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AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY

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REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

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January 1967

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FOREWORD

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

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

The last six reports issued
in this series are:

July 1966	ANL-7245
August 1966	ANL-7249
September 1966	ANL-7255
October 1966	ANL-7267
November 1966	ANL-7279
December 1966	ANL-7286

REACTOR DEVELOPMENT PROGRAM

Highlights of Project Activities for January 1967

EBWR Plutonium Recycle

EBWR has been operating at a steady power of 42 MW since November 11, 1966. The reactor will be shut down in February 1967 so that special fuel pins can be removed for shipment to Battelle Pacific Northwest Laboratory for isotope analysis. It is expected that the average exposure in the plutonium fuel will be about 1300 MWdt at that time.

EBR-II

In preparation for Run No. 25, the core size was increased from 81 to 91 subassemblies. Core changes were made in three increments with a subcriticality determination between steps. Run Nos. 25A and 25B were made at 30 kW for short periods with special subassemblies containing a total of 82 wire detectors for the purpose of flux-mapping the enlarged core.

Since the enlarged core has moved additional driver fuel out to Row 6 locations, the question of relatively high gamma heating in blanket Rows 7 and 8 contiguous to the fuel has assumed new importance in the operation of the reactor. Therefore, detailed calculations of expected temperatures and examination of blanket elements and their contained depleted uranium which have achieved varying degrees of burnup are being made before initiating Run No. 25.

Fuel Cycle Facility

In addition to routine processing and fabrication of fuel subassemblies the following work was accomplished: (a) repair of several pieces of equipment--an Argon Cell electromechanical manipulator which was repaired in-cell, with special tools, and an Air Cell crane trolley which was transferred to the cell roof enclosure for repair; (b) testing of the Mark-IB modified fuel restrainer with a thinner shank than is now used; and (c) fabrication of 3 experimental subassemblies, 7 special subassemblies with the materials therein controlled for later evaluation, and 5 subassemblies of the half-worth type for use in making reactivity adjustments in the reactor.

ZPR-3

Experiments were continued in Assembly 48, a plutonium-fueled critical assembly with a distribution of materials similar to a large carbide reactor. This assembly is being used to make measurements relevant to

the FFTF design. Heterogeneity measurements were made using thin U^{235} , U^{238} , and manganese foils in several configurations. Relative U^{238} capture rate was measured at seven positions near the core midplane, using a method developed at ZPR-3.

ZPPR

Excavation of the reactor pit is complete. The six reactor pit posts have been steel capped and the concrete tops for them have been poured. The AFSR reactor pit and room foundations have been poured. Placement of structural steel for the support wing is 95% complete. The precast concrete floor of the support wing is in place and all walls and support columns of the vault, workroom, and inside equipment rooms have been poured.

AARR

Supplement No. 2 to the Preliminary Safety Analysis Report was transmitted to the AEC on January 20, 1967. The consultant has started developing information on the seismic exposure of the facility. Construction Package No. 1 has been approved by the Laboratory and the AEC.

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I. PLUTONIUM UTILIZATION

A. Research and Development

1. Operation of Plutonium-fueled EBWR

The EBWR facility has operated since November 11, 1966 at a power level of 42 MW thermal with a generator output of 4.5 MWe. Operational problems were all minor. By the end of the month, control rods 1 through 8 were banked at 20 in., control rod No. 9 was at 31.2 in., and all boric acid had been removed from the reactor water. This represents a control-rod bank withdrawal of approximately 0.5 in. during the month.

The power generated by the plant is as follows:

	Thermal Power	Electrical Power
During January 1967	29,220 MWh	3,349 MWh
Total Since Start of Plutonium Recycle Program	79,985 MWh	8,868 MWh

The reactor will be shut down in February so that special fuel pins can be removed for shipment to Battelle Pacific Northwest Laboratory for isotope analysis. It is expected that the average exposure in the plutonium fuel will be about 1300 MWdt at that time.

II. LIQUID-METAL FAST BREEDER REACTORS

A. EBR-II

1. Operations

Following the completion of Run No. 24, fuel-handling operations were carried out in preparation for Run No. 25, in which the core size is increased from 81 to 91 subassemblies.

Core changes were made in three increments, subcriticality being determined between steps. The initial change involved loading of five special half-worth subassemblies in Row 4, experimental subassemblies XO19, XO20, and XO16 in positions 6-D-2, 6-B-5, and 4-D-3, the replacement of nineteen subassemblies which had achieved their burnup limit, and the insertion of eight special flux wire subassemblies to be used in flux plotting in Rows 1 through 8. At this point the control rods were inserted to determine the degree of subcriticality, which was measured at 1235 lh.

Four enriched inner-blanket subassemblies were then added in Row 6, and the reactor control rods were inserted for another criticality determination. Five additional enriched inner-blanket subassemblies were loaded to fill out the sixth row; the reactor remained subcritical by about 100 lh when the control rods were inserted. One half-worth subassembly was replaced by a core-type driver subassembly for reactivity adjustment, and criticality was achieved at 2345 hrs on January 15 with an available excess reactivity of 28 lh.

The flux wire subassemblies were irradiated by operating the reactor at 30 kW for one hr (designated as Run 25A). The reactor was shut down, and the flux wire subassemblies were removed and replaced with core and blanket subassemblies normally used in these locations.

A subsequent critical determination indicated an available excess reactivity of 183 lh, adequate for Run No. 25 scheduled for 700 MWd.

All control rods were then period calibrated (see Sect. II.A.2). Another Run (25B) for 1 hr was then made to complete the flux wire irradiations.

The final core configuration is as follows: Two safety rods, ten normal control rods, an oscillator rod in control-rod position No. 8, a special stainless steel control rod for rod-drop tests in control-rod position No. 1, twelve experimental irradiation subassemblies, thirty-three normal driver core-type subassemblies, twenty-eight enriched inner-blanket subassemblies, and four special half-worth subassemblies. This accounts for the total of 91 core and inner-blanket positions.

The increase of core size from 81 to 91 subassemblies has been made to provide a reference core configuration which represents that expected to be achieved in the near future with the expected maximum number of irradiation experiments in the core.

The partially fueled subassemblies (half-worth subassemblies) are used to provide a simulation, in terms of reactivity, of typical irradiation subassemblies and thus fix the core size to a nominal final 91 subassemblies. These subassemblies contain standard Mark-IA fuel and thus are subject to the burnup limitations for this fuel. Calculations as indicated in Table I below show the difference in mixed mean sodium outlet temperatures for these partially fueled subassemblies as compared to standard subassemblies in each row.

TABLE I. Mixed Mean Sodium Outlet Temperature (1°F) of Partially Fueled (P.F.) Subassemblies

Row	45 MW			50 MW		
	P.F.	Std.	Diff.	P.F.	Std.	Diff.
1	772	844	72	780	860	80
2	771	841	70	779	857	78
3	777	853	76	786	870	84
4	791	882	91	802	902	100
5	800	900	100	811	922	111

The Irradiation Guide establishes the criteria that the temperature difference between the average mixed mean outlet temperature of an experimental subassembly and the average temperature in a given row be not greater than 100°F. It can be seen from the table that this requirement is met for all rows at 45 MW and for all but Row 5 at 50 MW. During reactor operation, the actual temperature of some of the subassemblies will be measured and the limitation on possible future higher power levels will be assessed.

Calculations have been made of the relative fission rate in the 91-subassembly core with partially fueled subassemblies and with a typical 91-subassembly core containing experimental irradiation subassemblies, using the one-dimensional SNARG code. The radial fission rates for the two cores are within 1% of each other. This would indicate that the partially fueled subassemblies will provide the same overall flux conditions as a typical irradiation subassembly. Localized perturbations have not been calculated for either case, but will be determined using the special wire subassemblies and U^{235} wire as fission-rate detectors.

From the results of the above calculations, the pertinent temperatures for the hottest fuel element in each row were computed. New values for sodium flow were also calculated for each row. These results are given in Table II for 45-MW operation. The table shows that since the flux distribution for the large core is flatter, there is a reduction in temperature through the first four rows and an increase in temperature in Rows 5 and 6. However, in no place does the maximum fuel-surface temperature exceed the value of 994°F in Row 4, as measured in the 73-subassembly reference core.

TABLE II. Comparison of Temperatures for 73- and 91-subassembly Cores--45 MWt

Row Number	1		2		3		4		5		6	
Subassemblies	73	91	73	91	73	91	73	91	73	91	73	91
Sodium Flow, gpm per Subassembly	134	133	132	131	119	118	90	89	75	74	70	69
Flow Velocity, ft/sec	22.7	22.5	22.3	22.1	20.0	19.9	15.2	15.2	12.6	12.5	11.8	11.7
<u>Temperatures (°F)</u>												
Fuel Center, max	1074	1022	1066	1012	1056	1022	1064	1041	1038	1051	961	1001
Fuel Surface, max	975	937	970	931	970	955	994	976	981	993	919	953
Sodium at Channel Outlet, max	901	874	898	870	904	885	934	920	931	941	882	911
Sodium at Sub-assembly Outlet, mixed-mean	867	844	865	841	869	853	895	882	892	900	951	875
Sodium Channel ΔT , max	201	174	198	170	204	185	234	220	231	241	182	211
Sodium Subassembly ΔT , mixed-mean	167	144	165	141	169	153	195	182	192	200	151	175
Sodium at Point of Max Fuel Temp	872	849	870	845	874	859	934	920	931	941	882	911
Clad, ID max	956	921	951	915	954	930	979	963	969	980	911	943
Clad, OD max	926	895	922	890	927	905	956	940	950	960	896	943

Calculations were also made for blanket positions.

The most important blanket positions to be considered are Rows 7 and 8, since these are regions of relatively high U^{238} fission and gamma heating with relatively low sodium flow. The maximum uranium temperature and maximum cladding-fuel interface temperatures have been calculated for the maximum blanket elements in Rows 7 and 8. U^{238} fission rates were obtained from calculations for the reference 91-subassembly core using the SNARG code. Gamma heating at these positions was estimated from assumed gamma fluxes in the core. The results of these calculations are shown in Table III for 45-MW operation.

