenser walls; therefore, the major resistance to heat transfer must be associated with the overall characteristics of the film itself. A model describing the overall characteristics has been developed and preliminary tests indicate good agreement with the data.

4. Engineering Mechanics

a. Anisotropic Thermoelasticity Studies (R. A. Valentin)

Last Reported: ANL-7391, pp. 143-144 (Oct 1967).

In an attempt to simplify algebraic complexities of stress-function formulations such as those of Singh,* an alternative set of stress functions has been derived. These functions, although closely related to

*Singh, A., Axisymmetrical Thermal Stresses in Transversely Isotropic Bodies, Arch. Mech. Stoj., 12, 287-304 (1960).

those used in completed sample problems, do provide some simplifications; however, the appearance of complex separation constants for certain material-property combinations seems to be intrinsic in the class of problems being studied, and a solution method which avoids this difficulty has not proved possible. Attempts to derive functions analogous to the classical Papkovitch-Neuber isotropic formulation have not met with any success and have been discontinued.

b. Core Structural Dynamics (M. W. Wambsganss)

Last Reported: ANL-7399, pp. 129-130 (Nov 1967).

To make the test specimen more flexible, the brass rod that had been mounted in a clamped-clamped configuration in a previous parallel-flow-induced vibration test was mounted in a pinned-pinned configuration. Rod displacements and pressures on the surface of the test section were measured for flow velocities of 0-20 ft/sec. Significant rod displacements (2-6 mils, rms) were obtained. These greater displacements can be attributed to increased energy in the pressure fluctuations (as exhibited by the power spectral density distribution) in the frequency band near the fundamental natural frequency of the rod.

The rms displacements increase uniformly with flow velocity according to a power-function relationship. The displacement and pressure data are being further processed to construct frequency-response functions for the rod, assuming a single-mode response and neglecting the convection of the pressure field.

D. Chemistry and Chemical Separations

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

At the request of the AEC, the fluoride volatility development program on the reprocessing of light water reactor fuels is being discontinued, and the program is being reoriented toward an evaluation of the potential of the process for fast breeder reactor fuel application. The work reported below represents, in some instances, work which was done prior to reorientation of the program. Much of this work, however, is applicable to both fast breeder reactor and light water reactor fuels.

a. Fluorination Chemistry and Procedures (M. J. Steindler)

Last Reported: ANL-7391, pp. 145-147 (Oct 1967).

(i) Fluorination of UO_2 - PuO_2 -EP Pellets--Bench-scale Experiments. The fluorination of UO_2 - PuO_2 pellets containing inactive fission products is being studied in a 2-in.-dia fluidized-bed reactor to

define conditions that ensure minimum retention of plutonium in the final bed of alumina particles. In the process, the pellets are first oxidized to U_3O_8 - PuO_2 fines by reaction with oxygen. Next, the uranium oxide is converted to volatile UF_6 by BrF_5 , which converts PuO_2 to nonvolatile PuF_4 . In the subsequent recycle-fluorination step, plutonium is recovered as volatile PuF_6 by the reaction of PuF_4 with fluorine.

Runs Purse-25 and Purse-26, the final experiments in these studies, have been completed. The charge for each consisted of a 2-in.-deep bed (~650 g) of UO_2 - PuO_2 -FP oxide pellets plus 0.16 g RbF , 0.6 g CsF , and 0.5 g NpO_2 , which were added directly to the fluid-bed material.

In Purse-25, the initial alumina bed consisted of 990 g of -40 +170 mesh T-61 alumina and 67 g of AlF_3 fines to evaluate the effect of this bed composition on plutonium retention in the fines. The AlF_3 fines were added to the bed so that the fluid-bed filters would be coated with these fines at the start of fluidization rather than being coated by plutonium-bearing fines formed during the oxidation step; the AlF_3 fines were also expected to provide a vehicle for returning these solids from the filter to the bed when the filters were blown back.

In Purse-25, oxidation with 20 v/o oxygen at 450°C for 4 hr was followed by fluorination with 9-10 v/o BrF_5 at 300°C for 2 hr and fluorination with fluorine (recycle-fluorination) at temperatures of 300 to 550°C for 11 hr. The plutonium content of the final alumina bed, 0.014 w/o, represented 5.3% of the plutonium charged. This concentration of plutonium is about twice the amount found under similar conditions but in the absence of added AlF_3 fines. After the bed was dumped, 15.5 g fines were recovered from the reactor walls and filter, and a 15.7-g agglomerate was recovered from the bed; they contained 3.9 and 0.7%, respectively, of the plutonium charged. Therefore, the results obtained in this run indicate that the addition of AlF_3 fines is not desirable because of an excessive amount of plutonium retained in the fines and the bed.

Complete separation of uranium and neptunium from plutonium by BrF_5 fluorination would be highly desirable for a fluid-bed fluoride volatility process. To accomplish such a separation, a second BrF_5 step (at 550°C for 1.9 hr) was performed in Purse-26 before the fluorine step. Except for this additional fluorination step, the charge and process steps for runs Purse-25 and Purse-26 were similar. After the high-temperature BrF_5 step, however, 0.24% of the uranium and 20% of the neptunium charged remained in the alumina bed. Moreover, fluorination of neptunium and plutonium in the fluorine step were adversely affected. Therefore, high-temperature fluorination with BrF_5 is not recommended.

b. Neptunium Fluoride Chemistry (L. E. Trevorrow)

Last Reported: ANL-7399, pp. 131-132 (Nov 1967).

Experimental studies are in progress with the aim of establishing the chemical properties of neptunium fluorides that could be applied to the separation and recovery of ^{237}Np from nuclear fuel by fluoride volatility techniques. Separation of neptunium from uranium and plutonium is important from the standpoint of product purity; recovery of neptunium as a by-product may also prove to be economically desirable. The removal of NpF_6 from a gas stream by reaction with particulate NaF contained in a fixed bed is a potentially attractive method for separating neptunium from uranium and plutonium. This reaction produces a violet-colored complex tentatively identified as $3\text{NaF} \cdot \text{NpF}_5$. In preparation for future work to measure the equilibrium constants of the reaction, additional experimental work has been performed to obtain information about the stoichiometry of the reaction and the identity of the products.

Reactions of three samples of NaF with gaseous NpF_6 were carried out at 250°C . The pressure changes accompanying reaction were measured, as well as the final vapor density of the gas phase and the weight increments of previously weighed samples of NaF . The weight changes measured confirmed the earlier finding that the ratio of sodium to neptunium in the solid product is 3:1. The average valence state of the neptunium in the solid product was 4.6 in each sample. The derived stoichiometric coefficients are about $1/2$ for fluorine and about 3 for NaF . These values are all consistent with the representation of the reaction by the equation

c. Engineering-scale Development for UO_2 - PuO_2 Fuel
(N. M. Levitz)

Last Reported: ANL-7399, pp. 132-134 (Nov 1967).

(i) Engineering-scale Alpha Facility. Additional data have been obtained for a series of campaign-type fluorination experiments* intended to demonstrate the feasibility of producing and transporting PuF_6 and collecting it quantitatively. Three campaigns have been completed, consisting of twelve fluorination runs in which a total of 2300 g of PuF_4 was fluorinated in a 3-in.-dia fluorinator. Each campaign consisted of three successive fluorination runs, followed by a cleanup-fluorination to recover small amounts of PuF_4 deposited in lines and equipment. In each fluorination, ~ 200 g of plutonium (~ 260 g of -325 mesh PuF_4) was charged. A single bed consisting of 6500 g of 48-100 mesh prefluorinated alumina was used in each campaign.

* See Progress Reports for September 1967, October 1967, November 1967 (ANL-7382, pp. 116-117; ANL-7391, pp. 149-150; ANL-7399, pp. 132-134).

After the alumina and PuF_4 were charged and the bed was brought up to temperature (300°C in Campaign 1, 375°C in Campaign 2, and 450°C in Campaign 3), enough fluorine was added to the gas stream to give a concentration of ~ 90 v/o F_2 . Fluorination was continued while the temperature was increased with time (approximately 25°C every 15 min) to 550°C; the runs were continued at 550°C for a total run time of 5 hr in each run of Campaigns 1 and 2 and for a total run time of 3 hr in each run of Campaign 3.

At the end of each fluorination, the PuF_6 was transferred in an inert-gas stream from cold traps (which had been kept at $\sim -65^\circ\text{C}$) to NaF traps. After the third run in a campaign, a cleanup fluorination of the primary filters was carried out by exposing the filters at 300°C to a flowing fluorine (~ 90 v/o) stream for 2 hr. During the cleanup-fluorination period, other portions of the equipment were also held at 300°C while exposed to ~ 90 v/o fluorine to remove any deposited plutonium.

A plutonium material balance of 101% was obtained for the three fluorination runs of Campaign 2. This value is in the range expected on the basis of earlier sampling and analytical tests. Results also show that (1) greater than 99% of the plutonium charged was volatilized from the reactor and (2) the first of two in-series cold traps removed about 98% of the PuF_6 passing into the cold-trap system.

The percentage of plutonium removed from the fluorinator bed in Campaign 3 was 99.8, compared with 99.9 and 99.7% in the two earlier campaigns. Campaign 3 differed from the two other campaigns in having the highest starting fluorination temperature (450°C) and a shorter fluorination period (3 instead of 5 hr). Preliminary data indicate an overall PuF_6 production rate in Campaign 3 of 4.1 lb/(hr)(sq ft), compared with 2.4 lb/(hr)(sq ft) in the earlier runs. The preliminary value for overall fluorine efficiency of 28% in Campaign 3 is an improvement over the 17 and 22% observed in the other two campaigns.

The holdup of PuF_4 dust in the filter zone of the fluorinator was low in Campaign 3, about 0.7% of the plutonium charged. The holdup of plutonium in the cold traps and transfer lines, as a result of alpha-radiation decomposition of PuF_6 , was also low, about 1.4% of the plutonium charged. In both cases, the plutonium was recovered readily by fluorination of the equipment at 300°C.

Data have been obtained indicative of the corrosion rate of nickel surfaces of the fluorinator during Campaign 1. During this campaign, the bed temperature was in the range 300-550°C for ~ 8.5 hr and at 550°C for 10 hr. The average fluorine concentration was 85% throughout. The basis for calculation was the increase in nickel concentration of the alumina bed during the campaign. The corrosion rate is

about 1 mil/mo of exposure at the operating temperature if it is assumed that the nickel accumulation represents corrosion of the nickel-ball bed support and the 3-in.-dia section of the fluorinator; the rate is about 2.5 mils/mo of exposure at the operating temperature if it is assumed that the nickel represents only corrosion of the fluorinator section.

d. Investigation of Hexafluoride Distillation (N. M. Levitz)

The distillation unit is in a standby condition. This task has been deferred under the reoriented program.

e. Small-scale Studies with Irradiated Fuels (A. A. Chilenskas)

Last Reported: ANL-7382, pp. 117-118 (Sept 1967).

As part of an investigation of the processing of irradiated fuels by fluoride volatility methods, procedures are being developed for removing the reactive components of effluent gas streams. These procedures are intended specifically for use in experimental studies in the hot cell. Several runs were completed in which particulate soda lime and NaF were tested separately for the removal of HF from a flowing stream of oxygen. The 3-in.-dia bed (containing 535 g of sorbent) was heated to 125-150°C prior to introduction of the gas mixture. The soda lime showed a loading at breakthrough of 0.42 g HF/g soda lime and a HF removal efficiency of 99.5%. The NaF was poorer in comparison; it attained a loading of 0.17 g HF/g NaF and a HF removal efficiency of 99%.

g. Ancillary Studies (D. Ramaswami and M. J. Steindler)

Last Reported: ANL-7308, pp. 73-75 (Feb 1967).

This task designation has been cancelled. Work previously included under it is now assigned to other tasks.

2. Closed Cycle Processes--Research and Development--Compact Pyrochemical Processes

a. Process Chemistry of Molten Salt Systems (I. Johnson)

Last Reported: ANL-7399, p. 134 (Nov 1967).

An oxide-reduction experiment was reported previously (ANL-7399) in which U_3O_8 powder was reacted with a 47.5 m/o $MgCl_2$ -47.5 m/o $CaCl_2$ -5 m/o CaF_2 salt mixture. The reaction proceeded rapidly at first and then subsided quickly, suggesting that a film that inhibited further reaction was being formed on the surface of the U_3O_8 particles.

Petrographic analyses of the partially reacted U_3O_8 particles from the experiment have identified the film as magnesium uranate (MgUO_4). These results would suggest that the reaction which takes place is

Another experiment was conducted in which UO_2Cl_2 dissolved in 47.5 m/o MgCl_2 -47.5 m/o CaCl_2 -5 m/o MgF_2 salt was reacted with liquid magnesium. A solid salt-insoluble reaction product was collected and examined by X-ray and petrographic analyses. The results of both analyses showed the product to be UO_2 . These results indicate that the reaction of UO_2Cl_2 with liquid magnesium is probably

An experiment was conducted to determine the temperature of reaction of U_3O_8 with liquid magnesium using a thermal-analysis technique. Magnesium and U_3O_8 powders were charged in a ratio that provided sufficient magnesium to react with all of the oxygen in the U_3O_8 . The powders were mixed together in a drybox and sealed in a vessel under a helium atmosphere. A very rapid exothermic reaction was observed at 725°C . Petrographic analysis of the reaction product indicated the formation of MgUO_4 .

b. Unit Operations Development (R. D. Pierce)

Last Reported: ANL-7382, pp. 122-124 (Sept 1967).