The temperature values given in the table do not take into account any radial heat transfer even though the gradient across the subassembly

is as much as four. Hence, these values are conservative. More sophisticated calculations including radial heat flow will be performed before the power is increased above 45 MW.

TABLE III. Maximum Temperatures of Blanket Pins in Rows 7 and 8
91-subassembly Core at 45 MW

	Fuel Center Max (°F)	Fuel Surface Max (°F)	Flow (gpm)
Row 7	1063	1003	20
Row 8	994	989	6

It is not expected that there will be any marked change in the general reactor kinetics with the larger core. Transfer-function measurements made at 30 MW and 100% flow in the present core (Run No. 24 with 9 experimental subassemblies in the core region; see Sect. II. A.3.a) appear to be in general agreement with transfer-function measurements made on the original 71-subassembly core before any experiments were in the core. This would tend to indicate that the reactor kinetics are not particularly sensitive to core size providing the character of the driver fuel is not changed. Further, data calculated from the rod-drop experiments in Run No. 24 do not indicate any change in the prompt power coefficient.

2. Control-rod Calibrations

After the loading for Run No. 25 had been completed, all of the control rods were period-calibrated. The results are as follows:

<u>Control Rod No.</u>	<u>Total Worth (lh)</u>
1	Stainless Steel
2	153
3	142
4	163
5	147
6	159
7a	126
8	Oscillator
9	135
10	153
11	136
12	150

^aMark-I fuel.

The results show that the average worths of the odd-numbered rods have increased by about 9.4% from the initial 71-subassembly core loading

with Mark-I fuel, and the even-numbered rods have decreased by 1%. This is probably because the corner control rods are surrounded completely by fuel. The gain in their worths because of this more than compensates for the decrease caused by the control rods containing a smaller fraction of total fuel.

3. Reactor Physics

a. Transfer-function Measurements. Measurements made at 30 MW on Run No. 24, the 81-subassembly core, have been partially analyzed. Results to date show that the general form of the feedback is similar to data obtained at 30 MW with the 71-subassembly core. Data at 30 MW were taken at two different positions of the control-rod bank to assess the effect of expansion of the control-rod extension shaft on the feedback characteristics. The on-line digital-computer method of data acquisition and reduction appears to give more accurate results than those previously obtained using an analog computer.

In addition, zero-power data have been taken on the 91-subassembly core. The lowest frequency point was at 0.0013 cps.

Rod-drop data from Run No. 24 have been analyzed. The 45-MW data show no change in the prompt feedback coefficient. The 30-MW data have not yet been completely analyzed. Initial rod-drop data for the 91-subassembly core have been taken.

b. Flux Wires. Eighty-two wire detectors were irradiated in Rows 1 through 8 in two Runs, 25A and 25B, to measure the fission-rate distribution. Special care was taken to position wires such that data could be taken in the vicinity of the experiments. When all data are available, a complete map of the reactor through Row 8 will be available. The data from Run 25A that have been processed show very good average agreement with the SNARG one-dimensional cylindrical-geometry calculations. The local effects due to the experiment cause as much as 15% perturbation from the calculation.

c. Multigroup Cross-section Sets for EBR-II Analysis. As part of the physics effort for increasing the fuel-testing capability of EBR-II, three multigroup cross-section sets have been developed. The purpose of the effort is to construct a spatially dependent, coarse-group energy structure from a 22-group set, to be used for two-dimensional geometry.

The 22-group neutron cross sections were obtained from fundamental-mode computations in the core and blanket sections of EBR-II. The 22-group set, designated Set 236, was produced by the MC² code. Two MC² calculations, one using the core and the other the blanket composition, were performed assuming zero buckling but taking into account the

heterogeneities in the core and blanket design. The cross sections for the nuclides in the core and blanket were obtained on punched cards. The cross sections for nine additional nuclides were obtained from Set 224, a 22-group set weighted by a softer neutron spectrum. The Set 224 cross sections are considered sufficiently accurate for the present. As more accurate data become available, primarily from the ENDF-B Code, new cross sections will be added to Set 236. A description of Set 236 is given in Table IV.

TABLE IV. Description of Cross-section Set 236

Energy Group	Lower Energy Limit (eV)	Upper Lethargy Limit	Energy Group	Lower Energy Limit (eV)	Upper Lethargy Limit
1	3.68×10^6	1.0	12	1.50×10^4	6.5
2	2.23×10^6	1.5	13	9.12×10^3	7.0
3	1.35×10^6	2.0	14	4.31×10^3	7.75
4	8.21×10^5	2.5	15	2.62×10^3	8.25
5	4.98×10^5	3.0	16	2.03×10^3	8.50
6	3.02×10^5	3.5	17	1.23×10^3	9.0
7	1.83×10^5	4.0	18	9.61×10^2	9.25
8	1.11×10^5	4.5	19	5.83×10^2	9.75
9	6.74×10^4	5.0	20	2.75×10^2	10.5
10	4.09×10^4	5.5	21	1.01×10^2	11.5
11	2.48×10^4	6.0	22	2.90×10^1	12.75

Nuclide Number	Nuclide	Remarks	Nuclide Number	Nuclide	Remarks
1	U ²³⁵	EBR-II Core Spectrum Weighted	13	Ni	EBR-II Blanket Spectrum Weighted
2	Pu ²³⁹	EBR-II Core Spectrum Weighted	14	Cr	EBR-II Blanket Spectrum Weighted
3	U ²³⁸	EBR-II Core Spectrum Weighted	15	Na	EBR-II Blanket Spectrum Weighted
4	Fe	EBR-II Core Spectrum Weighted	16	Mo	Set 224, Large Carbide/Oxide Spectrum
5	Ni	EBR-II Core Spectrum Weighted	17	Nb	Set 224, Large Carbide/Oxide Spectrum
6	Cr	EBR-II Core Spectrum Weighted	18	Zr	Set 224, Large Carbide/Oxide Spectrum
7	Na	EBR-II Core Spectrum Weighted	19	B(natural)	Set 224, Large Carbide/Oxide Spectrum
8	Fission Product Pairs	EBR-II Core Spectrum Weighted	20	C	Set 224, W.B.L. Data
9	U ²³⁵	EBR-II Blanket Spectrum Weighted	21	O	Set 224, Large Carbide Spectrum
10	Pu ²³⁹	EBR-II Blanket Spectrum Weighted	22	W	Set 224, Large Carbide/Oxide Spectrum
11	U ²³⁸	EBR-II Blanket Spectrum Weighted	23	Ta	Set 224, Large Carbide/Oxide Spectrum
12	Fe	EBR-II Blanket Spectrum Weighted	24	B ¹⁰	Set 224, Cox et al., Data

By means of the 22-group structure of Set 236, a five- and six-group set were generated from the MACH 1 code. The group structure of the collapsed sets was determined by comparing calculations performed with various five- and six-group structures to those using Set 236 and then selecting the group structure producing the best comparison.

The quantities compared in the study were the real and adjoint flux spectra, and the reactivity worths of selected nuclides in the EBR-II

core. The reactivity worths were obtained in three regions of the core: Ring #1 (center), Rings #2 through #4, and Ring #5 (outer). The worths were calculated by perturbation method and were based on a 1% decrease in the atom density of the nuclides. The group structure of the five- and six-group sets selected from the study are given in Table V, and the flux spectrum and reactivity worths obtained from these sets are given in Fig. 1 and Table VI, respectively. The five-group set is designated as Set 23605 and the six-group set as Set 23606.

TABLE Va. Description of Cross-section Set 23606

Energy Group	Corresponding Set 236 Groups	Lower Energy Limit (eV)	Upper Lethargy Limit
1	1,2	2.23×10^6	1.5
2	3,4	8.21×10^5	2.5
3	5,6	3.02×10^5	3.5
4	7,8	1.11×10^5	4.5
5	9,10	4.09×10^4	5.5
6	11 through 22	2.90×10^1	12.75

Nuclide Number	Nuclide	Weighting Spectrum	Nuclide Number	Nuclide	Weighting Spectrum
1	U ²³⁵	EBR-II Core	14	Cr	EBR-II Blanket
2	Pu ²³⁹	EBR-II Core	15	Na	EBR-II Blanket
3	U ²³⁸	EBR-II Core	16	Mo	EBR-II Core
4	Fe	EBR-II Core	17	Nb	EBR-II Core
5	Ni	EBR-II Core	18	Zr	EBR-II Core
6	Cr	EBR-II Core	19	B(natural)	EBR-II Core
7	Na	EBR-II Core	20	C	EBR-II Core
8	Fission Product Pairs	EBR-II Core	21	O	EBR-II Core
9	U ²³⁵	EBR-II Blanket	22	W	EBR-II Core
10	Pu ²³⁹	EBR-II Blanket	23	Ta	EBR-II Core
11	U ²³⁸	EBR-II Blanket	24	B ¹⁰	EBR-II Core
12	Fe	EBR-II Blanket	25	EBR-II Core	-
13	Ni	EBR-II Blanket	26	EBR-II Blanket	-

TABLE Vb. Description of Cross-section Set 23605a

Energy Group	Corresponding Set 236 Groups	Lower Energy Level (eV)	Upper Lethargy Limit
1	1,2,3	1.35×10^6	2.0
2	4,5	4.98×10^5	3.0
3	6,7	1.83×10^5	4.0
4	8,9	6.74×10^4	5.0
5	10 through 22	2.90×10^1	12.75

^a Nuclide number, name, and corresponding weighting spectrum for Set 23605 are the same as those for Set 23606.

The real and adjoint flux spectra generated from Sets 23605 and 23606 agree well with those from Set 236. The overall values of reactivity are in close agreement for all sets.

TABLE VI. Reactivity Worths of Eight Nuclides in the Three Core Regions Calculated by the Three Cross-section Sets

Nuclide	$\Delta k/k/\Delta V$ Terms	Region 1			Region 2			Region 3		
		236	23605	23606	236	23605	23606	236	23605	23606
U ²³⁵	Source	-2.91×10^{-7}	-2.93×10^{-7}	-2.93×10^{-7}	-1.98×10^{-7}	-1.99×10^{-7}	-1.99×10^{-7}	-1.11×10^{-7}	-1.11×10^{-7}	-1.11×10^{-7}
	C + F	1.40×10^{-7}	1.41×10^{-7}	1.41×10^{-7}	9.51×10^{-8}	9.56×10^{-8}	9.57×10^{-8}	5.18×10^{-8}	5.17×10^{-8}	5.18×10^{-8}
	Leak	-1.33×10^{-10}	-1.33×10^{-10}	-1.34×10^{-10}	-3.86×10^{-9}	-3.86×10^{-9}	-3.90×10^{-9}	-6.34×10^{-9}	-6.34×10^{-9}	-6.42×10^{-9}
	Scat	-2.34×10^{-9}	-1.89×10^{-9}	-2.17×10^{-9}	-1.03×10^{-9}	0.77×10^{-9}	-0.95×10^{-9}	1.90×10^{-10}	2.65×10^{-10}	1.96×10^{-10}
	Total	-1.53×10^{-7}	-1.54×10^{-7}	-1.54×10^{-7}	-1.08×10^{-7}	-1.08×10^{-7}	-1.09×10^{-7}	-6.53×10^{-8}	-6.51×10^{-8}	-6.53×10^{-8}
	%	-	+0.6	+0.6	-	0.0	+0.9	-	-0.5	0.0
U ²³⁸	Source	-2.22×10^{-8}	-2.22×10^{-8}	-2.21×10^{-8}	-1.43×10^{-8}	-1.43×10^{-8}	-1.43×10^{-8}	-6.57×10^{-9}	-6.54×10^{-9}	-6.61×10^{-9}
	C + F	2.00×10^{-8}	2.02×10^{-8}	2.01×10^{-8}	1.35×10^{-8}	1.35×10^{-8}	1.34×10^{-8}	7.05×10^{-9}	7.04×10^{-9}	7.04×10^{-9}
	Leak	-1.24×10^{-10}	-1.25×10^{-10}	-1.26×10^{-10}	-3.60×10^{-9}	-3.61×10^{-9}	-3.64×10^{-9}	-5.89×10^{-9}	-5.91×10^{-9}	-5.96×10^{-9}
	Scat	-4.74×10^{-10}	-0.97×10^{-10}	-7.34×10^{-10}	2.19×10^{-10}	4.35×10^{-10}	-0.03×10^{-10}	8.05×10^{-10}	8.63×10^{-10}	6.36×10^{-10}
	Total	-2.74×10^{-9}	-2.26×10^{-9}	-2.92×10^{-9}	-4.23×10^{-9}	-3.96×10^{-9}	-4.46×10^{-9}	-4.60×10^{-9}	-4.55×10^{-9}	-4.89×10^{-9}
	%	-	-17.4	+6.6	-	-6.7	+5.4	-	-1.3	+6.3
Fe	C + F	9.10×10^{-10}	9.16×10^{-10}	9.17×10^{-10}	6.31×10^{-10}	6.32×10^{-10}	6.34×10^{-10}	3.60×10^{-10}	3.58×10^{-10}	3.60×10^{-10}
	Leak	-9.44×10^{-11}	-9.48×10^{-11}	-9.49×10^{-11}	-2.76×10^{-9}	-2.76×10^{-9}	-2.77×10^{-9}	-4.57×10^{-9}	-4.59×10^{-9}	-4.58×10^{-9}
	Scat	1.62×10^{-10}	-0.98×10^{-10}	1.83×10^{-10}	4.02×10^{-10}	2.13×10^{-10}	3.84×10^{-10}	5.88×10^{-10}	4.66×10^{-10}	5.43×10^{-10}
	Total	9.77×10^{-10}	7.23×10^{-10}	10.05×10^{-10}	-1.72×10^{-9}	-1.92×10^{-9}	-1.75×10^{-9}	-3.62×10^{-9}	-3.77×10^{-9}	-3.68×10^{-9}
	%	-	-26.1	+2.8	-	+11.3	+1.7	-	+4.1	+1.6
Ni	C + F	7.71×10^{-10}	7.63×10^{-10}	7.72×10^{-10}	5.08×10^{-10}	4.99×10^{-10}	5.08×10^{-10}	2.49×10^{-10}	2.42×10^{-10}	2.47×10^{-10}
	Leak	1.73×10^{-10}	-1.74×10^{-10}	-1.75×10^{-10}	-4.96×10^{-10}	-4.98×10^{-10}	-5.03×10^{-10}	-7.93×10^{-10}	-8.00×10^{-10}	-8.07×10^{-10}
	Scat	-1.22×10^{-10}	-1.07×10^{-10}	-1.09×10^{-10}	-4.44×10^{-11}	-3.13×10^{-11}	-3.39×10^{-11}	3.81×10^{-11}	4.43×10^{-11}	3.77×10^{-11}
	Total	6.32×10^{-10}	6.38×10^{-10}	6.46×10^{-10}	-3.18×10^{-11}	-2.99×10^{-11}	-2.94×10^{-11}	-5.06×10^{-10}	-5.14×10^{-10}	-5.23×10^{-10}
	%	-	+0.8	+2.2	-	-6.0	-8.2	-	+1.8	3.4
Cr	C + F	2.38×10^{-10}	2.40×10^{-10}	2.40×10^{-10}	1.63×10^{-10}	1.64×10^{-10}	1.65×10^{-10}	9.22×10^{-11}	9.17×10^{-11}	9.20×10^{-11}
	Leak	-2.90×10^{-11}	-2.86×10^{-11}	-2.95×10^{-11}	-8.46×10^{-10}	-8.35×10^{-10}	-8.63×10^{-10}	-1.41×10^{-9}	-1.39×10^{-9}	-1.44×10^{-9}
	Scat	1.35×10^{-10}	0.09×10^{-10}	1.10×10^{-10}	1.71×10^{-10}	0.85×10^{-10}	1.45×10^{-10}	1.93×10^{-10}	1.47×10^{-10}	1.70×10^{-10}
	Total	3.44×10^{-10}	2.20×10^{-10}	3.21×10^{-10}	-5.12×10^{-10}	-5.86×10^{-10}	-5.53×10^{-10}	-1.12×10^{-9}	-1.15×10^{-9}	-1.17×10^{-9}
	%	-	-36.0	-6.7	-	+14.2	+8.0	-	+2.4	+4.5
Na	C + F	1.40×10^{-10}	1.39×10^{-10}	1.41×10^{-10}	9.58×10^{-11}	9.49×10^{-11}	9.60×10^{-11}	5.31×10^{-11}	5.17×10^{-11}	5.22×10^{-11}
	Leak	-1.15×10^{-10}	-1.17×10^{-10}	-1.16×10^{-10}	-3.34×10^{-9}	-3.40×10^{-9}	-3.37×10^{-9}	-5.48×10^{-9}	-5.59×10^{-9}	-5.54×10^{-9}
	Scat	-2.69×10^{-9}	-2.49×10^{-9}	-2.26×10^{-9}	-1.47×10^{-9}	-1.34×10^{-9}	-1.15×10^{-9}	-2.57×10^{-10}	-2.22×10^{-10}	-0.49×10^{-10}
	Total	-2.67×10^{-9}	-2.47×10^{-9}	-2.24×10^{-9}	-4.71×10^{-9}	-4.65×10^{-9}	-4.41×10^{-9}	-5.69×10^{-9}	-5.76×10^{-9}	-5.53×10^{-9}
	%	-	-7.3	-16.1	-	-1.7	-6.4	-	+1.6	-2.8
Carbon	C + F	0	0	0	0	0	0	0	0	0
	Leak	-2.36×10^{-10}	-2.37×10^{-10}	-2.39×10^{-10}	-6.81×10^{-9}	-6.85×10^{-9}	-6.93×10^{-9}	-1.11×10^{-8}	-1.12×10^{-8}	-1.13×10^{-8}
	Scat	-1.12×10^{-8}	-1.00×10^{-8}	-1.05×10^{-8}	-6.68×10^{-9}	-5.80×10^{-9}	-6.10×10^{-9}	-1.93×10^{-9}	-1.52×10^{-9}	-1.52×10^{-9}
	Total	-1.14×10^{-8}	-1.02×10^{-8}	-1.08×10^{-8}	-1.35×10^{-8}	-1.27×10^{-8}	-1.30×10^{-8}	-1.31×10^{-8}	-1.27×10^{-8}	-1.29×10^{-8}
	%	-	-10.1	-5.3	-	-6.3	-3.7	-	-2.5	-1.5
Oxygen	C + F	0	0	0	0	0	0	0	0	0
	Leak	-4.34×10^{-10}	-4.15×10^{-10}	-4.46×10^{-10}	-1.26×10^{-8}	-1.20×10^{-8}	-1.30×10^{-8}	-2.06×10^{-8}	-1.96×10^{-8}	-2.16×10^{-8}
	Scat	-1.57×10^{-8}	-0.97×10^{-8}	-1.73×10^{-8}	-9.20×10^{-9}	-5.49×10^{-9}	-10.00×10^{-9}	-2.62×10^{-9}	-1.21×10^{-9}	-2.76×10^{-9}
	Total	-1.61×10^{-8}	-1.01×10^{-8}	-1.77×10^{-8}	-2.17×10^{-8}	-1.75×10^{-8}	-2.30×10^{-8}	-2.32×10^{-8}	-2.08×10^{-8}	-2.43×10^{-8}
	%	-	-37.3	+9.9	-	-19.3	+6.0	-	-10.3	+4.7

Fig. 1a. Comparison of Real and Adjoint Flux Spectra from Sets 236 and 23605

Fig. 1b. Comparison of Real and Adjoint Flux Spectra from Sets 236 and 23606

With only one exception, the reactivity worth ($\Delta k/k/\Delta V$) calculated for the 8 nuclides from Set 23606 agrees within 10% with those calculated from Set 236. The one exception is the worth of sodium in Region #1 (Ring #1). The discrepancy arises from the inelastic scattering term in the perturbation analysis. In fact, the greatest discrepancies for all the nuclides usually occur in the inelastic-scattering term rather than the source, absorption, or leakage terms.