A small experimental program is underway to study the phase-disengagement properties of various agitated metal-salt systems. The initial objective of this study is to provide an index of the relative ease of phase disengagement for the various metal-salt systems of current process interest.

In the procedure used, the metal alloy and salt mixture are stirred together to attain a steady-state dispersion of metal droplets in the salt phase. The melt is then allowed to settle for a fixed period of time (about one minute) and the bulk of the salt phase (~70%) is quickly poured into a receiver. The amount of metal remaining in the salt phase is then determined and compared with the amounts left in the salts in other tests with different salt compositions and alloys. The vessel geometry, mixing conditions, and salt-to-metal volume ratio (2 to 1) have been the same in all tests. The salt and alloy compositions and the temperature were varied.

Results obtained with a Cu-33 w/o Mg alloy and a MgCl_2 -30 m/o NaCl -20 m/o KCl salt mixture (the donor alloy and transport salt used in the salt-transport process) show that excellent phase-disengagement

properties can be achieved with this system through the addition of ~5 w/o MgF_2 to the salt. The observed metal entrainment in the salt phase when MgF_2 was added was lower by a factor of 90 than that observed without MgF_2 . Therefore, it has been recommended that MgF_2 additions be made to this metal-salt system in future reprocessing experiments.

A low-temperature, countercurrent column, operated at 85 to 90°C with Wood's metal (m.p. 71°C) and water as the two phases, has been employed to provide design and operating data for the high-temperature countercurrent extraction column. The low-temperature column has provided basic information on the behavior of liquid-liquid systems that have physical properties similar to those of a salt-metal system (a large density difference and a high interfacial tension). The pressure drop of the continuous phase (water) through a packed section of the column and the limiting flow rates were determined to establish operating conditions for the high-temperature extraction column. Initial tests in the low-temperature column used 3/16-in. metal saddles as the packing material. Recent tests utilized 1/4-in. Raschig rings. The results obtained on flow rates and pressure drops correlated well with data obtained from the high-temperature extraction column using liquid metal and molten salt with 1/4-in. Raschig-ring packing.

c. Materials Testing and Evaluation (R. D. Pierce)

Last Reported: ANL-7391, pp. 153-154 (Oct 1967).

Previous tests of CB-30 stainless steel crucibles (composition: 10% Cr, 2% Si, 0.17% C, <0.1% Ni, balance Fe) have shown that this material has good corrosion resistance to the Cu-Mg/halide salt systems employed in pyrochemical processing at 700°C. However, tests conducted with uranium present in the Cu-Mg alloy indicated that uranium was being lost from solution in amounts up to 25% of the uranium charged in 200 hr of contact with the CB-30 crucibles.

Because of the uranium losses experienced in these tests, it was decided that a solution-stability test should be performed using a plutonium-bearing alloy and CB-30 stainless steel before any further consideration was given to CB-30 as a candidate container material.

An experiment was conducted in which a CB-30 stainless steel ring was lowered into an alumina crucible containing an agitated Cu-33 w/o Mg-0.7 w/o Pu alloy and a MgCl_2 -30 m/o NaCl-20 m/o KCl salt mixture at 700°C. Samples were taken of the metal and salt phases before the CB-30 was exposed to the melt, during the exposure, and after the ring was removed from the melt. Analyses of plutonium in the samples showed that a loss of about 2% of the plutonium charged occurred during the exposure

of the CB-30 ring to the melt. No further testing of CB-30 as a container material will be undertaken. However, additional experiments will be conducted to establish the stability of plutonium solutions with other candidate container materials.

3. General Chemistry and Chemical Engineering--Research and Development

a. Characterization of Carbon-bearing Species in Sodium

(i) Solubility of NaCN in Liquid Sodium (F. A. Cafasso)

Last Reported: ANL-7382, p. 125 (Sept 1967).

The formation of cyanide* in liquid sodium from carbon- and nitrogen-bearing contaminants has generated interest in NaCN as a potential corrodent and plug-former. However, information needed to evaluate these possibilities is lacking. In fact, the Na-NaCN system itself has received such limited attention in the past that not even solubility data are available. As a first step in the investigation of cyanide behavior, the solubility of NaCN in sodium in the range 400-550°C is being determined.

The requisite data are being obtained by equilibrating a sodium melt with excess NaCN at a preselected temperature, sampling this melt with a tantalum pipette provided with stainless steel filters of 5 μ porosity, and assaying the sample for cyanide. Recent experience has shown that appreciable radial and longitudinal segregation of cyanide occurs upon solidification of sodium in the sample pipette. The bulk of the dissolved cyanide crystallizes near and on the top surface of the filter. Because the segregation of cyanide was not suspected in the preliminary experiments, quantitative assay was not accomplished, and the previously reported solubility (~ 0.4 ppm NaCN at 500°C) was too low. By the use of a technique which insures the assay of all of the cyanide in the filtered samples, a higher solubility, 750 ± 110 ppm NaCN at 550°C, has been measured.

b. Studies of the Colloidal Behavior of Carbon-bearing Species in Sodium (F. A. Cafasso)

Last Reported: ANL-7329, pp. 41-42 (April 1967).

Considerable experimental evidence strongly suggests that carbon-bearing particles, varying in size and nature, are dispersed in

* Hobart, E. W., and Bjork, R. G., Nucl. Appl. 1, 490 (1965); Luner, C., and Anderson, K., Chemical Engineering Division Semiannual Report, July-December 1965, ANL-7125, p. 141 (May 1966).

reactor-grade sodium.* The tendency of sodium to act as a dispersion medium for particulates may, therefore, be as important in carburization phenomena as its tendency to act as a solvent. For a rational study of carburization in sodium systems, stable and well-characterized dispersions of colloidal carbon in sodium will be needed; their preparation is being attempted.

Of the many techniques which have been used to create colloids in liquids, ultrasonic dispersion and the arc-discharge appear to offer the greatest promise for use in this study. Ultrasonic dispersion has been a particularly powerful tool for creating colloids in a wide variety of nonmetallic liquids; arc-discharge** has been especially useful for generating one of the few well-characterized colloids in a metallic liquid, an iron sol in mercury. Accordingly, attempts will be made to disperse chosen particulate solids in NaK eutectic and in Hg at room temperature by the arc-discharge method and by ultrasonic stimulation. Selection of the NaK eutectic and Hg as the dispersing media for initial studies arose from considerations of convenience, similarity with the systems to be studied later, and extent of published information.**† By performing parallel experiments in Hg and NaK, the ability of each liquid to act as a dispersing medium for identical particulate solids (e.g., carbon, Fe, rare-earth hydrides, and Al_2O_3) may be compared. Important clues concerning the parameters which must be controlled in order to form and stabilize dispersions in liquid sodium may be gained from such studies. Appropriate apparatus is being assembled and equipment ordered.

* Luner, C., Cosgarea, A., Jr., and Feder, H. M., "Solubility of Carbon in Sodium," Alkali Metal Coolants, Proc. IAEA Symp., Vienna, Nov. 28-Dec. 2, 1966, Intern. Atomic Energy Agency, Vienna, 1967, pp. 171-179.

** Mayer, A., and Vogt, E., Z. Naturforsch. **7a**, 334 (1952).

† Luborsky, F. E., J. Phys. Chem. **61**, 1336 (1957); Warf, J. M. C., Karol, W. L., and Hardcastle, K. I., Inorg. Chem. **5**, 1726 (1966).

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Multiple Scattering Correction Using a Combination of Monte Carlo and Analytical Methods

S. A. Cox

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Polarization of Neutrons Elastically Scattered from Silicon and Magnesium

S. A. Cox

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Engineering Development of Fluid-bed Fluoride Volatility Processes.

Part 12. Exploratory Tests on Fluorination of Uranium Oxide with BrF_5 in a Bench-scale Fluid-bed Reactor System

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ANL-7362 (Aug 1967)

Reprocessing of Reactor Fuels by Fluoride Volatility and Pyrochemical Techniques

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Physics of Plutonium-Fuelled Thermal Research and Test Reactors

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Thermal Properties of Compositions in the System $\text{UO}_2\text{-ZrO}_2\text{-CaO}$

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Analysis of the Initial Critical Experiments of the EBWR Plutonium Recycle Program

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Lattice Constants and Melting Points of Actinide-Group IVA-VIA Compounds with NaCl -type Structures

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Kinetics of the Titanium-Nitrogen Reaction with Application to the Design of Gettering Systems

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Stress Analysis of a Reactor Core Support Structure Consisting of Two Interconnected Multiregion Plates

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The Thermal-Neutron Capture Cross Section of Cadmium

J. W. Meadows
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Flux Conversion in Resonance Reactions

P. A. Moldauer
Proc. Intern. Nuclear Physics Conf., Gatlinburg, Tennessee, September 12-17, 1966, Ed. R. L. Beeker and A. Zucker. Academic Press, New York, 1967, pp. 815-818

T-Violation Effects in Nuclear Reactions

P. A. Moldauer
Bull. Am. Phys. Soc. 12, 1173 (Dec 1967) Abstract

Intermediate Structure in Fast Neutron Processes: Fact, Fantasy, and Maybe

A. B. Smith
Bull. Am. Phys. Soc. 12, 1168 (Dec 1967) Abstract

Laboratory Investigations in Support of Fluid-bed Fluoride Volatility Processes. Part XV. Estimation of Rates of Thermal Decomposition of Plutonium Hexafluoride in Process Streams

L. E. Trevorrow and M. J. Steindler
ANL-7347 (July 1967)

Electron Microscopic Studies of Structural Changes in High Burnup UC-PuC

Y. K. Yoon, H. Watanabe, and J. H. Kittel
Plutonium as a Research Fuel, Proc. IAEA Symp., Brussels, March 13-17, 1967. Intern. Atomic Energy Agency, Vienna, 1967, pp. 455-467

IV. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor-- Research and Development

1. Instrumentation and Control (J. H. Tessier)

Last Reported: ANL-7399, pp. 149-152 (Nov 1967).

a. Dynamics Analysis. The dynamics analysis includes studies of loss-of-coolant accidents and transient stress analysis.

(i) Loss-of-coolant Accidents. The difficulties in attempting to run the FLASH code for a system representative of the AARR have proven to be due to intersections and crossovers of constant-pressure data for values of liquid enthalpy and density in the low enthalpy and pressure range. This effect, which yields spurious oscillations and errors in calculated system pressure, was confirmed by replacing the physical-property data for water, stored in the code, with a data set designed to obviate the intersections. Subsequent calculations with FLASH yielded results devoid of previous pressure oscillations.

Because the substituted water-property data were chosen arbitrarily without regard for high accuracy, an accurate set of data in regions of interest to the AARR blowdown problems is being generated with a computer program based on methods of water-property formulation given in the 1967 Steam Tables.* These values will be substituted in FLASH and a set of trial calculations performed.

(ii) Transient Stress Analysis. The one-dimensional stress analysis of the involute plate subjected to a uniform transverse pressure (see ANL-7399, pp. 150-152) has been modified to include the effect of variable transverse pressure and temperature gradients. The plate distortions at a uniform temperature are shown in Fig. IV.A.1. The expansion results in a configuration that depends on the imposed boundary conditions: pinned, fixed, or rotated edges.

Fig. IV.A.1
Involute-plate Expansion under
Uniform Temperature

* 1967 Steam Tables, published for the Electrical Research Association by Edward Arnold, Ltd., London (1967).

This one-dimensional involute-plate model can treat a variable pressure along the length, and a temperature change along the length and across the thickness of the plate. Thus, averaged plate variations along the radial coordinate of the core can be analyzed to yield information on the deformation and strength of the fuel plates.

Because the loading of the fuel plate varies significantly in both directions, a two-dimensional stress-analysis model is being developed. The reference* used for the one-dimensional case also provides the base for the two-dimensional analysis. The equations of equilibrium of a cylindrical body, modified for the involute shape, become

$$\left. \begin{aligned} \frac{\partial N_{\theta y}}{\partial \theta} + a\theta \frac{\partial N_y}{\partial y} &= 0, \\ \frac{\partial N_{\theta}}{\partial \theta} + a\theta \frac{\partial N_{y\theta}}{\partial y} - Q_{\theta} &= 0, \\ \frac{\partial Q_{\theta}}{\partial \theta} + a\theta \frac{\partial Q_y}{\partial y} + N_{\theta} + pa\theta &= 0, \\ \frac{\partial M_{\theta}}{\partial \theta} - a\theta \frac{\partial M_{y\theta}}{\partial y} - a\theta Q_{\theta} &= 0, \\ \frac{\partial M_{\theta y}}{\partial \theta} + a\theta \frac{\partial M_y}{\partial y} - a\theta Q_y &= 0, \end{aligned} \right\} \quad (1)$$

where θ is the angular coordinate of the involute generated from a cylinder with radius a , and y is the coordinate along the axis of the cylinder. Normal loads per unit length are designated by N , shear by Q , bending moment by M , and the transverse pressure by p , which is a function of θ and y .