The results of the reactivity worth calculations using Set 23605 are also given in Table VI. Again, the major difficulty arises in trying to predict the worth due to inelastic scattering. The discrepancies in the scattering term and total worth are generally greater for Set 23605 than for Set 23606. Oddly enough, Set 23605 gives one of its closest comparisons for sodium which Set 23606 predicts the poorest. Rearranging the group structure for the five-group set improved the comparison for particular nuclides, but resulted in an overall comparison that was poorer. It was concluded that Set 23605 resulted in the best comparison for a cross-section set limited to only five groups.

Set 236 is available on magnetic tape in the MACH format and Set 23606 is also on tape in both MACH and XLIBIT formats. A complete set of nuclides for Set 23605 will be available shortly.

4. Reactor Improvements

a. Irradiated Fuel Monitor. Preliminary design drawings and specifications of the Irradiated Fuel Monitor (or Monitor for Spent-fuel Subassembly, see Progress Report for June 1966, ANL-7230, pp. 2-3) were reviewed and minor modifications were incorporated into the final design (see Fig. 2). These were:

(i) The lower end of the thimble was shortened 14 in. to eliminate any possible interference with the subassembly catch-basin structure. This was accomplished by reducing slightly the background-radiation shielding of the lower section.

(ii) The requirement was added for a final helium leaktest of the welds before installation.

Fig. 2. Final Design for EBR-II Irradiated-fuel Monitor

Additional changes in the detail drawings were made necessary because of difficulties in purchasing manufactured parts specified in the original design. Delivery schedules for some purchased parts will be longer than expected, which will set back the completion date to approximately the last of April 1967.

b. Fuel Element Mark IB. Further tests were performed in the series being conducted to evaluate various configurations of fuel-pin restrainers (see Progress Report for October 1966, ANL-7267, pp. 8-9). A new configuration, Model #3 (see Fig. 3), having a straight shank with a crossbar at the free end, was devised to keep that end centered in the element tube. In previous tests with Models #1 and #2, a well-defined dimple was formed in the tube wall on each sample because the free end of the restrainer slipped on the end of the simulated fuel pin and was forced against the side of the tube.

Fig. 3. EBR-II Fuel-pin-restrainer Model No. 3

noted in the type of tube-wall distortion. There was no evidence of dimples in the tube wall on Model #3, which indicated that the crossbar configuration was successful in keeping the restrainer centered. A nominal 0.002-in. distortion of the tube wall was the result of the buckled wire shank forcing the tube to assume an out-of-round shape.

Other tests may be performed on different restrainer designs to develop an optimum configuration.

c. Rotating Plug Seal Cleaning System. To prevent a recurrence of the oxide buildup in the tin-bismuth eutectic alloy which serves as a seal in the dip-ring troughs, a system is being designed to agitate and pump out the oxidized alloy, which will then be purified in a cleaning system and returned to the seal troughs. Previously, the oxides were removed by a brushing and vacuuming process which eliminated the sticking problems caused by the oxide accumulation (see Progress Report for October 1966, ANL-7267, pp. 5-7).

A conceptual design has been completed and preliminary testing has started on a prototype pump, 5/8 in. in diameter, which will fit into the access holes located above each of the two seal troughs. The test pump is operated by an air-motor at approximately 10,000 rpm and delivers 1.2 to 1.5 gpm of the alloy at a 34-in. outlet head. Methods of separating the alloy metal from the oxides are being investigated to determine if a continuously circulating cleanup system can be employed or if it will be necessary to use a batch cleanup method.

d. Primary-tank Sodium Pump and Filter. Design of a sodium filter and pump assembly has started. This unit is to provide a means of determining the amount, type, and size of impurities that might exist in the region of the primary-sodium-pump intake elevation. It is planned to accomplish this by pumping a portion of the primary-tank sodium through a filter and periodically examining the filter. The pump and filter assembly will be attached to the lower end of a special plug that will be located in an existing spare nozzle of the primary-tank cover.

The pumping is to be accomplished with a piston that forces sodium downward through a cylinder into the filter section. A check valve will prevent the sodium from being drawn upward through the filter on the piston upstroke. The sodium-intake port on the cylinder will be at the same elevation as the primary-pump intake.

The preliminary concept will employ a cylinder 6 in. in diameter with a piston stroke of 18 in. At a velocity of 9 ft/min, this will provide an effective 6-gpm sodium flow through the filter. The piston drive mechanism will be a ball-nut screw design.

e. Control-rod Model IHE. Fabrication was completed on the water-loop test model of the proposed IHE control rod (see Progress Report for October 1966, ANL-7267, p.8); preliminary hydraulic flow tests were performed to evaluate the change in lifting force provided by the new design. From an analysis of the data it was apparent that no conclusive results could be obtained due to unforeseen problems imposed by design limitations of the water test loop. An available water test loop is being modified; it has the desired design features and flow capacity to furnish the more complete and accurate data necessary for a proper evaluation.

f. Irradiation Subassembly, Mark C37. Fabrication of the prototype model Mark C37 irradiation subassembly (see Fig. 4) was completed; a preliminary description can be found in the Progress Report for September 1966, ANL-7255, p. 11. Initial hydraulic tests have been started to provide information regarding the flow and pressure-drop characteristics of the subassembly at various bypass flow rates of the coolant.

Fig. 4. Prototype Model Mark-C37 Irradiation Subassembly

deflection in proportion to the amount of force. Some force is also required to move each shaft through its packing gland. This force produces more spring deflection which must be subtracted from the total deflection to obtain the net force applied inside the tank. When the net force reaches the predetermined limit, fuel-handling procedures are stopped automatically, and an alarm occurs.

The new gripper and holddown force-limit devices will provide continuous measurements of the forces applied by these mechanisms. The deflection of each force spring will be monitored by a linear-motion resistance potentiometer instead of by limit switches presently employed.

To improve the sensitivity of the gripper and holddown push-pull force-limiting device, it is necessary to minimize the effect of

The first series of tests was performed with no bypass flow in order to measure the pressure drop through the specimen tube section at various flow rates for each core position, Rows 1 through 5. These data provided the basic information for calculations, presently being made, to determine the proper internal orificing necessary for the second series of tests, which will be carried out to measure the percent of the total flow that will bypass the specimen section at the different flow rates. Thus, it will be possible to provide a range of temperatures in the specimen tube section by selecting the proper orifice sizes to give the desired values of specimen coolant and bypass flow.

g. Improved Gripper and Holddown Force-limiting Device.

A new design to improve the reliability, accuracy, and sensitivity of the core gripper and holddown push and pull force-limiting devices is being developed. The gripper and holddown shafts, which move through packing glands and extend into the primary tank, are each supported by a spring arrangement so that when they exert force within the tank there is some change in spring

friction on the force-spring carriage assembly. A special ball bearing will replace the main bronze bearing. Also, the carriage roller assemblies which ride in the shroud tube will be modified to prevent binding of the carriage assembly.

Design drawings have been initiated and some parts of the carriage guide-roller assembly are ready to be sent out for fabrication. Testing of the guide-roller design will then be performed in the force-limit test rig for evaluation of the design.

h. Instrumented Subassemblies. As one step toward advancing the usefulness of EBR-II as an irradiation test facility, instrumented irradiation subassemblies are being studied. According to the basic concept, these instrumented test subassemblies will be inserted in the core in one or more control-rod positions and, after irradiation, will be removed by using the normal EBR-II fuel-handling system supplemented by special tools to handle the instrument leads.

(i) Subassembly Attachment and Lead Severance. Emphasis is being placed on the development of the instrument lead connecting the top of the instrumented subassembly and the extension tube (see Progress Report for November 1966, ANL-7279, pp. 5-6). Most schemes investigated thus far employ continuous sheathed leads. These leads will be severed at the top of the test subassembly before removal. Then all the leads can be withdrawn, leaving the subassembly in place in the reactor, from which it will be removed by the existing fuel-handling procedure and equipment.

Another scheme under consideration is one in which an electrical connector is used within the subassembly such that more conventional (rather than sheathed) leads can be used. This scheme appears to have a number of problems, which will be more closely studied. Also being studied is a variation in which the sensor leads are terminated within a dry well above the subassembly; conventional leads also could be used for this scheme.

Of the several types of subassembly-to-extension-tube couplings being investigated, Fig. 5 shows three types: a gripper coupling, a welded coupling, and a threaded coupling. With the gripper concept, six leaves, which are welded onto the extension tube, are locked by a retaining sleeve. When it is decided to terminate subassembly irradiation, the instrument leads are severed by turning a cutter inserted into the extension tube, and then a sleeve-lifting tool (not shown) is inserted to remove the retaining sleeve by its six lifting lugs. Then the extension tube can be withdrawn, leaving the subassembly to be removed from the reactor by the normal fuel-handling equipment.

Fig. 5. Three Methods for Coupling and Decoupling Instrumented Irradiation Subassembly and Extension Tube in EBR-II

With the welded-coupling concept, after the instrument leads are cut (by the type of lead-cutter shown in the gripper-coupling sketch) a knife-action or a rotary-disc cutting tool (not shown) is inserted and severs the coupling; two 180° opposing cutters are used to balance the lateral cutting force. A number of cutting operations have been performed on a 2-in.-dia, 0.020-in.-wall tube section by both types of cutters in dry bench tests indicate that approximately 50 ft-lb torque is required to cut with the knife-edge tool but only 20 ft-lb is needed with the disc-type tool. Most of the torque is believed due to the flanging out of the tube metal; the deformed metal apparently causes less friction for the rotary cutters. Low cutting torque is desired to reduce twist in the extension tube and to limit the forces transferred to the guide tube and adjacent subassemblies.

To disconnect the threaded-type coupling, a vertical cutter first severs the instrument leads. To sever the leads, a push down on the extension rod deforms the leads outward, while the sleeve clamps them against the extension tubes; when the push overcomes the spring force, the knife edge on the bottom of the extension rod shears the leads against the anvil. With the cutting edge and anvil appropriately shaped, the leads can

Fig. 6. EBR-II Instrumented Subassembly Full-scale Mockup

be cut in succession, which minimizes the force required. After the leads are cut, a castellated tool is inserted and rotated, which causes the threaded anvil to disengage the extension tube and subassembly.

(ii) Sodium-environment Tests. In addition to the bench tests for severing leads and extension tubes, sodium-environment tests are being planned. For these, an existing apparatus has been reactivated; it will be used jointly for the FFTF and EBR-II instrumented-subassembly work.

(iii) Full-scale Mockup. To establish feasibility of fit and assembly of a complete full-size instrumented subassembly and extension tube, a full-scale reactor installation mockup has been designed and is being constructed (see Fig. 6). This mockup uses an existing 50-ft-high experimental tower. The upper part simulates the control-rod nozzle section through the small rotating plug and the operating floor; the central part simulates the penetration through the reactor vessel cover; the lower part simulates the guide tube and instrumented subassembly as situated in the core.

During the fuel-handling operations in the reactor, the instrumented subassembly will be in a raised position sufficient to clear the tops of adjacent subassemblies and will move about as the rotating plugs are positioned. Thus, the instrumented subassembly will be subjected to inertial and sodium drag loads each time the small and large plugs are repositioned. It is planned to simulate

and analyze these loadings in the mockup. Also, instrument-lead severance tests will be conducted and torque and twist measurements of the subassembly, extension tube, and the gripper attachment will be observed.

5. Experimental Irradiations

The status of all experimental irradiations in the reactor and the approximate exposure at month's end is shown below:

Subassembly	Date Loaded	Capsule Content and Number of Capsules ()	Experimenter	Approximate Accumulated Exposure (MWd)	Goal Exposure (MWd)
XG02	7/16/65	UO ₂ -20 w/o PuO ₂ (1) Stainless Dummies (18)	GE	9,744	13,600
XG03	7/16/65	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	9,744	19,450
XG04	7/16/65	UO ₂ -20 w/o PuO ₂ (2) Stainless Dummies (17)	GE	9,744	39,000
XG05	9/3/65	UO ₂ -20 w/o PuO ₂ (9) U-15 w/o Pu-10 w/o Zr (1) U-15 w/o Pu-10 w/o Ti (1) UC-20 w/o PuC (3) Structural (5)	GE ANL ANL ANL GE	9,317	10,300
XG06	9/3/65	UO ₂ -20 w/o PuO ₂ (12) U-15 w/o Pu-10 w/o Zr (1) U-15 w/o Pu-10 w/o Ti (1) Structural (5)	GE ANL ANL	9,317	20,600
XA08	12/13/65	UC-20 w/o PuC (8) Structural (11)	ANL	7,495	19,800
XO10	3/24/66	UO ₂ -20 w/o PuO ₂ (4) Structural (11) Structural (4)	GE ANL PNWL	6,675	19,600
XO11	5/9/66	UO ₂ -20 w/o PuO ₂ (7) UO ₂ -20 w/o PuO ₂ (9) 304 SS-30 w/o PuO ₂ (1) 304 SS-20 w/o PuO ₂ (1) 304 SS-20 w/o UO ₂ (1)	ANL GE PNWL PNWL PNWL	5,104	8,300
XO12	8/10/66	UO ₂ -20 w/o PuO ₂ (19)	NuMEC	2,975	20,600
XO14	7/17/66	Structural (5) Graphite (2) Structural (7) Structural (5)	PNWL PNWL GE NRL	3,674	a
XO15	11/15/66	PuO ₂ -UO ₂ (11) PuO ₂ -UO ₂ (2) (Pu U)C (4) Mk-I-A (Metal) (2)	NuMEC GE ANL ANL	1,320	11,000
XO16	1/13/67	Structural (9) Structural (10)	ANL GE	1,020	3,000
XO17	11/15/66	PuO ₂ -UO ₂ (11) (Pu U)C (3) Mk-I-A (Metal) (5)	NuMEC UNC ANL	1,320	6,500

Subassembly	Date Loaded	Capsule Content and Number of Capsules ()		Experimenter	Approximate Accumulated Exposure (MWd)	Goal Exposure (MWd)
XO18	12/6/66	Structural	(3)	GE	630	21,300
		Structural	(1)	PNWL		
		Structural	(2)	ANL		
		Structural and Heavy Metal Fission Yield Samples	(1)	ANL		
XO19	1/13/67	UO ₂ -20 PuO ₂	(7)	GE	0	7,500 ^b
		(U.8 Pu.2) C	(3)	UNC		
		Structural	(8)	PNWL		
		Graphite	(1)	PNWL		
XO20	1/13/67	UO ₂ -20 PuO ₂	(9)	GE	0	7,500 ^b
		(U.8 Pu.2) C	(3)	UNC		
		Structural	(4)	PNWL		
		Structural	(2)	ANL		
		Graphite	(1)	PNWL		

^aMaximum attainable before reaching experimental subassembly capacity.

^bAre being considered for further exposure.

Experimental capsules from UNC, PNWL, and GE were shipped to the TAN hot shop and transferred to Idaho Nuclear Corp. for interim handling and transfer to the respective experimenters. The following capsules were included in this transfer:

<u>Capsule</u>	<u>Content</u>
UNC-78	(U Pu) C
UNC-79	(U Pu) C
UNC-80	(U Pu) C
GE-L4C	Structural Materials
GE-L4D	Structural Materials
PNWL-5P13	Pu-SS Cermet
PNWL-5P14	Pu-SS Cermet
PNWL-A1	Structural Materials
PNWL-A2	Structural Materials
PNWL-A5	Structural Materials
PNWL-A6	Structural Materials

Subassembly XO09, which had been in a maximum flow position in the reactor, was selected as the second subassembly to have its filter tube inspected for possible retention of foreign material. Subassembly XO13 had been inspected in a like manner in November 1966 at the TAN hot cell at the Idaho Site and its filter was found to be clean. XO09 has been disassembled in the unwashed condition, and the lower end fixture is in the FCF Air Cell awaiting transfer to TAN as soon as space is available at that site for the inspection.

6. Surveillance

a. Examination of Irradiated EBR-II Fuel Jacketing. Tube-burst tests at 600, 700, 750, 800, and 900°C reported (see Progress Report for December 1966, ANL-7286, pp. 8-9) indicate that (n, α) reactions cause a nonrecoverable decrease in strength and loss of ductility in Type 304 stainless steel at test temperatures above 700°C, but do induce a slight increase in strength at temperatures no higher than 700°C. Helium generated by the (n, α) reaction was reported not to be a problem in EBR-II under present operating conditions.

Additional tube-rupture tests at 500°C have been made on five sections from each of two irradiated fuel jackets of Type 304L stainless steel. These two jackets were from fuel elements that had been postirradiation bonded by heating to 450°C in the course of about an hour, held at temperature approximately 5 min and furnace cooled. The sections from one jacket were given a pretest anneal for 10 hr at 500°C; the five sections from the other jacket were annealed for 100 hr at the same temperature. Rupture strength and average percent diameter change are compared in Table VII for (unannealed) irradiated sections of a third jacket (not bonded after irradiation) and for the sections (of the first two jackets) annealed for 10 and 100 hr at 500°C. The three fuel jackets had been irradiated in the same sub-assembly. The maximum fluence was 1.2×10^{22} n/cm² (fast neutrons) and the minimum was about 5×10^{21} n/cm². Irradiation temperatures varied from 371°C at the bottom to 520°C at the top of the fuel jackets.

TABLE VII. Tube-rupture Tests at 500°C on Irradiated Type 304L Fuel-jacket Sections

Section	As Irradiated		Irradiated and Annealed			
			10 hr at 500°C		100 hr at 500°C	
	Rupture Strength (psi)	% $\Delta D/D$	Rupture Strength (psi)	% $\Delta D/D$	Rupture Strength (psi)	% $\Delta D/D$
A (top)	66,600	0.2	63,400	1.2	61,600	2.9
B	76,300	1.3	67,600	0.5	69,400	0.4
C	83,400	0.4	76,800	0.2	76,800	0.4
D	88,300	0.1	84,100	0.1	79,700	0.6
E (bottom)	90,300	0.2	85,500	0.1	79,100	0.2

Annealing at 500°C for 10 hr has reduced the rupture strength of the irradiated material. The longer annealing time resulted in an additional strength reduction for three of the fuel sections. The greater strength

reductions occur toward the bottom of the jackets, where the irradiation temperatures were lower. The ductility as determined from diameter measurements does not seem to have been affected by the annealing treatments.

Some unirradiated Type 304L tube sections were annealed for 100 hr at 500°C and for one hour at 900°C prior to burst testing at room temperature, 500, 700, and 900°C. The results are compared with those from as-received material in Table VIII. There were no obvious effects of the annealing treatments on the rupture strength. There is a slight indication that the annealing treatments may have increased the ductility at the higher test temperatures.

TABLE VIII. Tube-rupture Tests on Unirradiated Type 304L Specimens

Test Temp (°C)	Rupture Strength (psi)			% $\Delta D/D$		
	As Received ^a	Annealed 100 hr at 500°C	Annealed 1 hr at 900°C	As Received ^a	Annealed 100 hr at 500°C	Annealed 1 hr at 900°C
R.T.	77,300 65,000	66,200	67,000	20.0 19.5	19.6	18.6
500	47,800 48,500	50,100	47,800	9.2 10.1	11.4	14.7
700	33,800	34,900	34,500	11.1	15.0	15.1
900	19,300 19,300	17,900	18,800	15.8 19.5	20.3	24.7

^aAnnealed by vendor.

7. EBR-II VAD Machine

One of the important check points of the new schedule for the Vertical Assembler-Disassembler (VAD) was January 20, when all major components should have been delivered to the Remote Control Engineering Division. These parts include the rails being produced by Rock Island Arsenal, upper frame being fabricated by Ideal Tool & Mfg. Company, and the base frame section being fabricated by Central Shops. As of this writing, none of the parts have been received. Consequently, the start of assembly of the machine has dropped further behind schedule than originally indicated.

All major frame components should be delivered between February 2 and February 6. To get these delivery dates, we asked the fabricators to omit the plating and phosphating called for. This will permit an earlier starting date on the assembly and operational testing. We plan to have the parts phosphated at some time when the machine has been disassembled for modification or other purposes.

The design of the welding unit has been completed and detailing has begun. The Miller ES Welder, which was sent back to the factory for re-adjustment, has been received, and preliminary tests indicate it is working satisfactorily. Further tests will be needed to verify this.

Work on installation drawings, wiring diagrams, and control-console design is continuing.