Equation (1) can be reduced to a set of three equations when Q_{θ} and Q_y are solved in terms of other loads and substituted into the remaining expressions:

$$\left. \begin{aligned} \frac{\partial N_{\theta y}}{\partial \theta} + a\theta \frac{\partial N_y}{\partial y} &= 0, \\ \frac{\partial N_{\theta}}{\partial \theta} + a\theta \frac{\partial N_{y\theta}}{\partial y} + \frac{\partial M_{y\theta}}{\partial y} - \frac{1}{a\theta} \frac{\partial M_{\theta}}{\partial \theta} &= 0, \\ N_{\theta} + a\theta \frac{\partial^2 M_y}{\partial y^2} + \frac{\partial^2 M_{\theta y}}{\partial y \partial \theta} - \frac{\partial^2 M_{y\theta}}{\partial y \partial \theta} + \frac{1}{a\theta} \frac{\partial^2 M_{\theta}}{\partial \theta^2} - \frac{1}{a\theta^2} \frac{\partial M_{\theta}}{\partial \theta} + pa\theta &= 0. \end{aligned} \right\} \quad (2)$$

*Timoshenko, S., and Woinowski-Krieger, S., Theory of Plates and Shells, McGraw-Hill Book Co., Inc., Second Edition (1959).

Expressions for loads and moments are also given in terms of stress. When adapted to a cylindrical-involute system they become

$$\left. \begin{aligned} N_{\theta} &= \int_{-h/2}^{h/2} \sigma_{\theta} dz, & M_{\theta} &= \int_{-h/2}^{h/2} \sigma_{\theta} z dz, \\ N_y &= \int_{-h/2}^{h/2} \sigma_y \left(1 - \frac{z}{a\theta}\right) dz, & M_y &= \int_{-h/2}^{h/2} \sigma_y z \left(1 - \frac{z}{a\theta}\right) dz, \\ N_{\theta y} &= \int_{-h/2}^{h/2} \tau_{\theta y} dz, & M_{\theta y} &= \int_{-h/2}^{h/2} \tau_{\theta y} z dz, \\ N_{y\theta} &= \int_{-h/2}^{h/2} \tau_{y\theta} \left(1 - \frac{z}{a\theta}\right) dz, & M_{y\theta} &= \int_{-h/2}^{h/2} \tau_{y\theta} z \left(1 - \frac{z}{a\theta}\right) dz, \end{aligned} \right\} (3)$$

where σ and τ are the normal and shear stresses in the plate, respectively, and z is the coordinate across the plate thickness h . Stresses for thin curved bodies may be expressed in terms of strains (ϵ and γ), changes of curvature (χ), and temperature (T). These relationships are

$$\left. \begin{aligned} \sigma_{\theta} &= \frac{E}{1-\nu^2} \left[\epsilon_{\theta} + \nu \epsilon_y - z(\chi_{\theta} + \nu \chi_y) - (1+\nu) \alpha T \right], \\ \sigma_y &= \frac{E}{1-\nu^2} \left[\epsilon_y + \nu \epsilon_{\theta} - z(\chi_y + \nu \chi_{\theta}) - (1+\nu) \alpha T \right], \\ \tau_{\theta y} = \tau_{y\theta} &= \frac{E}{2(1+\nu)} \left[\gamma_{y\theta} - 2z\chi_{y\theta} \right], \end{aligned} \right\} (4)$$

where the temperature may be a function of θ , y , and z . The strain and changes in curvature can in turn be related to displacements u , v , and w in the coordinate directions of θ , y , and z , respectively. These are given* as

$$\left. \begin{aligned} \epsilon_{\theta} &= \frac{1}{a\theta} \left[\frac{\partial u}{\partial \theta} - w \right], & \epsilon_y &= \frac{\partial v}{\partial y}, & \gamma_{y\theta} &= \frac{\partial u}{\partial y} + \frac{1}{a\theta} \frac{\partial v}{\partial \theta}, \\ \chi_{\theta} &= \frac{1}{a^2 \theta^2} \left[\frac{\partial v}{\partial \theta} + \frac{\partial^2 w}{\partial \theta^2} \right], & \chi_y &= \frac{\partial^2 w}{\partial y^2}, & \chi_{y\theta} &= \frac{1}{a\theta} \left[\frac{\partial u}{\partial y} + \frac{\partial^2 w}{\partial y \partial \theta} \right]. \end{aligned} \right\} (5)$$

* Timoshenko, S., and Woinowski-Krieger, S., loc cit.

Integration of Eqs. (3) can be performed when Eqs. (4) are substituted therein. This yields expressions for the internal loads and moments in terms of strain, change in curvature, and temperature. However, strain and the change in curvature are given in terms of displacements by Eqs. (5). Thus, the internal loads can be related to the displacement components by the introduction of Eqs. (5). Finally, substitution can be made into Eqs. (2), yielding three partial differential equations with three unknowns u , v , and w .

When the foregoing equations have been checked, an attempt will be made to solve them subject to given boundary conditions; apparently, practical solutions depend on numerical methods.

2. Core Research and Development

a. Heat Transfer Analysis (R. R. Rohde)

Last Reported: ANL-7399, pp. 143-146 (Nov 1967).

(i) Cooling of Neutron Windows. The neutron windows, large sections of aluminum positioned between the reflector shroud and pedestal and the pressure-vessel wall, are provided to obtain the neutron intensities needed for the neutron-detection instruments outside the vessel. Coolant flows through holes in the windows to keep the aluminum and coolant temperatures within reasonable ranges. The main design criteria are (1) to minimize perturbations in the core neutron flux near the windows, (2) to prevent temperatures from exceeding allowable aluminum-scale limits, and (3) to provide a capability for natural-convection cooling during emergencies.

A preliminary heat-transfer analysis was made to establish an initial coolant-hole pattern in the window and to investigate whether natural-convection cooling without a phase change is permissible for full-power operating conditions.

Figure IV.A.2 shows typical dimensions of two neutron windows being considered. Calculated heating rates at full-power (100 MW) operation were used in the analysis, and an overpower factor of 1.15 was applied. The spacing of the holes was based on an analytical model consisting of an externally insulated hollow cylinder with uniform volumetric heat generation, cooled on the inside surface. The heat flux at the horizontal centerline was limited to $\leq 50,000$ Btu/hr-ft² at 115 MW.

At the interface of the shroud and neutron window, with a coolant gap of 0.0625 in., natural-convection cooling with laminar flow yields a heat-transfer coefficient of approximately 200 Btu/hr-ft²-°F. The 0.250-in. gap between the beryllium and shroud would have a coefficient of approximately 50 Btu/hr-ft²-°F. Under these conditions the shroud would reach an acceptable maximum temperature of ~260°F.

Fig. IV.A.2. Plan (top) and Elevation (bottom) Section Views of AARR Neutron Windows. Table IV.A.1 summarizes locations of coolant holes.

Nonboiling natural-convection cooling of the window would yield film coefficients of 200-600 Btu/hr-ft²-°F, depending on whether laminar- or turbulent-flow conditions exist in the coolant channel. For the channel dimensions assumed, the natural-convection flows would be in the laminar-turbulent transition range. The surface temperature of the neutron window would be above 500°F with the lower film coefficient and about

360°F with the higher one. Flow oscillations that would perturb neutron-flux measurements can occur because the flow is in the transition range, so the growth of aluminum oxide scale would be a problem at either temperature. Therefore, natural-convection cooling is not recommended for normal reactor operation.

To avoid these difficulties, forced flow is recommended to cool the neutron window. Cylindrical coolant holes should have a minimum diameter of 3/16 in., and the inlet and outlet should be chamfered in the same proportion as the reflector coolant holes. Table IV.A.1 summarizes the recommended dimensions and locations for the coolant passages.

TABLE IV.A.1. Locations of Coolant Holes in Neutron Windows

Radial Distance ^a (in.)	Maximum Distance between Holes at Radial Position (in.)	Minimum Number of Holes ^b at Radial Position	
		For 16-in.-wide Window	For 24-in.-wide Window
0.5	1.0	16	24
1.5	1.0	16	24
2.57	1.14	14	21
3.79	1.30	13	19
5.22	1.56	11	16
6.97	1.94	9	13
9.22	2.56	7	10
12.35	3.7	5	7
18.00	7.6	3	4

^aFrom inside edge of neutron window.

^b3/16-in.-dia holes in triangular array.

A velocity of 11 ft/sec would provide a film coefficient of ~3000 Btu/hr-ft²-°F and require a pressure differential of ~3.5 psi; this would keep the neutron window below 150°F and require ~30 gpm of coolant. Aluminum-scale growth under these conditions would be negligible.

Each neutron window can be made in one large piece or in a series of vertical slabs in which the radial parting line is selected at the radial distance listed in Table IV.A.1. The coolant inlet and outlet regions should provide a number of passages to prevent gross blockage of coolant flow.

b. Core Structure Development (W. J. Kann)

Last Reported: ANL-7399, p. 142 (Nov 1967).

(i) Permanent Reflector. The initial rough machining of the beam-tube holes to $5\frac{1}{4}$ -in. diameter is proceeding. The beryllium block was cored, and tensile properties were measured for samples from both top and bottom sections of the core:

Mechanical Property	Top of Block		Bottom of Block	
	Longitudinal	Transverse	Longitudinal	Transverse
Ultimate strength, psi	44,600	54,300	44,600	56,400
Yield strength, psi	35,900	35,200	34,100	34,600
% elongation	1.1	2.7	1.1	4.0

These values exceed considerably the specified minimums.

After the block was cored, a second ultrasonic test was performed, this time in the radial direction. No defects were discovered, confirming the soundness of the block.

A check analysis of the material after pressing and sintering showed the content of magnesium to be within specified limits.

Several changes are being made in the permanent reflector because of the increased coolant-hole size, including coolant passages in the vertical experimental-facility liners rather than slots in the holes.

(ii) Stress Analysis. Work has begun on Phase I of Task I under the stress-analysis contract with Franklin Institute. A check of the beam-tube-adapter wall for compliance with ASME Code Section III indicated that it was thicker than the code allows.

The mathematical models for pressure- and thermal-stress analysis of the blind and through beam tubes were completed except for some minor dimensional details in the shield plug and collimator flange area.

Work was also initiated on Phase I of Task II.

3. Component Development

a. Sample Irradiation Devices (R. W. Seidensticker)

Last Reported: ANL-7399, pp. 146-148 (Nov 1967).

Static leakage tests are underway on the type of seal that will connect the internal-thermal-column (ITC) rabbit tubes to the ITC valve-head assembly. The vertical irradiation-facilities mockup has been revised, and the static irradiation-facilities handling tools will be retested.

The preliminary design of the instrument thimbles is complete and is being reviewed.

Preoperational work on the gas-operated test loop is about 50% complete. The flow system and operating procedures were modified, and parts were ordered to implement modifications.

b. Beam Tubes Development (A. R. Jamrog)

Last Reported: ANL-7399, p. 148 (Nov 1967).

(i) Beam-tube Test Facility. Assembly of the test facility is about 75% complete. The prototype beam-tube adapters at both ends of the test facility have been installed, including the pool seal bellows assembly. The adapters were aligned and connected to the 10-in. nozzles with the Grayloc clamps, which were torqued to ~640 ft-lb. The pool-seal bellows with flanges and appropriate gaskets was bolted to the shield housings to complete the simulated vessel nozzle-to-biological-shield sleeve transition.

The pool-tank cover was gasketed and bolted to the top flange of the tank, which was then pressurized with helium to 5 psig and leaktested. No leakage was detected, so the tank was depressurized, filled with water, pressurized to 15 psig (equivalent to ~30 ft of reactor-pool water), and isolated. After 72 hr, all critical seal regions of the nozzle-to-pool-seal-bellows assembly transition were inspected. No leakage was observed.

The work involved in fitting, welding, and leaktesting the piping and valves of the test facility to the general-purpose hydraulic loop is about 85% complete.

c. Internal Thermal Column Development (R. W. Seidensticker)

Last Reported: ANL-7399, p. 149 (Nov 1967).

Final design calculations and material specifications are being prepared for the preconstruction safety review of the ITC valve-head assembly. Detailed construction drawings for the ITC core assembly have been started.

4. Physics Experiments and Analysis (D. H. Shaftman)

Last Reported: ANL-7391, p. 166 (Oct 1967).

a. Neutron Inventories. Analyses of neutron inventories have been made to investigate further certain reactor-physics characteristics of the AARR. The results summarized in the following paragraphs cannot be correlated simply with reactivity losses and gains, since average "importance" weightings for fission events, net neutron leakage, and neutron capture must be applied on an energy-group-by-energy-group basis. Nevertheless, the energy-integrated summaries are aids to a deeper insight into the physics behavior of the system.