8. Fuel Cycle Facility (FCF)

a. Operations. Processing and fabrication of fuel subassemblies was continued routinely, along with equipment repair and maintenance.

(i) Crane and Manipulator Repair. The drive unit failed for electromechanical manipulator No. 6 in the Argon Cell, owing to shearing of a pin in the universal joint which connects the drive unit to the outer (driven) wheels of the bridge. Failure of the pin allowed the drive shaft to hang from the sheel truck and to fall against the outer wall of the cell. By the use of a special tool, held in the grip of an adjacent manipulator, the ends of the universal joints were remated remotely and were held in position until a new pin could be inserted. Insertion of the new pin, which was equipped with ball detents to hold it in place, was accomplished by another special tool manipulated by the carriage on manipulator No. 6. The manipulator has been restored to operation.

The Air Cell crane trolley was transferred to the cell roof enclosure for repair of the hoist unit and modification of the trolley-drive system. Repair was effected in a plastic enclosure after decontamination by thorough wiping. Some additional contamination accompanied the need to use a crane-lifting beam which had become contaminated in the Argon Cell. Maintenance and repairs included checking of the load brake system and slip clutch, adjustment of the hoist motor brake which had been slipping, checking oil level, and addition of oil for the hoist drive gear reducer and load brake system, installing a noncoasting clutch for the trolley drive, and replacing a broken bolt in the trolley-drive clutch. The crane is now operating satisfactorily.

(ii) Mark-IB Fuel-element Evaluation. The Mark-IB fuel element is proposed to utilize a modified restrainer which has a thinner (0.040-in.) shank than that presently used. In some previous tests (see Progress Report for November 1966, ANL-7279, p. 9), the lower end of the restrainer was observed to not be centered in the jacket. Consideration is being given to incorporating a cross-bar at the lower end to cause this to be centered (see Sect. II.A.4.b). Twenty such prototype elements were prepared and radiographed. Centering of the restrainer appeared to be accomplished reasonably well, but eddy-current testing for sodium level revealed that the

location of the cross-bar was not indicated reliably. Such an indication is desirable to facilitate measurements of sodium level.

(iii) Fabrication of Special Subassemblies. Three Subassemblies (XO16, XO19, and XO20) containing experimental irradiations were fabricated. One (XO16) is a Mark-B-19 type for a core position and contains structural-material capsules, while the other two contain fuel specimens, graphite, and structural materials for irradiation in inner-blanket positions.

Seven special fuel subassemblies were prepared from various types of fuel elements (some including unirradiated fuel) to aid in clarifying the causes of the anomalous swelling observed with Mark-IA fuel. The effects of fuel-pin length and enrichment will be evaluated in controlled tests. Assemblies were also fabricated entirely of consolidated, recast, recycle scrap, which generally has shown a low swelling rate, to evaluate the feasibility of a higher burnup limit for such fuel. These subassemblies will be irradiated as normal driver fuel.

Five subassemblies fabricated were the first of the half-worth type, which contain 46 normal fuel elements and 45 dummy stainless steel elements. These will be used to facilitate reactor loading reactivity adjustments.

A summary of production operations is contained in the following table:

Production Summary for January 1967

Subassemblies received:	27 (+ 4 flux wire assemblies)
Subassemblies dismantled:	11
Subassemblies fabricated:	15 (5 are half-worth)
Subassemblies transferred to reactor:	17 (5 are half-worth)
Elements decanned:	985
Melt Refining:	8 Irradiated 3 Recycle
Pour Yield (avg):	91.9 93.0
Injection-casting runs:	11
Elements processed:	
Accepted:	1080
Rejected:	67
Elements welded:	1205
Elements leak tested:	
Accepted:	1399
Rejected:	38
Elements bonded and tested (completed runs):	
Accepted:	1013
Rejected:	48
Subassemblies subjected to surveillance:	9

B. Physics Development

1. ZPR-3

Assembly 48, a plutonium-fueled critical assembly with a distribution of materials similar to a large carbide reactor, has been under study for some months, and its general characteristics and justifications have been previously discussed in this series. Uranium-foil capture measurements have been made that indicate some of the fine details of the local flux variations. Recently, measurements have been undertaken on the axial distribution of reaction rates and material worths of a number of different samples selected to meet the needs of the fast reactor program generally and to be particularly relevant to the analysis of the design of the Fast Flux Test Facility (FFTF), Pacific Northwest Laboratory (see also pages 27-33). Of the various cores proposed for ZPR-3 physics studies in the next year(s), Assembly 48 bears the nearest resemblance to the FFTF composition. Thus, it was determined that, following the axial distribution measurements, the core, with some modifications, would be suitable for the performance of a series of experiments which would aid in the evaluation of FFTF control rods. Results of the U^{238} capture distribution measurements and the axial distributions of reaction rate and worth measurements are given here, together with a brief outline of the FFTF control-rod experiments.

a. Foil Irradiations. In the heterogeneity measurements with Assembly 48, thin U^{235} , U^{238} , and manganese foils were irradiated in each configuration. The U^{238} capture data is presented here; the U^{235} and manganese data are presently being analyzed.

The relative U^{238} capture rate was measured at seven positions through the central drawer (1-P-16) at a distance of 1 in. from the core midplane. Thin, 2-mil by 0.375-in.-dia, U^{238} foils wrapped in aluminum were placed between the plates, irradiated, and the U^{238} captures determined by counting the U^{239} gamma activity using the method described by Davey.¹

The relative U^{238} capture rate is shown in Figs. 7, 8, and 9, the data in each being normalized to unity for the foil having the highest capture rate.

SODIUM
GRAPHITE
SODIUM
GRAPHITE
DEPLETED U
PU - U ALLOY
PLUTONIUM
PU - U ALLOY
DEPLETED U
GRAPHITE
SODIUM
GRAPHITE
GRAPHITE

Fig. 7. Relative U^{238} Capture through the Bunched Central Drawer of ZPR-3 Assembly 48

b. Axial Traverses of Reactivity Samples. The analysis of the data from the reactivity traverse experiments conducted during the latter part of December 1966, was completed.

¹ Davey, W. G., The Ratio of U^{238} Capture and U^{235} Fission Cross Sections in Fast Reactors, ANL-7034 (May 1965).

GRAPHITE
SODIUM
DEPLETED U.
PLUTONIUM
DEPLETED U.
GRAPHITE
SODIUM
GRAPHITE
GRAPHITE
PU-U ALLOY
PU-U ALLOY
SODIUM
GRAPHITE

Fig. 8. Relative U^{238} Capture through the Normal Central Drawer of ZPR-3 Assembly 48

DEPLETED U.
SODIUM
PU-U ALLOY
GRAPHITE
SODIUM
GRAPHITE
PLUTONIUM
GRAPHITE
SODIUM
GRAPHITE
PU-U ALLOY
GRAPHITE
DEPLETED U.

Fig. 9. Relative U^{238} Capture through the Unbunched Central Drawer of ZPR-3 Assembly 48

For these traverses, a channel was constructed axially through drawers 1- and 2-P-16, with the materials arranged in the drawers as indicated for

Fig. 10. Material Arrangements around Axial Traverse Channel in Assembly 48

1-P-16 in Fig. 10. Small annular or cylindrical samples of plutonium, enriched and depleted uranium, enriched boron, tantalum, and steel were traversed through the channel and their worths measured at a number of axial positions. Table IX gives the specifications of the samples, and Table X lists the results, expressed as inhours per kg, versus position. A number of anomalies were found in the asymmetries of the worth profiles about the midplane. It is suspected that the axial fuel-plate arrangements are responsible for these asymmetries. The continuity of the Pu-U alloy fuel columns is broken, by about 3/8 in., at the midplane, at 3 and 8 in. in Half No. 1, and at 7 and 12 in. in Half No. 2. Also, the coefficients at the core center are generally different, by up to 9%, from the results obtained previously with the samples mea-

sured in a 2 x 2 x 2-in. void at the center of the core (see Progress Report for October 1966, ANL-7267, pp. 12-21). This is probably due to fuel-proximity effects.

TABLE IX. Specifications of Reactivity Samples Used in Assembly 48 Traverses

Sample Material	Shape	Material Dimensions (in.)			Material Mass (g)	Can Mass ^a (g)	Material Composition (w/o)
		Length	OD	Thickness			
Plutonium	Annulus	1.86	0.34	0.020	9.187	6.63	95.05 Pu ²³⁹ , 4.50 Pu ²⁴⁰ , 0.45 Pu ²⁴¹
Enriched Uranium	Annulus	1.90	0.39	0.010	7.128	1.79	93.16 U ²³⁵ , 6.84 U ²³⁸
Depleted Uranium	Cylinder	2.00	0.19	-	19.63	-	99.78 U ²³⁸ , 0.21 U ²³⁵
Enriched Boron	Annulus	1.90	0.39	0.010	0.406	11.55	85.48 B ¹⁰ , 7.32 B ¹¹ , 7.2 impurity ^b
Tantalum	Annulus	1.98	0.38	0.015	9.67	1.77	>99% Ta
Stainless Steel	Cylinder	2.00	0.42	-	35.90	-	Type 304, approx 72 Fe, 18 Cr, 10 Ni

^aStainless steel can or bracket.^bThis composition is suspect pending confirmation of additional analyses.

TABLE X. Results of Axial Traverses of Reactivity Samples through 1- and 2-P-16 in Assembly 48

Position, in. from Midplane ^a	Reactivity Coefficient (lh/kg) ^b					
	Plutonium (±2)	Enriched U (±3)	Depleted U (±0.8)	Enriched B (±50)	Tantalum (±2)	304 Stainless (±0.5)
-24	1	-	+1.1	12	-	0
-21	11	10	+0.9	-39	+1	+0.4
-18	34	27	+2.0	-318	+2	+3.5
-16	90	-	+3.1	-655	-	+8.8
-15	122	93	+2.6	-1192	-15	+9.6
-14	152	-	0	-1826	-	+8.1
-12	213	-	-4.6	-3141	-	+2.9
-11	-	180	-	-	-70	-
-9	311	-	-11.0	-5046	-	-4.7
-6	398	288	-16.9	-6889	-129	-10.6
-4	448	-	-18.4	-7455	-	-14.7
-2	472	339	-19.7	-7906	-153	-16.1
-1	476	-	-21.6	-8036	-	-16.9
0	469	340	-21.8	-8063	-156	-16.5
1	474	-	-21.6	-8036	-	-16.0
2	464	330	-21.7	-7901	-155	-15.9
4	-	-	-20.1	-	-	-
6	389	281	-15.2	-6506	-124	-
9	299	-	-10.1	-4937	-	-4.3
11	-	174	-	-	-67	-
12	-	-	-3.7	-2954	-	-
14	-	-	-0.6	-	-	-
15	132	88	+2.4	-1136	-16	+9.7

^aNegative in Half No. 2; positive in Half No. 1.^bFor sample material as specified in Table IX, after accounting for mass and worth of canning, if any.

c. Measurements of Control-rod Interaction. This phase of the experiments involves measurements for the FFTF project in an Inconel reflector. Due to a lack of materials for a complete reflector loading, only two quadrants of the depleted uranium blanket used for Assembly 48 were replaced with a reflector of nickel, stainless steel, and sodium. The experiments were then carried out in the region of the reflector segments.

Axial traverses have been made in a reflector drawer adjacent to the core. Material-worth traverses were made for stainless steel, empty sample carrier, plutonium, depleted uranium, tantalum, and boron-10. Reaction-rate traverses involved plutonium and depleted-uranium fission counters and a BF_3 proportional counter.

The worth of a 54-in. poison-rod mockup with enriched B_4C was measured in several individual positions in the reflector adjacent to the core. Following this, the worths of several two-rod and three-rod combinations were measured to determine the interaction effect between poison rods. Next, the enrichment of the B_4C was reduced by replacing one-half the enriched B_4C with natural B_4C in the rod mockups and remeasuring a few of the interaction experiments. The effect of bunching of the B_4C in a rod mockup was also measured.

The worths of rhenium and europium were measured relative to an equal volume of enriched B_4C in a reflector drawer adjacent to the core by measuring the subcritical multiplication of the spontaneous fission source.

Two poison-rod mockups, 54 in. long, using enriched B_4C were then loaded into the reflector at the core edge on opposite sides of the core and the core size was increased to achieve criticality.

Analyses of the experimental results obtained with the reflector in place have not yet been completed.

d. Subcritical Measurement of k_{eff} . In order to investigate the dependence of the inverse of multiplication on k_{eff} for subcritical measurements in Assembly 48A, a series of computations were performed using the MACH 1 diffusion code. The core, with Assembly 48 core composition, was in the form of a thick spherical shell. The spherical region inside the shell was filled with material composed of homogenized empty matrix drawers and structure. The core was surrounded by 30 cm of the U^{238} blanket material used in Assembly 48. The 22-group cross sections were typical of those used for large oxide cores. This assembly was made critical by varying the thickness of the core shell.

One set of computations was then performed, each with a different concentration of B^{10} in the central region. For each of these k_{eff} was computed, and the neutron flux and source distribution resulting from the spontaneous-fission source in the core were determined. The neutron flux integrated over all 22 groups at a point in the blanket 10 cm from the core-blanket interface was used as a number proportional to multiplication. The location was chosen to be outside the core, as a neutron detector which measures multiplication is outside the core, and yet is not close enough to the edge of the blanket to be disturbed by the zero in neutron flux at that surface, which was artificially introduced as a boundary condition.

In Fig. 11 one thousand over total flux is plotted as circled points against k_{eff} . This curve is to be compared to the straight dashed line drawn through the first computed point. From the small deviation of the curve from the linear assumption it can be seen that $10^3/\phi$, or the inverse of multiplication, is proportional to $1 - k_{eff}$ within about 4% at 4% subcritical.

Fig. 11. Relation of Reciprocal Flux (or counts) to Reactivity for Subcritical Reactor

On the same plot another set of points is plotted as solid circles. These show the relation of $10^3/\phi$ to k_{eff} at the same 10-cm point in the blanket when B^{10} is added uniformly in the first 5 cm of the blanket next to the core instead of in the central region. The obvious nonlinearity is due to the depression of the flux in the blanket by the B^{10} as the k_{eff} is also reduced. Of course this corresponds to the experimental case in which the neutron-flux detector sees the core through a control rod which moves during the sequence of subcritical measurements or in which some other effect disturbs the relationship between total flux at a given point and multiplication of the core.

e. Analysis of Reflector Properties and Worth of Control Rods.

In Assembly 48A, full-reflector properties and worth of control rods in a full reflector can be observed and measured for an Inconel-sodium mixture without replacing the whole blanket by reflector.

In Fig. 12 the reflector savings for a core of Assembly 48 composition is plotted against the thickness of an Inconel-sodium reflector in spherical geometry. The reflector savings approaches its maximum value when the thickness is 30 cm. The circled point marks the reflector savings obtained when a 20-cm-thick reflector is backed by 10 cm of blanket. It is found that the savings are greater, in this case, due to the U^{238} source of fast neutrons, but it is not as great as the maximum reflector savings obtainable with an Inconel-sodium reflector. As a result, the reflector savings near the maximum amount can be obtained without replacing all of the blanket thickness by reflector.

Fig. 12. Reflector Saving vs Reflector Thickness for ZPR-3 Assembly 48
Core Composition with Inconel Reflector (spherical geometry)

Control rods in a reflector consisting of two opposing 100° sectors embedded in blanket material have a worth which is within a few percent of that for the same reflector-blanket combination extending 360° around the core. Sectors of 100° are used to provide reflector regions of large enough angular width to allow measurement of worth of three control rods in the region at the same time without unduly large edge effects. It is desirable to measure the worth of three rods at a time to evaluate accurately the amount of reduction in single-rod worth caused by rod interference.

Considering three rods as part of a complete, equally spaced ring of rods, the interaction of the two outside rods with the center one can properly be termed nearest-neighbor interaction, in that the flux depression about the center rod due to the two adjacent rods is symmetrical.

Two 100° opposing sectors were chosen in order to maintain flux symmetry about the core axis in the plane which bisects the sectors.

Table XI lists calculated reactivities and rod worths for the reflector-blanket combination shown in Fig. 13. Five-group cross-section sets for the stainless steel-reflected systems of Assembly 46 were considered adequate and used for these preliminary calculations. SNARG-2D in the XY frame of reference with all regions infinite in Z was used for the computation. A fictitious poison was added to simulate loss by diffusion of neutrons in the Z direction. Rod compositions were the enriched rods of Assembly 46. The total region of computation was either the right half-space with proper boundary conditions for twofold symmetry about the center of Row 16, or one quadrant with boundary conditions for fourfold symmetry about Row 16 and Row P.

TABLE XI. Core with an Opposing Pair of Reflector Sectors
Embedded in Blanket (see Fig. 13)

Twofold symmetry about center of Row 16	
(a) Reactivity without rods	0.995850
(b) $\Delta k/k$ for rod in L-10 (also rod in L-22 by symmetry)	-0.011613
(c) $\Delta k/k$ for rod in L-10 alone (assume no interaction across line of symmetry)	-0.005807
(d) $\Delta k/k$ for rods in L-10 and T-10 (also rods in L-22 and T-22)	-0.023387
(e) $\Delta k/k$ for rods in L-10 and T-10 (assume no interaction across line of symmetry)	-0.011694
Fourfold symmetry about Row 16 and Row P	
(f) Reactivity without rods	0.995838
(g) $\Delta k/k$ for rod in L-10 (also T-10, L-22, T-10 by symmetry)	-0.026111
(h) $\Delta k/k$ for rods in L-10, T-10	-0.013055
(i) $\Delta k/k$ for rod in P-9 alone	-0.007391
(j) $\Delta k/k$ for rods in P-9, L-10 and T-10	-0.016038

Fig. 13. Loading Diagram for ZPR-3 Assembly 48A Diagram for Sector of Reflector

The k_{eff} of the assembly without control rods is 0.995850, and the reactivity changes in Items (b) and (e) of Table XI for twofold symmetry are referred to this. By comparing (e) with (c) it may be concluded that the interaction between rods in L-10 and T-10 is less than about 1%, which is less than the accuracy of the computation itself. Since L-22 is more remote from L-10 than is T-10, the assumption of no interaction between L-10 and L-22 is essentially correct. This means that interaction of rods in P-9 and P-23 can also be neglected.

The results of computations (f) through (j) in Table XI were based on fourfold symmetry, and the reactivity changes obtained are with respect to the k_{eff} of 0.995838 in (f) for the assembly without rods. Changes in grid structure and boundary conditions alone are responsible for the differences between (a) and (f) and between (e) and (h). Computation (j) of the $\Delta k/k$ for the cluster of rods in P-9, L-10, and T-10 should be compared to the sum of (h) and (i) for the same rods without interaction. The interaction effect is found to be 0.004408 $\Delta k/k$.

Table XII lists reactivities for the same core with a 360° reflector as shown in Fig. 14. A comparison of Tables XI and XII shows how closely the worth of rods and their interactions in a 360° reflector can be duplicated in the sectorized reflector. The advantages of an Inconel reflector can be seen from the k_{eff} of 1.005419 without rods in computation (k). The $\Delta k/k$ for a rod in position P-9 in computation (l) indicates an increase in

TABLE XII. Core with 360° Reflector Backed by Blanket
(see Fig. 14)

Fourfold symmetry about Row 16 and Row P	
(k) Reactivity without rods	1.005419
(l) $\Delta k/k$ for rod in P-9 alone	-0.007743
(m) $\Delta k/k$ for rods in L-10, T-10	-0.013039
(n) $\Delta k/k$ for rods in P-9, L-10, T-10	-0.016919

Fig. 14. Loading Diagram for ZPR-3 Assembly 48A with Complete 360° Reflector Partial Blanket Thickness

the value for the sector case [computation (i)] by about 5%. Computation (m) is within 1% agreement with case (h) for rods in L-10 and T-10. Case (n) is greater than (j) by about 5%. The interaction of a rod in P-9 with rods in L-10 and T-10 is 0.003863.