The analyses constitute interpretations of the printout of a one-dimensional (cylinder) 16-group k_{eff} calculation with the diffusion-theory program MACH-1. The reactor state is an idealized system: the (super-critical) fresh hot reactor, without fission products and with control rods fully withdrawn. The Internal Thermal Column (ITC) is assumed to be fully loaded with aluminum capsules, with a total additional neutron poisoning of the ITC equivalent to 0.5 g natural boron, representing the total neutron capture cross section of the actual target materials. With a normal HFIR loading of boron as a burnable neutron poison for this system, the calculated k_{eff} is 1.076.

In the following summaries, the fuel zone is defined as the volume made up of a 20-in. active core height and radial dimensions from the inner surfaces to the outer surfaces of the active fuel plates in the inner and outer fuel elements. Table IV.A.2 displays the input group-averaged η 's in the fuel, and the inferred fractional distributions of fission events in various radial locations and the average over the entire fuel zone.

Table IV.A.3 summarizes the fractional distribution of neutron losses due to absorption and leakage. The losses are sums over all 16 groups for each of the two fuel elements. Other small neutron losses due to capture by other materials, including impurities in the aluminum and oxygen, have been ignored in compiling Table IV.A.3.

TABLE IV.A.2. Group-averaged η 's and Energy Distributions of Fission Events in the AARR Fuel Zone.

Group Number	Energy Range	Group-averaged η^a	Local Fractional Distributions of Fission Events ^b				
			Inner Fuel Element		Outer Fuel Element		Average over Fuel Zone ^c
			Near Inner Boundary	Central Region	Central Region	Near Outer Boundary	
1	3- ∞ MeV	2.819	0.0007	0.0017	0.0018	0.0006	0.0016
2	1.4-3 MeV	2.453	0.0015	0.0036	0.0042	0.0015	0.0036
3	0.9-1.4 MeV	2.308	0.0007	0.0018	0.0022	0.0008	0.0018
4	0.4-0.9 MeV	2.196	0.0010	0.0027	0.0032	0.0012	0.0027
5	0.1-0.4 MeV	2.104	0.0013	0.0033	0.0039	0.0015	0.0033
6	17-100 keV	1.924	0.0017	0.0041	0.0048	0.0019	0.0041
7	3-17 keV	1.853	0.0022	0.0053	0.0063	0.0025	0.0053
8	0.55-3 keV	1.722	0.0038	0.0089	0.0106	0.0043	0.0090
9	100-550 eV	1.637	0.0087	0.0203	0.0238	0.0098	0.0203
10	30-100 eV	1.517	0.0119	0.0271	0.0313	0.0132	0.0268
11	10-30 eV	1.452	0.0109	0.0244	0.0279	0.0120	0.0241
12	3-10 eV	1.167	0.0107	0.0236	0.0267	0.0117	0.0231
13	1-3 eV	1.867	0.0078	0.0172	0.0196	0.0085	0.0168
14	0.4-1 eV	2.143	0.0147	0.0317	0.0359	0.0159	0.0311
15	0.1-0.4 eV	2.007	0.1440	0.1897	0.1983	0.1351	0.1894
16	0-0.1 eV	2.075	0.7784	0.6346	0.5993	0.7794	0.6370

^aThe group-averaged parameter η_j is the assigned average number of fission neutrons produced per neutron absorbed in the uranium fuel.

^bThe local fractional distribution is normalized to a total fission rate of unity locally. Four decimal places are included to avoid too much roundoff at high energies; these are the calculated values, and they bear no stamp of certitude to four decimals.

^cAt the inner and outer boundaries of the fuel zone, more than 90% of the fissions are caused by neutrons of "thermal" energies, say groups 15 and 16 (0-0.4 eV). Averaging over the fresh-fuel zone, more than 80% of the fissions occur in this energy range.

TABLE IV.A.3. Fractional Distributions of Neutron Losses in the AARR Fuel Zone (Summed over Neutron Energy)

Type of Loss	Inner Fuel Element	Outer Fuel Element	Fuel Zone
Net axial leakage	0.024	0.042	0.066
Net radial leakage out inner boundary	0.047	-0.010	0.037
Net radial leakage out outer boundary	0.036	0.253	0.289
Captures by aluminum (in fuel zone)	0.0055	0.0082	0.014
Captures by hydrogen	0.0075	0.0121	0.020
Captures by boron	0.027	-	0.027
Absorptions by ²³⁸ U	0.0010	0.0026	0.004
Captures by ²³⁵ U	0.033	0.070	0.103
Fissions of ²³⁵ U	0.142	0.297	0.439
Total fractional loss	0.32	0.68	1.00

Notes: (1) There are slight inconsistencies between individual and total losses; they are caused by roundoff.

(2) A net negative leakage out a boundary is equivalent to a net source of neutrons into the zone at that boundary, summed over all neutron energies.

In the idealized case represented in Table IV.A.3 the control rods are withdrawn fully, leaving the aluminum control-rod followers in place at the outer radial boundary of the outer fuel element. More realistically, control rods are partially inserted, and there is a radial shift in power toward the ITC, and hence a greater radial leakage of high-energy neutrons into the ITC. There would be also a larger net "leakage"

out of the outer fuel element, for the control rods would absorb some neutrons that otherwise would enter the fuel zone from last collisions in the outer radial reflector zones. Moreover, xenon builds up early in the reactor operation and compensates for a large fraction of the initial available excess reactivity. Consequently, the overall effects of the control rods on the inventory of neutron losses in the fuel zone should be small. Of course, since the values in Table IV.A.3 are normalized to unit total loss in the fuel zone, all values must be adjusted to account for the reactivity losses to xenon and to control-rod absorptions.

b. Gamma-ray Heating of the Reactor Pool due to Radiations from the Operating Core. The measured heating rate in the HFIR reactor pool has been reported to be higher than the heating rates calculated as direct heat losses from the primary-coolant system, including pipe runs and the reactor vessel. Scientists of Oak Ridge National Laboratory have calculated a total heating rate of ~4 kW from gamma-ray heating of the pool, which is very much smaller than the above-mentioned discrepancy (~150 kW). Nevertheless, an independent estimation of this heating rate has been made at Argonne National Laboratory. Conservatively, the total gamma-ray heating rate of the HFIR reactor pool is 20 kW during normal reactor operation at 100 MW. For this estimate, the computer program MAC was used; the MAC limitation to infinite-slab media for the gamma-ray calculation was compensated by a geometry correction for the gamma rays leaking from the sphere-like reactor core. The contribution to the pool heating from radiations originating in reflector regions closer to the pool was determined and corrected appropriately for geometry. Actually, the intensities of the radiation sources in these noncore regions were calculated in spherical geometry in the Argonne version of the MAC program, so the final geometry corrections were relatively small.

The estimated heating rate of 20 kW is considered to be somewhat conservative; it is too low to account for the indicated discrepancy between the measured and calculated heating rates, in agreement with the conclusion of the ORNL scientists.

A $1\frac{1}{2}$ -in.-thick steel ring inside the AARR reactor vessel, close to the vessel wall, supports various structures and also provides thermal shielding against radiations arising from the operating reactor core and beryllium reflector. Further, the AARR reactor vessel is approximately 1.8 in. thicker than the HFIR vessel. This additional radiation shielding reduces the gamma-ray heating rate in the AARR Reactor Pool to a calculated value of ~3 kW.

PUBLICATION

ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

Analog Analysis of Flow Reversal and Water-hammer Pressure in a Typical
Closed AARR Hydraulic Rabbit-tube Facility

Raymond R. Rohde and Alan L. Winiecki

ANL-7336 (Aug 1967)

V. NUCLEAR SAFETY

A. Other Reactor Kinetics--Research and Development

1. Fuel Meltdown Studies with TREAT (C. E. Dickerman)

a. Meltdown Studies

(i) Transient Coolant Behavior

(a) Studies of Transient Sodium Behavior. A knowledge of the transient behavior of sodium under accident conditions is necessary for analysis of the course of core response of a fast reactor during an accident. Coolant vaporization and expulsion of liquid can influence the reactor power level both directly through the void coefficient of reactivity and indirectly through their effects on fuel heating and movement. This task has been established with the two specific goals of (1) investigating application of an established analytical model for analyzing transient coolant behavior during TREAT loop experiments with test-section geometry simulating core conditions much more closely than in the usual "basic" laboratory heat transfer experiments and (2) performing TREAT loop experiments to test the theoretical predictions. These loop runs are to be made with the goal of checking predictions about coolant behavior made with a model assuming an intact coolant-pin cluster geometry rather than fuel-meltdown behavior. Thus, in general, experiments will be terminated short of gross pin damage.

Four Mark-II TREAT integral sodium loops* are under construction for the transient coolant behavior or expulsion experiments. Each loop is to be outfitted with pressure transducers and electromagnetic flowmeters at both inlet and outlet to the test section. Coolant thermocouples will be placed at inlet and outlet; at present, the thermocouples with reliable performance have a relatively slow response, ~25 msec, and will be used primarily to check calculation results for relatively mild and slow temperature excursions. In principle, channel-void detection is possible using the fast-neutron hodoscope at TREAT, since a sodium void decreases the background counting rate of the hodoscope, by decreasing the number of fast neutrons scattered into the hodoscope system. Such decreases in the signals from hodoscope channels lined upon sodium channels have been detected in TREAT experiments using the thinner-walled Mark-I integral sodium loop. These decreases have correlated well with other instrument data and calculations. Measurements of background and attenuation from steel indicate that this method of detecting voiding is marginal for the thick-walled Mark-II loop and should be checked directly before it is considered to be feasible for the coolant expulsion experiments.

*Reactor Development Program Progress Report for March 1967, ANL-7317, p. 106.

The purpose of this task is to investigate analytically certain aspects of the model developed by MacFarlane.* The intention is to update, where possible, details such as thermophysical property correlations and to determine if any new methods of analysis are available that could be easily incorporated into the code to provide better descriptions of the physical phenomena. The scope of the project is deliberately limited in that no extensive modifications of MacFarlane's code are to be made; the goals are to check MacFarlane's model and (based on calculations closely linked to experimental conditions) to specify a first set of TREAT sodium loop experiments designed to check applicability of the model.

Thus far, the main effort has been a detailed study of the existing code to determine how it functions and to what degree certain input variables may be changed. Studies of the model used have been made to investigate the assumptions and approximations necessary to utilize the model. Also, some study has been devoted to the hydrodynamic and heat-transfer phenomena in liquid metals to survey the most recent work and to determine how well the existing model may or may not describe the physical situation on the basis of the limited data available.

Some initial calculations concerning certain aspects of the existing code were made to determine if some improvements might easily be made. A survey of the most recently published correlations for liquid sodium was made to determine if any significant differences existed between those correlations and the ones originally available in the code. The recent review work of Golden and Tokar** proved to be a very good source for most of the correlations. Calculations comparing the best correlations available with those used by MacFarlane show very little difference. The newer correlations are retained in the code since they are thought to be the best available and in some cases cover a wider temperature range.

Some studies of the Martinelli flow parameter were made to determine if an improved correlation for liquid metals was available or if the correlation could be modified in any way that would yield more realistic results. Very recent experimental studies by the French presented at the International Conference on the Safety of Fast Reactors at Aix-en-Provence (France), September 19-22, 1967 (Proceedings in press) were compared with results predicted by the methods now used in the code. For the range of the Martinelli parameter X_{tt} from 6.0 to 150, the calculated results were somewhat above the experimental results, but were not outside the uncertainties common to such experiments, especially at high liquid fractions. The French paper did not give all the necessary parameters that were used in obtaining the experimental results, and some values

*MacFarlane, D. R., An Analytic Study of the Transient Boiling of Sodium in Reactor Coolant Channels, ANL-7222 (June 1966).

**Golden, G. H., and Tokar, J. V., Thermophysical Properties of Sodium, ANL-7323 (Aug 1967).

were estimated for purposes of the calculation. Also, at the same conference a group from Atomics International presented an alternate correlation for "maximum" liquid fraction as a function of the Martinelli parameter. Upon calculation it was observed that their correlation gave results significantly different from the French data (or the correlation in the code). Further investigation beyond the information available in the paper determined that the correlation is applicable only at the critical point. As a result of this review of recent work, the present methods will be retained.

Some study has been made concerning heat-transfer correlations which may be improvements upon present methods. At present the same correlation is used in the single- and two-phase region, and local boiling is not treated (this assumption is consistent with the flow model). On the basis of the available data it appears that no change in the code is justified now.

Further work on this project will consist of outlining and conducting a number of computer case studies that will serve as a basis for specifying TREAT experiments to check the analytical model.

(ii) Development of Analytical Methods

(a) Development of Advanced Transient Heat-transfer Calculations. The ARGUS transient heat-transfer code is a flexible multi-region heat-transfer code written for cylindrical geometry. It was developed especially for, and has been used extensively in, analysis of fast reactor safety experiments conducted in TREAT. Experience gained in using the code and new experimental requirements have resulted in new specifications requiring nearly complete revision of the physics content of the code.

To aid in the evaluation of code output, a flexible graphing package has been added. This package permits input specification of graphs to be made of output as a function of time, axial or radial position, with spatial positions or time as parameters, as appropriate.

No changes are being made in the capability for more than one coolant channel and both upflow and downflow in a single problem.

The heat transfer equations for solid regions have been reformulated based on those used as an "experimental" technique in a hybrid program.* The new model gives much more accurate results than the original ARGUS equations for the case of materials with variable conductivity, like UO_2 . In addition, the new formulation rigorously conserves energy, whereas the original model did not under certain conditions. The original provision for special "thin region" equations to describe heat transfer in thin regions like fuel-element cladding without excessive running

*Bryant, L., Amiot, L., Dickerman, C., and Stephany, W., Transient Temperature Calculations Using a Hybrid Computer, Argonne National Laboratory report, to be published.

time is to be replaced by a modification of the ARGUS boundary-node calculation method coupled with a change in the node-point geometrical model. Although this will improve running time generally, the main reason for the change is to make possible certain calculations involving very thin regions that could not be done previously because of the excessively small time steps required for mathematical convergence of the forward differencing scheme of the "thin region" equations.

Under special conditions, mathematical instabilities have been encountered in coolant temperatures calculated by the code; methods of alleviating these are under study.

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

a. Equation of State (C. E. Dickerman)

Last Reported: ANL-7391, pp. 175-176 (Oct 1967).

(i) Surface Tension and Critical Points of the Alkali Metals

The surface tensions of the alkali metals are of interest for theoretical and practical applications. Their theoretical importance is due to the relative simplicity of the electronic structure of the alkali metals and their plasma nature. Surface tension also gives a measure of the rate of decrease of interatomic forces in the vicinity of an interface. Practical application includes this quantity as a parameter in relations involving bubble growth and collapse rates, as in boiling heat transfer or cavitation, the size of drops and their adhesion to surfaces of solids, and as a principal determinant in the performance of heat pipes under agravitic conditions.

At very high temperatures, approaching the thermodynamic critical point, the surface tension decreases rapidly with temperature and can be used to estimate the critical point, thus serving as a test of an extended principle of the law of corresponding states. The surface tension vanishes at the critical temperature T_c .

Recent experiments of Bohdanský and Schins* extend the data to higher temperatures than previously available and are in good agreement with several other investigators at lower temperatures. Curve fitting of these data was attempted in two forms. The first is simple linear extrapolation

$$\sigma = \sigma_0 - bT, \quad (1)$$

*Bohdanský, J., and Schins, H. E. J., The Surface Tension of the Alkali Metals, J. Inorg. and Nucl. Chem. 29, 2173 (1967).

where

σ is the surface tension in dynes/cm;

T is the absolute temperature in °K;

σ_0 is the surface tension if the liquid metal existed at absolute zero;

b is the temperature coefficient $d\sigma/dT$.

The second form is one given by Guggenheim,* used by others, derivable from empirical relations between physical properties:**

$$\sigma = \sigma_0[1 - (T/T_c)]^p. \quad (2)$$

Here the symbols have their previous meanings; supposedly p is a universal constant which is equal to $11/9$ for normal liquids.*,** Others have found p to be slightly greater than this value,^{†,††} whereas theory predicts $1.20 < p < 1.29$.

Grosse[†] examined the relative merits of these two equations and showed that linear extrapolation gave a slightly higher value of σ_0 for the liquid metals with the exception of sodium.

The experimental data of Bohdanský and Schins may be slightly in error due to the use of density equations in the reduction of their data, which are somewhat different from those recommended after a study of recent experimental data.[§] However, these data were used to determine the constants in Eqs. (1) and (2) by a least-squares technique. In applying Eq. (2), two methods were used:

(a) The critical temperatures of Miller, Cohen, and Dickerman[§] were assumed, and σ_0 and p determined; this is termed σ_0^a .

(b) The exponent $11/9$ was assumed, and σ_0 and T_c determined; here the value is termed σ_0^b .

The results are given in Table V.A.1 with the values for xenon given for comparison.

*Guggenheim, E. A., The Principle of Corresponding States, J. Chem. Phys. **13**, 253 (1945); Proc. Phys. Soc. (London) **85**, 811 (1965).

Rykov, V. I., Scheinfel'd, V. L., Application of the Frenkel-Gubanov Formula to Normal Molecular Liquids, Izvestia VUZ. Fizika **4, 108 (1965).

[†]Grosse, A. V., The Relationship between the Surface Tensions and Energies of Liquid Metals and Their Critical Temperatures, J. Inorg. and Nucl. Chem. **24**, 147 (1962).

^{††}Smith, B. L., Gardner, P. R., and Parker, E. H. C., Surface Tension and Energy of Liquid Xenon, J. Chem. Phys. **47**, 1148 (1967).

[§]Miller, D., Cohen, A. B., and Dickerman, C. E., Estimation of Vapor and Liquid Density and Heat of Vaporization of the Alkali Metals to the Critical Point, Proc. International Conf. on the Safety of Fast Breeder Reactors, Aix-en-Provence (France), September 19-22, 1967.

TABLE V.A.1. Constants Fitted to Experimental Surface Tension Data

Material	σ_0 (Eq. 1) (dynes/cm)	b (dynes/cm-°K)	T_c (Eq. 1) (°K)	σ_0^a (Eq. 2) (dynes/cm)	p	$T_c^{(a)}$ (°K)	σ_0^b (Eq. 2) (dynes/cm)	T_c (°K)	$\frac{\sigma_0^a V_c^{2/3}}{T_c}$
Lithium	473.8 ± 2.1	$0.163 \pm 0.5 \times 10^{-5}$	2913	494 ± 1.4	1.770	4310 ± 100	482.2	3306	2.62
Sodium	227.5 ± 0.97	$0.0951 \pm 0.18 \times 10^{-4}$	2392	234 ± 0.97	1.248	2733 ± 40	233.1	2697	2.27
Potassium	138.8 ± 1.3	$0.0787 \pm 0.18 \times 10^{-5}$	1764	142.8 ± 1.1	1.479	2287 ± 40	141.2 ± 1.19	2005	2.24
Rubidium	105.0 ± 1.1	$0.0657 \pm 0.17 \times 10^{-5}$	1598	108.7 ± 1.0	1.582	2164 ± 40	106.7 ± 1.0	1800	2.02
Cesium	86.80 ± 0.82	$0.0569 \pm 0.87 \times 10^{-6}$	1524	90.24 ± 0.73	1.586	2040 ± 40	88.4 ± 0.82	1720	2.10
Xenon	43.06 ± 0.51	$0.1537 \pm 0.15 \times 10^{-4}$	286.5	54.59 ± 0.32	1.287	289.74	54.00	283.60	4.56

(a) Miller, D., Cohen, A. B., and Dickerman, C. E., Estimation of Vapor and Liquid Density and Heat of Vaporization of the Alkali Metals to the Critical Point, Proc. International Conf. on the Safety of Fast Breeder Reactors, Aix-en-Provence (France), September 19-22, 1967.

Values of σ_0 are not strongly dependent on the method of fitting the data, but the critical temperatures, although lower than those estimated from density and heat of vaporization,* agree rather well with those calculated from Eq. (2) by assuming $p = 11/9$. Slight increases in p to the 1.29 predicted by theory bring the estimated critical temperatures into better agreement, although there are indications that the surface tension data for lithium and cesium are both on the low side.

The corresponding-states parameter,**† $\sigma_0 V_c^{2/3}/T_c$, where V_c is the critical volume in cc/g-mole, is roughly constant as expected, but about half the value found by Guggenheim and others for normal fluids such as organics and rare gases (cf. the value for xenon). A quantum-mechanical effect† should cause the value for lithium to be lower than the other alkali metals. The constants given in Table V.A.1 can be used in Eq. (2) to calculate the surface energy E from the relation

$$E = \sigma_0 - T \frac{d\sigma}{dT} \quad (3)$$

or to extrapolate to higher temperatures where data are unavailable.

Equation (2) can be substituted in Eq. (3) with the result

$$E = \sigma_0 \left(1 - \frac{T}{T_c}\right)^{p-1} \left\{1 + \frac{T}{T_c} (p\sigma_0 - 1)\right\}. \quad (4)$$

If p were a universal constant, E would be a function only of σ_0 and the reduced temperature T/T_c . This relation also predicts zero surface energy at the critical point.

*Miller, D., Cohen, A. B., and Dickerman, C. E., *ibid.*, p. 144.

**Guggenheim, E. A., *ibid.*, p. 144.

†Watkinson, A. P., and Lielmezs, J., Corresponding-states Principle and Surface Tension, *J. Chem. Phys.* **47**, 1558 (1967).

b. Material Behavior and Energy Transfer

- (i) Studies of High Temperature Properties of Fast Reactor Materials: Heat Capacity of Liquid UO_2 (L. Leibowitz and M. G. Chasanov)

Last Reported: ANL-7382, pp. 144-146 (Sept 1967).

Work is continuing on the program to determine the heat capacity of liquid uranium dioxide, an important quantity in fast reactor safety calculations. Accurate values of the heat capacity and heat of fusion of molten UO_2 will facilitate estimation of the total energy released and the temperatures attained in the core during destructive nuclear excursions. The techniques employed (induction heating in conjunction with a drop calorimeter) in the program are conventional, although they are being extended to temperatures which have not previously been explored.

The chief effort during the current reporting period has been devoted to testing the operation of the equipment by means of repeated electrical calibrations and by preliminary experimental drops with solid tungsten test bodies.

The initial electrical calibration revealed some difficulties which were corrected. Electrical calibrations of the modified Parr adiabatic calorimeter were then carried out over a sixfold range of heating rates for roughly constant total energy inputs. Of six acceptable runs, a standard deviation of 0.1% was found in the energy equivalent of the calorimeter; this is considered to be satisfactory precision.

Work was then begun on the next phase of equipment testing, namely, measurements of the enthalpy of solid tungsten test bodies. These test bodies were prepared from tungsten rod to be identical in size with the capsules that will be used to measure the enthalpy of tungsten powder and liquid uranium dioxide. Tests have been carried out with the tungsten test bodies at 2075, 2275, and 3225°K. Two duplicate runs at 2275°K agreed within 0.36%, which is considered to be satisfactory precision. Calculated enthalpy values were within a few percent of published data.*

Measurements of the enthalpy of tungsten powder are now being made. The conventional calorimetric procedure of taking the difference in enthalpy between filled and empty capsules will be followed. Data will be reported when the measurements are completed.

*Kirillin, V. A., Sheindlin, A. E., Chekhovskoi, V. Ya., and Petrov, V. A., Thermodynamic Properties of Tungsten in the Range of 0 to 3500°K, Russian Journal of Physical Chemistry 37(17), 1212 (Oct 1963).

Measurements with UO_2 will begin when the tungsten powder work is completed. Fabrication of tungsten capsules containing UO_2 is now being carried out.

3. Reactor Control and Stability

a. Reactor Stability Design Criteria (C. Hsu)

Last Reported: ANL-7371, pp. 114-115 (Aug 1967).

A stability criterion has been derived for a class of nonlinear, spatially dependent reactor systems, by use of the Bihari's inequality. The criterion obtained from this derivation relates the allowable region of the system nonlinearity to the region of asymptotic stability of the system states; it can be used easily for practical problems. Since global asymptotic stability of a power reactor system is generally not required, this development can be applied to such systems. A topical report is being written.

4. Coolant Dynamics

a. Critical Flow (H. K. Fauske)

Last Reported: ANL-7399, p. 154 (Nov 1967).

(i) Sodium Tests. The second phase of the sodium study, involving detailed measurements and analyses of nonequilibrium effects in flashing liquid metals, has begun. The new test section (of 1/8-in. ID and with $L/D = 50$), instrumented with eight pressure transducers and 17 thermocouples, has been installed in the sodium-loop facility. The loop is ready for preliminary shakedown tests and calibration.

Detailed measurements will be made of the superheat required to initiate flashing, as well as two-phase nonequilibrium properties. Parameters to be studied include: pressure, temperature, and velocity under both steady-state and transient conditions.

Preliminary design studies are underway for measuring the rate of flashing (vapor growth) for nonmetallic as well as metallic fluids using fast photography with an X-ray apparatus. Bubble growth rates during stagnant as well as flowing conditions (varying external-pressure field) for high Jacob numbers (high superheat and low pressure) are of primary interest. These data are particularly needed to determine whether the vapor growth is controlled by heat conduction, evaporation, or mass inertia. The information should lead to a better evaluation of two-phase expulsion, critical flow, recondensation, and shock phenomena.

(ii) Sonic Velocity. A new air-water mixer being installed in the velocity-of-sound loop should give higher void fractions than the previous mixer. The effect of flow regime on the compressibility of a two-phase mixture will be studied.

b. Coolant Dynamics (R. M. Singer)

Last Reported: ANL-7399, p. 154 (Nov 1967).

(i) Superheat. A series of tests with a maximum previous pressure of 41.2 psia has been completed; the results have been compared with those from tests with a maximum previous pressure of 15.9 psia. The pressure history markedly influences the incipient superheats; the measured results are now being compared to a theoretical model.

The system has been pressurized to 64.7 psia, and the tests will be repeated to investigate more fully the pressure-history effect. After these tests, a new test vessel having an artificially roughened surface will be installed.

(ii) Expulsion. Correction of minor difficulties with the alignment of the electron-beam gun continues. The temperature-overload safety circuit has been completed and installed. Final assembly of the heater system will be followed by installation of the sodium system.

c. Convective Instability (R. P. Heinisch)

Last Reported: ANL-7399, p. 155 (Nov 1967).

Runs using the vertical rectangular test section with each face heated are complete. The data have been reduced, and are being compared with existing data and theory. A model of the test section is being built to obtain information on the mechanisms involved in the energy-transport process.

d. Core Component Dynamics (M. W. Wambsganss)

Last Reported: ANL-7399, pp. 155-156 (Nov 1967).

In the investigation of the plastic deformation of cylindrical shells under impulsive loads, the particular case of the response to a dynamically applied ring load has been studied in some detail before the study of more general loadings.

Consider a circular cylindrical shell made of a rigid, perfectly plastic material, and let ρ , θ , and ξ be dimensionless cylindrical coordinates, with $\xi = 0$, giving the location of the ring load $\psi(\tau)$ where τ is

dimensionless time. As discussed in ANL-7399, two phases may occur in the deformation history of ring loads. In Phase 1, hinge circles form at $\xi = 0$ and $|\xi| = \xi_1(\tau)$; during Phase 2, there is a hinge circle at $\xi = 0$ and hinge bands in the regions

$$\xi_1(\tau) \leq |\xi| \leq \xi_2(\tau).$$

Partial integration of the equations of motion of the shell and employment of the boundary conditions result in a pair of coupled nonlinear differential equations:

$$\begin{aligned} \xi_1^2 \dot{\eta}_0 &= 4\psi \xi_1 - 3 - \xi_1^2; \\ \xi_1^2 (\dot{\eta}_0 - \dot{\eta}_1) + \xi_1 (\eta_1 - \eta_0) \dot{\xi}_1 &= 6(\psi \xi_1 - 1), \end{aligned} \quad (1)$$

where η_0 and η_1 are the dimensionless radial velocities at the cross sections $\xi = 0$ and $|\xi| = \xi_1(\tau)$, respectively. Given a pulse shape $\psi(\tau)$, these equations are to be solved subject to the initial condition that the shell is at rest at $\tau = 0$.

Let $\psi(\tau)$ be defined by

$$\begin{aligned} \psi &= \psi_i(\tau) \text{ for } 0 \leq \tau \leq \tau_m; \\ \psi &= \psi_d(\tau) \text{ for } \tau \geq \tau_m, \end{aligned} \quad (2)$$

where ψ_i and ψ_d are monotonically increasing and decreasing functions of time, respectively, and

$$\psi_i(\tau_m) = \psi_d(\tau_m) = \psi_m > 1. \quad (3)$$

In other words, the loading increases to a maximum value ψ_m and decreases from then on.

No deformation occurs until the time when $\psi = 1$, because the stresses are below the yield point. At the instant $\psi = 1$, the shell begins to deform in a Phase-1 pattern. Then Eqs. (1) must be solved numerically subject to the appropriate Phase-1 restrictions:

$$\eta_1(\tau) \equiv 0; \quad \dot{\eta}_1(\tau) \equiv 0; \quad \psi(\tau) \xi_1(\tau) < 3/2. \quad (4)$$

If the above inequality is satisfied for the entire interval $0 \leq \tau \leq \tau_m$, the shell motion remains in Phase 1 for all time.

If, however, $\psi \xi_1 = 3/2$ at some time τ_0 for which $0 \leq \tau_0 \leq \tau_m$, the deformation switches to a Phase-2 pattern at this instant. Equations (1) must then be solved subject to the initial conditions

$$\xi_1(\tau_0) = \alpha; \quad \eta_0(\tau_0) = \beta; \quad \eta_1(\tau_0) = 0, \quad (5)$$

where α and β are known from the numerical solution for Phase 1. A closed-form solution of Eqs. (1) has been found for the Phase-2 deformation pattern. It is given by

$$\begin{aligned} & \text{(a) } \underline{\tau_0 \leq \tau \leq \tau_m} \\ & \xi_1(\tau) = \frac{3}{2\psi_1(\tau)}, \\ & \eta_0(\tau) = \beta - (\tau - \tau_0) + \frac{4}{3} \int_{\tau_0}^{\tau} \psi_1^2 d\tau, \\ & \eta_1(\tau) = \eta_0(\tau) - \frac{\beta}{\alpha} \xi_1(\tau) - \frac{8}{9} \xi_1(\tau) \int_{\tau_0}^{\tau} \psi_1^3 d\tau; \end{aligned} \quad (6)$$

$\xi_2(\tau)$ is found from

$$\beta \left(1 - \frac{\xi_2}{\alpha}\right) - (\tau - \tau_1) + \frac{4}{3} \int_{\tau_0}^{\phi(\xi_2)} \psi_1^2 d\tau - \frac{8}{9} \xi_2 \int_{\tau_0}^{\phi(\xi_2)} \psi_1^3 d\tau = 0, \quad (7)$$

with ϕ defined by

$$\phi(x) = \psi_1^{-1} \left(\frac{3}{2x} \right). \quad (8)$$

$$\text{(b) } \underline{\tau > \tau_m}$$

$\xi_1(\tau)$ is found from

$$\begin{aligned} & 3\phi(\xi_1) - 3\tau + 2\xi_1 \int_{\phi(\xi_1)}^{\tau} \psi_1 d\tau + 2\xi_1 \int_{\tau_m}^{\tau} \psi_d d\tau = 0, \\ & \eta_0(\tau) = \beta - (\tau - \tau_0) + \frac{3}{\xi_1^2} [\tau - \phi(\xi_1)] + \frac{4}{3} \int_{\tau_0}^{\phi(\xi_1)} \psi_1^2 d\tau, \\ & \eta_1(\tau) = \eta_0(\tau) - \frac{\beta}{\alpha} \xi_1 + \frac{3}{\xi_1^2} [\phi(\xi_1) - \tau] - \frac{8}{9} \xi_1 \int_{\tau_0}^{\phi(\xi_1)} \psi_1^3 d\tau; \end{aligned} \quad (9)$$

$\xi_2(\tau)$ is given still by Eq. (7).

B. Operations

1. TREAT Operations (J. F. Boland)

Last Reported: ANL-7399, pp. 167-168 (Nov 1967).

a. Reactor Operations. A wireway was installed from the collimator of the fast-neutron hodoscope* to a room which is being modified for permanent installation of the readout equipment for the hodoscope.

A concrete-block shielding wall is being installed in the equipment room for the large TREAT sodium loop in order to shield components which are expected to contain radioactive and contaminated sodium.

Neutron radiographs were made of several melt-wire temperature determination capsules that are prototypes of those being considered for use in EBR-II.

b. Development of Automatic Power Level Control System.

A temperature-limited transient with an energy release of 1000 MW-sec was run to obtain power and energy-release data on magnetic tape. A relationship between reactivity feedback and energy release that is suitable for use in a computer simulation of the reactor was obtained from these data. Work continued on the development of a reactor-kinetics subroutine and reactivity-feedback subroutine for the IBM "Continuous System Modeling Program."

A standard EBR-II fuel element with a hole in the gas plenum was irradiated to determine if a detectable amount of fission gas would be released from the element during or following a short steady-state irradiation in TREAT. A very small amount of activity was detected during a 2-hr period after the first irradiation. This small release was followed by a sudden 200-fold increase in fission-gas-trap activity. Four days later, this same pin was irradiated again, and the maximum fission-gas-trap activity occurred shortly after reactor shutdown. The reason for the delayed fission-gas release following the first irradiation is unknown at present, but one possibility is that a sodium oxide layer on top of the bond sodium may have trapped the gas until the oxide dissolved in the liquid sodium, which was maintained at 250°C following the irradiation.

This test was made as a first step in determining the feasibility of making tests in TREAT to locate leaky fuel elements from EBR-II subassemblies.

*Reactor Development Program Progress Reports, ANL-7249, p. 78 (Aug 1966) and ANL-7371, p. 100 (Aug 1967).

C. Chemical Reaction--Research and Development

1. Chemical and Associated Energy Problems (Thermal)

a. Analysis of Loss-of-coolant Accidents (J. C. Hesson)

Last Reported: ANL-7399, pp. 168-169 (Nov 1967).

Calculational studies of the effect of the reaction of steam with cladding and core materials in a loss-of-coolant accident have continued. The models and methods for making these calculational analyses have become progressively more sophisticated. The latest method of analysis, CHEMLOC-II, extends the calculations of core heating and chemical reaction through core melting, core movement (slumping), and heating of the bottom grid plate. A detailed description of the CHEMLOC-II program was given previously (see Progress Report for May 1967, ANL-7342, p. 117) along with the results of CHEMLOC-II computer runs for three reactors (a 2255-MWt boiling water reactor; a 2758-MWt pressurized water reactor; and the 50-MWt Loss of Fluid Test (LOFT) reactor).

The CHEMLOC-II program requires a number of input parameters such as the slump (melting or collapse) temperature of the fuel and cladding material, the fraction of the core material that falls to the bottom of the core when an increment of rod slumps, the fraction (if any) of slumped material that falls completely out of the core, the extent of increase in axial heat transfer caused by running or dripping of the melting core material, the amount of slumping required to stop steam flow in a given zone, and a number of other factors. The lack of experimental results makes the choice of values for these factors somewhat arbitrary.

In addition, some of the physical properties of the core materials are not well-known. These include specific heats and thermal conductivities at high temperatures, as well as latent heats of fusion of the fuel, cladding, and other core materials and the mixed oxides formed in the collapse of the reactor core. In order to determine the effects of varying the input parameters, additional computer runs are being made.

CHEMLOC-II computer runs have now been made in which input parameters were chosen to be typical of a 3300-MWt boiling water reactor. Three different core slumping temperatures were assumed: 1750, 2200, and 2800°C. Since the actual slumping temperature of a core has not been determined, these three values were selected to represent the range from the lowest to the highest feasible temperatures.

The low value of 1750°C was chosen because it is below the melting point of zirconium (1852°C) and because it is possible that physical failures not requiring melting may occur or that eutectics may form having

melting points below that of zirconium. The high value, 2800°C , was chosen because this is the melting point of UO_2 . Thus, these temperatures establish maximum and minimum feasible slumping limits.

The intermediate value of 2200°C was selected because recent tests involving induction heating of Zircaloy-2-clad UO_2 -core fuel rods (in flowing steam) indicated some initial fusing of the rods at 2140°C . However, in the environment of a reactor core, higher temperatures may be more realistic. Since we define slumping temperature as the average temperature at which a fairly large amount of fuel undergoes a gross change in geometry, then temperatures representing the melting of a mixture of UO_2 and the partially oxidized cladding are probably more appropriate for use in the program. These mixtures of UO_2 and partially oxidized cladding will have melting temperatures somewhere between that of the pure oxidized materials (ZrO_2 , $\sim 2700^{\circ}\text{C}$; UO_2 , $\sim 2800^{\circ}\text{C}$) and that noted above for initial fusing ($\sim 2200^{\circ}\text{C}$).

Table V.C.1 presents the dimensional parameters for the reactor core chosen; Table V.C.2 shows the assumed values for the important variable parameters in the calculations. The rate of steam generation is based on a reactor operating temperature of 285°C (1000 psig saturated steam pressure) and a water boil-off temperature of 127°C . The lower pressure-vessel head was considered to be full of water at the end of blowdown. A boil-off temperature of 127°C was chosen because it corresponds to a saturated steam pressure of 20 psig, which is the assumed pressure in the reactor vessel after blowdown.

TABLE V.C.1. Assumed Dimensional Parameters for
a 3300-MWt Boiling Water Reactor Core

Power, MWt	3300
ID of Reactor Vessel, in.	251
Reactor Pressure, psig	1000
<u>Fuel Rods (UO_2)</u>	
No. of Fuel Rods	37436
No. of Fuel Assemblies	764
Length of Fuel, in.	144
Thickness of Zircaloy Cans, in.	0.085
OD of Zircaloy Cladding, in.	0.562
Cladding Thickness, in.	0.032
Fuel Pellet Diameter, in.	0.488
Pitch (Square Array), in.	0.738
<u>Control Rods</u>	
No. of Assemblies	185
Rods per Assembly	1
Control-rod Material	Boron in Stainless Steel
Thickness, in.	0.312
Width, in.	9.75, cruciform

TABLE V.C.2. Assumed Values for Variable Parameters for
a 3300-MWt Boiling Water Reactor Core

Specific Reactor Operating Power, cal/cm ³	47.6
Steam Evaporation Temperature, °C	127
Ambient Temperature, °C	285
Slumping Temperature ^a of Fuel-cladding Mixture, °C	2800, 2200, 1750
Fraction of Core Heat Retained at End of Blowdown, %	~75
Average Fuel Temperature at End of Blowdown, °C	646
Weights: Bottom Grid Plate, g	18.04 x 10 ⁶
Top Grid Plate, g	8.33 x 10 ⁶
Core Wall, g	24.33 x 10 ⁶
Radiation Emissivity of Cladding	0.60
Ratio of Heat Conductivity of Molten and Solid Fuel Mix	2.00
Fraction of Core Material Falling to Bottom of Core on Slumping	0.40
Fraction of Core Material Falling from Core on Slumping	0.02
Slump Factor	0.50
Fraction Slumping in a Zone Required to Stop Steam Flow	0.16
Drip Factor	10.0
Steam Flow Rate (Variable), lb/hr, at Indicated Time (sec)	
after Blowdown: 0	57600
25	40729
50	33255
100	25729
500	12569
1000	8670
2000	4305
Core Decay Heating (Variable), % of Reactor Operating Power, at Indicated Time (sec) after Blowdown:	
0	4.80
25	4.30
50	4.05
100	3.63
500	2.61
1000	2.20
2000	1.87
4000	1.58

^aThree different cases.

It should be recognized that the values of the parameters which govern the core slumping model (as shown in Table V.C.2) are chosen somewhat arbitrarily, although within reasonable limits. The purpose of the calculations is to examine the sensitivity of the results to the variation of specific parameters, in this case the slumping temperature, and not to perform an analysis for a specific reactor.

The computed axial variation of fuel temperatures are fairly similar for the three slumping temperatures, particularly for periods of time less than 200 and more than 1000 sec. There is a thermal arrest at the slumping temperature in each case. When slumping of the fuel starts near the axial center of the zone, there is a rapid rise in fuel temperature near the bottom of the zone. The temperature increase, in each case, is

the result of part of the slumping fuel falling to the bottom of the core. (This effect occurs earlier for the lower slumping temperatures.)

The computed values of percent cladding reacted, percent can material (reactable material other than cladding) reacted, quantity of hydrogen released, steam flow rate, and decay heat for the three slumping temperatures are very similar for periods of time less than 600 sec. However, when a sufficient fraction (here assumed to be 16%) of the fuel in the zone has slumped, the steam flow to that zone is blocked. The steam flow from a blocked zone is diverted to unblocked zones and steam-cladding reaction ceases in the blocked zones. When all zones are blocked, no further reaction of cladding and can material occurs, and hydrogen release ceases. This occurs sooner for lower slumping temperatures.

The fractions of cladding and can material oxidized, as well as the quantity of hydrogen produced, are primarily dependent upon the steam supply to the surfaces of these oxidizable materials. Hence, if low slumping temperatures cause an early total steam blockage, the chemical reaction, and thus the amount of hydrogen produced, may be greatly reduced. The computed extent of chemical reaction and quantity of hydrogen produced for the slumping temperature of 1750°C are only about 45% of those for the slumping temperature of 2800°C. The major effect of low slumping temperatures is to lessen the quantity of hydrogen generated.

The calculated values of percent hydrogen in the exit gas show a temporary drop (at about 600 sec) for the 1750°C slumping temperature; this is due to the early fuel slumping in the central zones, which forces the steam to the outer zones when the cladding is not yet at sufficient temperature for complete cladding-steam reaction.

The computer results show that the average fuel temperature is only slightly affected by the slumping temperature, although the steam flow to the various zones is blocked earlier in the cases of lower slumping temperatures. The average wall and top grid plate temperatures are relatively unaffected by slumping temperature; however, the average temperature of the bottom grid plate is significantly affected. At lower slumping temperatures the bottom grid plate heats up faster, due to hot slumped material falling to the bottom of the core. This early heat-up indicates that the bottom grid plate would fail earlier in the case of lower slumping temperatures.

The bottom grid plate may fail (and allow the slumped core to fall into the lower vessel head) when the center of the grid plate reaches a temperature of about 1050°C. At this temperature a supporting grid structure made of stainless steel would be considerably weakened and might collapse. This center temperature of 1050°C would correspond to an average temperature of the grid plate of about 670°C.

Based on the input parameters used, the estimated time when the grid plate would reach these temperatures for the three cases of assumed slumping temperature would be:

Slumping Temperature (°C)	Time for Initial Failure of Grid Plate (sec)
1750	1850
2200	1950
2800	2250

The computations, based on the assumed slumping model and input parameters used, indicate that the main effects of changing the assumed slumping temperatures are as follows:

1. At lower slumping temperatures, the percent cladding and can reacted (hydrogen released) may be significantly less. The differences in the quantity of hydrogen produced may be on the order of 100%.
2. At lower slumping temperatures, the temperature of the bottom grid plate increases more rapidly and reaches a failure temperature earlier. However, the differences in time to reach failure temperatures are only of the order of 20%.

The most important effect of slumping temperature appears to be in the quantity of hydrogen released. Additional computer runs to check the effects of variation in other input variables (sensitivity analyses) are planned.

b. Pressure Generation due to Particle-coolant Energy Transfer
(R. O. Ivins)

Last Reported: ANL-7371, pp. 111-114 (Aug 1967).

In order to analyze adequately the consequences of an accident in a water-cooled thermal reactor in which hot fuel materials may be injected and dispersed into the water coolant, a knowledge of the manner in which large amounts of energy are transferred from these particles to the liquid coolant is needed. Such a process can occur in either an excursion or a loss-of-coolant accident and result in an explosive generation of steam.

(i) Experimental Analysis. An experimental program has been initiated to measure the forces generated during heat transfer from hot spheres falling through a column of water, thus simulating the effect of hot fuel particles being injected into water. The forces being generated are

caused by a momentum change of liquid being rapidly displaced upward by void (or bubble) formation. Apparatus has been designed to give a quantitative measure of the dynamics of this void formation. The effects of initial sphere temperature and water temperature on the measured forces are being investigated.

The primary apparatus used in the experimental program consists of an 8-in.-long quartz tube (of 1-in. ID) filled with distilled water and mounted on a Kistler quartz-crystal piezoelectric transducer. The quartz tube is balanced on the transducer so that all vertical forces are transmitted directly to the transducer. In this situation, the quartz-crystal transducer functions as a force-measuring instrument. The output of the transducer, recorded on an oscilloscope via a Kistler charge amplifier, was calibrated as a function of weight (force) added to the system. Because the force transducer in this apparatus was designed to measure a reaction force as the water column acts against it, the apparatus has been called the reaction-force-column apparatus.

Current experiments are being performed using a 1/4-in.-dia platinum sphere. The sphere is heated electrically in a furnace consisting of an alumina tube wound with Nichrome wire. The sphere temperature, which has been varied from 100 to 1000°C, is measured with a chromel-alumel thermocouple adjacent to the sphere in the heater. The dimensions of the heater are such that a relatively constant temperature is maintained in the sphere-heating region. The sphere is released from the heater upon reaching the desired initial temperature; the release mechanism triggers the oscilloscope via a photocell circuit. As the sphere drops through the water column, the transducer output is recorded by Polaroid photographs of the oscilloscope trace. An electrical heater, fitted loosely around the water column, is used for the heated-water experiments (at 60 and 98°C). In all experiments thus far, the distance from the heater to the water surface has been kept constant; hence, the particles have entered the water at approximately the same velocity, ~3.40 ft/sec (this was verified by high-speed motion pictures).

A second apparatus of different design has also been employed. In this apparatus, a Kistler pressure transducer was made an integral part of the water-tube system. Hence, this system was designed to measure pressure increases in the water and not the reaction forces of the complete water-tube system. Both systems, which are freely vented to the atmosphere, allow liquid to be displaced by vapor without any pressurization effects.

The force measured during each drop of the sphere through the water column is plotted in Fig. V.C.1 as a function of sphere temperature and water temperature. The data were obtained from the oscilloscope records as the sphere first entered the water so that a temperature correction

due to cooling in the water would not have to be applied. Averaged force values were obtained from the records over a time period short enough so that the sphere temperature would be considered constant.

Fig. V.C.1
Forces Generated by 1/4-in.-dia
Heated Platinum Sphere Falling
through Water

Figure V.C.1 shows that relatively small forces were measured as high-temperature ($>500^{\circ}\text{C}$) spheres contacted the water; much higher forces were measured at lower sphere temperatures (100 to 500°C). The sphere temperature at which the transition to larger forces occurs is highly dependent upon the water temperature. As the temperature of the water is increased, this transition temperature decreases and approaches the boiling point of water, 100°C .

As stated above, the data points in Fig. V.C.1 were taken as the sphere first entered the water. However, it was observed that as the sphere traversed the tube and cooled from initial temperatures above approximately 500°C (in 23°C water), the time from entry into the water until a transition to larger forces occurred was a smooth function of the initial sphere temperature. Figure V.C.2 shows this relationship graphically and indicates that the length of time to this transition to larger forces is dependent upon the surface temperature of the sphere. Thus, if a sphere at a relatively high temperature is dropped into the water, it takes longer for this sphere to

Fig. V.C.2. The Effect of Sphere Temperature
upon the Time from Entry until
Transition (23°C water)

cool and to reach the required transition surface temperature than for a sphere dropped at a lower temperature.

High-speed motion pictures were taken of a heated sphere falling through water at various temperatures. These motion pictures were made to relate the measured forces to the regimes of boiling (film, transition, and nucleate) that would occur as hot particles come into contact with water. Table V.C.3 summarizes the conditions of the experiments and the results obtained from the motion-picture sequences. It should be noted that for sequences shot near the bottom of the tube, the sphere had cooled substantially from the initial temperature.

TABLE V.C.3. Conditions and Results of Motion-picture Sequences of a 1/4-in.-dia Platinum Sphere Falling through Water (Sphere velocity = 3.40 ft/sec)

Water Temperature (°C)	Initial Sphere Temperature (°C)	Position in Tube	Boiling Regime	Relative Forces
25	25	top	a	none
25	500	top	nucleate	large
25	820	top	film	small
60	800	top	film	small
60	820	top	film	small
95	820	top	film	small
98	830	top	film	small
28	820	bottom	nucleate	large
30	840	bottom	nucleate	large
60	820	bottom	nucleate	large
95	820	bottom	film	small
95	820	bottom ^b	transition	c

^aNonboiling.

^bAfter 17 sec.

^cNot measured.

The film sequences of a cold sphere falling through 25°C water show that only a small amount of air is entrained as the sphere enters the water. For a sphere initially at 500°C being dropped into 25°C water, vapor is formed irregularly on the sphere surface as discrete bubbles and "patches" with very short lifetimes. This distinctive type of behavior is characteristic of the transition from film to nucleate boiling and corresponds to rather large measured forces. For a sphere at 820°C falling through 25°C water, the vapor "separates" from the sphere surface, at a position $3\pi/4$ radians from the forward point, with a relatively small wake region.

In contrast, a sphere at 820°C falling through 95°C water shows a wake region extending several diameters behind the sphere, with "separation" occurring near the $\pi/2$ position.

These experiments have established that, at high temperatures, film boiling is the mechanism for heat transfer from the sphere to water. The shape of the vapor wake and the point on the sphere surface at which the vapor film grows very thick are dependent on the temperature of the water. During film boiling, the vapor occupies essentially a constant volume, i.e., the rate of evaporation near the sphere surface is approximately equal to the rate of condensation in the wake region.

The mechanism of vapor production as the sphere comes to rest at the bottom of the tube has also been recorded photographically for several conditions. One sequence of interest shows the rapid collapse and re-establishment of a vapor film around a sphere that was dropped at 830°C into 30°C water. The vapor film was observed to collapse completely and re-establish itself in approximately 0.5 msec. The maximum thickness of vapor film was a substantial fraction of the sphere radius. The violence of this oscillation caused the sphere to spin very rapidly.

No measurable signals were recorded with the second apparatus, which was designed to measure pressure increases in the water, when a sphere was dropped into the water under conditions where the reaction-force-column apparatus had recorded relatively large forces. It was concluded that the forces were being generated by the momentum change of liquid water being displaced upward by the formation of vapor around the sphere.

(ii) Theoretical Analysis. A simplified model for predicting forces due to vapor formation has been developed. The vapor volume V_i of a bubble forming or collapsing at a depth h_i is assumed to displace an equal volume of liquid which moves across an imaginary plane, X_i . The weight W_i of liquid above the bubble is

$$W_i = \rho A h_i, \quad (1)$$

where A is the cross-sectional area of the liquid column and ρ is the density of the liquid. The velocity of the liquid, U_i , in the vertical direction due to the change in volume of the bubble is

$$U_i = \frac{1}{A} \frac{dV_i}{dt}, \quad (2)$$

where t is time.

The dynamic vertical force F_i on the liquid-column support (which for measurement is a transducer) due to the change in volume of the bubble is

$$F_i = \frac{1}{g} \frac{d}{dt} (W_i U_i) = \frac{\rho}{g} \frac{d}{dt} \left(h_i \frac{dV_i}{dt} \right), \quad (3)$$

where g is the gravitational conversion factor. It will be noted that A does not appear in the final equations for force.

The total dynamic vertical force F due to the change in volume of all bubbles is

$$F = \sum F_i = \frac{\rho}{g} \sum \frac{d}{dt} \left(h_i \frac{dV_i}{dt} \right) = \frac{\rho}{g} \sum \left(h_i \frac{d^2 V_i}{dt^2} + \frac{dh_i}{dt} \frac{dV_i}{dt} \right). \quad (4)$$

For the case where all bubbles form and collapse at the same depth,

$$F = \frac{\rho}{g} \sum h_i \frac{d^2 V_i}{dt^2}, \quad (5)$$

which gives the dynamic force as a function of void volume-time relationship.

If the bubbles are spheres of radii r_i , then

$$V_i = (4/3) \pi r_i^3 \quad (6)$$

and

$$F = \frac{\rho}{g} \frac{4}{3} \pi \sum h_i \frac{(d^2 r_i^3)}{dt^2}. \quad (7)$$

If h_i and r_i are measured in cm, t in sec, and ρ in g/cc, the g will be 1 and F will be in dynes.

The apparent dynamic pressure P at the bottom of the liquid column would be

$$P = F/A. \quad (8)$$

A computation of forces based on Eq. 5 requires a knowledge of void dynamics at the sphere surface. Analyses of the available motion-picture sequences have been adequate to establish the boiling regimes, but more photographic data and analyses are required to obtain sufficient information regarding the void dynamics to correlate quantitatively the measured force with void formation.

Gunther* has obtained bubble radius-time relationships covering the growth and collapse of the bubbles for subcooled nucleate boiling from a plane surface with water flowing at 10 ft/sec. One of the cases that he studied was for 27.7°C water; this was assumed to be closely related to our case of 23°C water. From his graphs of bubble radius versus time, an estimate of the maximum values of d^2V/dt^2 for individual bubbles can be obtained. By substituting these values of d^2V/dt^2 in Eq. 5, it was found that the simultaneous growth and collapse of about six bubbles would be capable of giving dynamic force pulses of about one pound; this is in the range measured in our experiments as being associated with transition boiling. Since Gunther's graphs of bubble radius versus time indicate an average bubble lifetime of about 300 msec, the random, simultaneous growth and collapse of six bubbles seems reasonable.

Direct temperature measurements of the sphere temperature were not made during the experimental studies since thermocouple wires attached to the sphere would have impeded the motion of the sphere through the water column. However, as noted previously, the time (measured from the point of entry into 23°C water) required for a sphere above 500°C to give relatively large forces is a smooth function of the initial sphere temperature (see Fig. V.C.2). This suggests that the surface temperature of the sphere is the important variable in the collapse of the vapor film. When a sphere is dropped at a higher temperature, it takes longer to cool to the surface temperature (where instability of the film sets in) than for a sphere dropped at a lower temperature.

If a suitable heat transfer coefficient h could be selected to represent the subcooled film boiling process, it might be possible to predict rather than measure the transition temperature ($\sim 500^\circ\text{C}$ for 23°C water). Heat-transfer calculations (based upon the above postulation) have been performed with good success. For the basis of this calculation, it was assumed that the heat-transfer process over the temperature range of interest could be characterized by a constant value of h . For this case, a solution for the transient heat-conduction equation has been obtained,** assuming that angular symmetry exists. The solution for the sphere temperature consists of an infinite series dependent upon h , position in the sphere, time, and the thermal properties of the sphere material. It was found that for the range of h connected with this study, five terms of the series were more than adequate for convergence even at very short times. The computing technique consisted of selecting an average value of h for some initial sphere temperature and computing the sphere surface temperature at the end of the transition time, as given in Fig. V.C.2. If

** Gunther, F. C., Photographic Study of Surface-Biology Heat Transfer to Water with Forced Convection, Trans. ASME 73, 115 (1951).

** Gröber, H., Erk, S., and Grigull, U., Fundamentals of Heat Transfer, McGraw-Hill, New York (1961).

the correct value of h were selected, the predicted sphere-surface transition temperature should agree with that observed during the experiments. If it did not, a different value of h was selected and the computation was repeated.

It was found for this set of experiments that a heat-transfer coefficient of $0.21 \text{ cal}/(\text{cm}^2)(\text{sec})(^\circ\text{C})$, applied over the times from sphere entry into the water to the transition to larger forces, predicted approximately 500°C as the transition temperature. This value agrees well with our experimental observations. Therefore, for a particle of known characteristics under conditions for which the vapor-film-instability criteria are known (velocity, particle temperature, and coolant temperature), the time required to reach the transition temperature (i.e., the temperature where vapor is being formed in a mode characterized by the generation of relatively large forces) can be predicted if the heat-transfer coefficients are known.

D. Effluent Control--Research and Development--Gaseous Effluent Studies

1. Plutonium Volatility Safety

a. Plutonium Hexafluoride Safety (R. W. Kessie)

Last Reported: ANL-7391, pp. 180-181 (Oct. 1967).

The volatility of PuF_6 , combined with the high toxicity of plutonium, requires that a very high degree of air cleanup be provided in the event of an accidental release of PuF_6 in an enclosure. The hydrolysis and filtration of low concentrations of PuF_6 is being studied.

In an experiment simulating continuous steady-state release, for example, a small leak of PuF_6 from equipment inside a glovebox, PuF_6 was first diluted with very dry air, then rapidly mixed with air of controlled moisture content. The following reaction occurs:

The resulting aerosol of PuF_6 hydrolysis product was then passed through a series of pretested high-efficiency filters (10-cm diameter). The quantity of plutonium on each filter was determined by standard analytical procedures. Each experiment continued for 1 hr. The range of the variables were: pressure of PuF_6 , 10^{-4} to 0.3 Torr; pressure of H_2O , 0.3 to 5 Torr; air velocity, 0.25 to 2.5 cm/sec.

The penetration of four filters in series was measured in each of 15 tests. The fraction of the released plutonium penetrating the first filter was normally 10^{-6} to 10^{-9} . The fraction penetrating the first two filters ranged from 10^{-9} to $<10^{-12}$.

The penetration under these conditions of operation (continuous release of PuF_6) was about three orders of magnitude less than in previous experiments involving burst release of PuF_6 . These results lend further confidence to the ability to contain accidentally released PuF_6 .

E. Engineered Safety Features Technology--Research and Development

1. Hydrodynamic Response of Primary Containment to a High-energy Excursion (S. H. Fistedis)

Last Reported: ANL-7391, pp. 182-183 (Oct 1967).

In the basic two-dimensional hydrodynamic equations for shock waves in a compressible fluid, there are five unknowns (density of the fluid (ρ), radial and axial velocities of the fluid (u, v), pressure in the fluid (P), and fluid energy (E)) and four partial differential equations. To obtain a unique solution to these equations, a relationship between the various unknowns, called a constitutive equation, must be established by thermodynamic theory. Once such an equation has been established it can be used for the numerical (finite difference) solution of the hydrodynamic equations, since it is possible for the computer to keep track of the detailed history of each mass point. To this end, an extensive survey of reported work was conducted to determine the best equation of state for use in the two-dimensional hydrodynamic computer code, taking into account the various pressure levels, structural materials, and coolant to be used in the postburst studies.

For shock waves in compressible fluids, the first step is to determine a single-valued equation of state from measurements with liquids and solids subjected to strong shock waves from high explosives. It is assumed that in the experiments the shock-wave profile remains constant in time, equilibrium is achieved quite rapidly (in 10^{-6} sec) behind the shock front, and the shear strength of the material is negligible. With these assumptions, the Rankine-Hugoniot conditions lead to an equation of the form

$$E_H - E_O = \frac{1}{2} (P_H + P_O)(V_O - V), \quad (1)$$

where the subscripts O and H refer to the material ahead of and behind the shock front, respectively, and V is the specific volume ($1/\rho$).

Equation 1 represents the loci of all states which may be reached from the unstressed state. To provide a complete constitutive equation, a relationship must be found between the unknowns P , ρ , and E for states neighboring the Hugoniot that can be reached by neighboring adiabats or isotherms. The usual procedure is to assume a functional equation of

state based upon solid-state quantum theory or thermodynamic agreements. The equation of state that holds for the purposes of this code is the Mie-Grüneisen equation*

$$P_H - P_K = \frac{\gamma(V)}{V} (E_H - E_K); \quad (2)$$

$$P - P_H = \frac{\gamma(V)}{V} (E - E_H); \quad (3)$$

$$\gamma(V) = V \left(\frac{\partial P}{\partial E} \right)_V = \frac{V}{C_V} \left(\frac{\partial P}{\partial T} \right)_V = \frac{V}{C_V} \left(\frac{\partial P}{\partial V} \right)_T \left(\frac{\partial V}{\partial T} \right)_P, \quad (4)$$

where the subscript K refers to the °K isotherm, and the subscript H refers to the Hugoniot term; γ is the Grüneisen ratio, the ratio of the thermal component of pressure to the lattice vibrational-energy density, and C_V is the specific heat at constant fluid volume. P_H and $\gamma(V)$ are analytically fitted to the experimental measurements in the form

$$P_H = A + B\mu + C\mu^2; \quad (5)$$

$$\gamma(V) = \gamma_0 + D\mu + E\mu^2; \quad (6)$$

$$\mu = \frac{\rho - \rho_0}{\rho_0}, \quad (7)$$

where A, B, C, D, and E are constants. By means of Eq. 5, E_H is calculated from Eq. 1. For the materials and pressures being considered, it is sufficiently accurate to take $\gamma = \gamma_0$ (the Grüneisen ratio when $P = T = 0$), which leads to a simplification of the equation of state. Thus for compressive effects, Eqs. 1, 2, 3, 5, and 6 taken together provide a complete description of the constitutive relationship needed for the code calculations. For the isentropes (expansion of shocked material), the Mie-Grüneisen equation becomes

$$P_A - P_H = \frac{\gamma(V)}{V} \left[\int_V^{V_0} P_A dV - E_H \right], \quad (8)$$

where P_A is the adiabatic pressure. For the materials and pressures being considered, the entropy change is negligible, so the right-hand side of Eq. 8 is zero and

$$P_A = P_H \text{ (for isentropic behavior).} \quad (9)$$

*Hermann, W., Stress Wave Propagation and Spallation in Uniaxial Strain, ASD-TR-62-399 (Sept 1962), AD-288885.

Hence, Eq. 5 provides a complete constitutive equation for the code calculations during the expansion phase. Table V.E.1 shows examples of the experimental results. The tables were analytically fitted by a least-squares technique according to Eqs. 5 and 7. From thermodynamic and solid-state theories, γ_0 was calculated to be 2.14 by thermodynamic theory and 2.16 by solid-state theory.* To complete the calculations, E_H is found using Eq. 1. Thus one now has a constitutive relationship to use in the two-dimensional hydrodynamic codes. The same methods can be applied to other materials and liquids.

TABLE V.E.1. The Experimental Hugoniot Data for Steel and Iron at High Pressures

Pressure P (megabars)	Shock-wave Velocity u_s (cm/sec)	Particle Velocity u_p (cm/sec)	Relative Volume (V/V_0)
<u>Material: Iron^a</u>			
0.477	0.577	0.109	0.804
0.843	0.654	0.164	0.749
0.857	0.657	0.166	0.748
0.911	0.665	0.174	0.738
0.943	0.671	0.179	0.733
0.968	0.663	0.186	0.720
1.024	0.689	0.189	0.726
1.033	0.695	0.189	0.728
1.267	0.742	0.217	0.707
1.276	0.742	0.219	0.705
1.397	0.766	0.232	0.697
1.393	0.758	0.234	0.692
1.618	0.800	0.257	0.679
1.728	0.822	0.268	0.675
1.730	0.820	0.268	0.673
<u>Material: Mild steel^b</u>			
1.09	0.706	0.197	0.721
1.12	0.718	0.199	0.723
1.12	0.699	0.205	0.707
1.67	0.808	0.263	0.675
1.70	0.790	0.275	0.652
1.72	0.811	0.270	0.667
1.74	0.807	0.275	0.659
3.09	1.013	0.388	0.617
3.42	1.066	0.409	0.616
3.70	1.090	0.432	0.604

^aMcQueen, R. G., and Marsh, S. P., Equations of State of Metallic Elements to Two Megabars, J. Appl. Phys. 31, 1253 (1960).

^bSkidmore, L. C., and Morris, E., "Experimental Equation of State for Uranium," Thermodynamics of Nuclear Materials, IAEA, 173, Vienna (1962).

All these remarks apply to solids and liquids in which shear effects are neglected and void fractions are absent. Equations of state for porous material and material with shear effects will be covered in future work.

*Hermann, loc cit.

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