In Table XIII reactivities are given for the same core with a full reflector which completely replaces the full blanket of Assembly 48. Computation (p) shows that the full reflector is less reactive than the reflector-blanket combination, as was illustrated in Fig. 12. Comparison of (p) with (r) of Table XIV shows that both of the above cases have a higher reactivity than the full blanket of Assembly 48. Case (q) gives the $\Delta k/k$ for rods in P-9, L-10 and T-10, and differs from (j) and (n) by <7 and 1%, respectively.

TABLE XIII. Core with Full Reflector
(Completely replacing blanket)

Fourfold symmetry about Row 16 and Row P	
(p) Reactivity without rods	1.004509
(q) $\Delta k/k$ for rods in P-9, L-10, T-10	-0.01713

TABLE XIV. Core with Full Blanket

Fourfold symmetry about Row 16 and Row P	
(r) Reactivity without rods	0.989362

The above calculations show that the worths of control rods in the reflector and their interactions can be determined to a good accuracy for confirming computational methods by using a sector reflector embedded in a blanket as shown in Fig. 13 rather than a 360° reflector backed by blanket as shown in Fig. 14.

f. Preparation of a Cross-section Set. Cross-section sets in 22 groups have been prepared using the MC² fundamental-mode spectrum with both the homogeneous Assembly 48 core composition and a heterogeneous subdivision of material which simulates the actual fuel-plate arrangement as closely as possible with a two-region cell. Additional 22-group sets were prepared by MC² with homogenized composition and zero buckling for U²³⁸ blanket material and nickel reflector material. The cross-section sets were produced for U²³⁵, U²³⁸, Pu²⁴⁰, Fe, Ni, Cr, Na, C, and O. These are the only isotopes presently available on the MC² library tape. Cross sections for other isotopes of interest were taken from the existing ANL-224 set. Using these cross sections, the central core worths are in the process of being computed for comparison with the Assembly 48 experimental values.

2. ZPR-6

For Assembly 5 of ZPR-6 a preliminary graphical evaluation was made of the fundamental prompt decay period, measured with the pulsed

TABLE XV. Decay Constants for
ZPR-6 Assembly 5

Subcriticality (1h)	Decay Constant (10/sec); Error = ± 0.1
362	3.81
218	2.93
74	2.42
38	2.37
0	2.2 (extrapolated from measurements)
0	2.56 (calculated)

neutron source, for various degrees of subcriticality. Table XV shows the results and compares them to a calculated value of the prompt period at delayed critical (calculated Rossi-alpha). The calculated decay constant is about 15% larger than the value extrapolated from the measured data.

3. ZPR-9

a. Normalization

Integral. The normalization integral (denominator of perturbation expression) for Assembly 12 of ZPR-9 was measured by a method using a Cf^{252} spontaneous neutron source.² The value was $(4.47 \pm 0.02) \times 10^5$, and was obtained by normalizing the quantities $\sum_i (\nu \Sigma_f)_i \phi_i = 1$ and

$\sum_i \chi_i \phi_i^+ = 1$ at the center of the core. The normalization integral for Assembly 11 of ZPR-9 was $(4.22 \pm 0.02) \times 10^5$. The measured spatial distribution of the importance of fission neutrons in Assemblies 11 and 12 are shown in Figs. 15 and 16, respectively. In both cases the calculated values are underestimated in the driver and buffer regions.

Fig. 15. Distribution of Importance of Fission Neutrons in Assembly 11 of ZPR-9

Fig. 16. Distribution of Importance of Fission Neutrons in Assembly 12 of ZPR-9

² Karam, R. A., "Measurement of the Spatial Distribution of the Importance of Fission Neutrons in a Fast Reactor," Trans. Am. Nucl. Soc. 9, 490 (Nov 1966).

b. Doppler Measurements. Preliminary Doppler measurements in ZPR-9 Assembly 12 have been completed. The results are shown in Table XVI. The principle purpose of making these measurements was to test the spectral matching of the ZPR-6 Assembly 4Z with the center of Assembly 12 with respect to the Doppler effect. When adjusted for the difference in perturbation-theory denominators and other small effects between the two assemblies, the data of Table XVI agree, within statistics, with comparable measurements in ZPR-6 Assembly 4Z. This indicates that the spectrum of Assembly 12 was sufficiently matched to Assembly 4Z to allow interpretable Doppler measurements to be made in the small zoned system in lieu of using the larger assembly.

TABLE XVI. Results of Doppler Measurements in Assembly 12

Date of Measurement	Sample No.	Nominal Dia and Length (in.)	Material	Enrichment (%)	Total U Mass (kg)	Sample Type ^a	Sample Environment ^b	Experimental Precision, σ (lh/kg-U)	Measured Reactivity Change versus Temp (451 lh • 1%) ^c			
									Temp (°K)	Data normalized to 0.000 lh/kg-U at 293°K and uncorrected for expansion.		
12/16/66	N1	1 x 12	UO ₂	Natural	1.116	FE	Regular Core	±0.004	335 -0.036	514 -0.147	678 -0.216	867 -0.294
12/20/66	N1	1 x 12	UO ₂	Natural	1.116	FE	Min Het	±0.002	520 -0.153	682 -0.231	867 -0.291 ^c	1062 -0.361
1/9/67	N1	1 x 12	UO ₂	Natural	1.116	FE	Min Het	±0.003	854 -0.306			
1/3/67	HE5	1 x 12	UO ₂	97.7	0.712	NNE	Min Het	±0.005	325 +0.032	364 +0.038	425 +0.053	479 +0.060
									664 +0.077	847 +0.077		

^aFE, freely expandable sample; NNE, sample in which both the axial and radial expansion is constrained.

^bRegular Core: the vertical-plane structure is unmodified in the vicinity of the sample, the central drawer is merely removed to provide access; Min Het, the eight drawers surrounding the center (sample) drawer were changed to an arrangement with a minimum heterogeneity, 6 in. into each half was reloaded with thin plates oriented normal to the sample axis.

^c $\sigma = \pm 0.011$ lh/kg uranium for this particular measurement.

Assembly 13 has been installed in the machine. This assembly is similar to Assembly 12 in that it is a zoned reactor and has identical composition except for the central zone where approximately 9 v/o polyethylene has been substituted for depleted uranium. Assembly 13 is the first of four assemblies specifically designed to investigate the systematics of the Doppler effect of the fissile material U²³⁵. Previous Doppler measurements made with Assemblies 11 and 12 on the ZPR-9 and machine Assembly 4Z on the ZPR-6 machine have shown large discrepancies between measurement and theory for this material. Central worth measurements have been completed on Assembly 13 and the results are being analyzed. Doppler measurements are currently being made.

4. ZPPR

The reactor cell of the ZPPR facility, which is on the critical path for construction, is progressing slowly. The excavation of the reactor pit is complete. The six posts of the reactor pit area have been steel capped, and the concrete top of these posts have been poured. Major excavation of the facility is completed with minor excavation required for the cell grade beam, instrument seal pits, blanket storage and the cell corridors. The

AFSR reactor pit and room foundations have been poured. Ninety-five percent of the structural steel for the support wing has been placed. The pre-cast concrete floor of the support wing (main floor) has been placed. All walls and support columns of the vault, workroom, and inside equipment room have been poured.

The present status of the reactor components is as follows:

- a. Items Completed: These items have been inspected, accepted, and are presently being stored at the Idaho Division: Bed and tables, fueled control and safety rod drives, matrix tubes, louvers for the reactor cooling system, tube plug clamping assemblies, reactor frame, pit bumper, ramp assembly, platform support bracket, seismic switch and associated electronics, beta-gamma air particulate monitors, area gamma monitors, nuclear instrumentation, reactor control consoles and cabinets, drawer springs, pit conveyor, loading platform, plugs for the rod drive mounting plates, and the reactor wire way conduits.
- b. Matrix Drawers: All rear drawers have been received and accepted. The vendor requires 280 front drawers to complete the order. Stainless steel strips have been perforated for the remaining drawers.
- c. Poison Safety Rod Drives: The first rod is partially assembled but has not been checked out.
- d. Reactor Knees: The knees have been finished and inspected. The vendor is now crating the knees for shipment.
- e. Personnel Shields: The personnel shield order was placed with McGee-Hogan Company of Salt Lake City on January 19.
- f. Matrix Alignment: The order for the alignment plates was placed on January 30 with the Challenge Machine Company. The bid packages on the supporting frame for the alignment plates are being prepared.
- g. Source Drives: Work is continuing.
- h. Rod Drive Mounting Plates: The second mounting plate has been repoured without core holes and has been sent to the machine shop for rough machining. The first mounting plate is being repaired.
- i. Matrix Tie-down: The matrix tie-down has been completed and is being painted prior to shipping.
- j. Fueled Control and Safety Rod Drawers and Ball Joint Assemblies: The drawers and ball joint assemblies have been received and have been rejected due to dimensional tolerances.

k. Poison Safety Blades and Sheaths: The samples received from Scottsonizing were satisfactory. (Scottsonizing is a surface hardening treatment.) The remaining blades and sheaths have been sent for Scottsonizing.

l. Boron Fabrication: The bid packages have been sent out and the final bids are due back on February 15.

m. Stack Gamma Monitor: The stack gamma monitor is 90 percent complete.

n. Reactor Control Cable Stand: The order for the reactor control cable stand has been placed with Rupert Iron Works, Rupert, Idaho.

C. Other Fast Reactor Physics

1. Burnup Measurements for Fast Reactors

The program for the development of analytical methods for the determination of burnup of fast reactor fuel and for the measurement of fast-fission yields is continuing. Analytical methods have been established for the determination of a number of fission products in irradiated fuel; two of these methods, for technetium-99 and lanthanum-139, are being used to monitor burnup in EBR-II; other methods, for total molybdenum, total ruthenium, total cerium, and praseodymium-141, are applicable to other reactor fuels. Each method has been evaluated using irradiated uranium; the results indicate that accuracies of about $\pm 1\%$ are achievable. Comparably accurate burnup measurements are, however, not presently obtainable since the existing fast-fission yield data are not only sparse but are also of questionable accuracy.

To establish fast-fission yields with the desired accuracy, pure materials which have been irradiated to a relatively high burnup (20 to 30%) are required. At this level of burnup the number of fissions can be determined accurately from pre- and postirradiation heavy-element mass-spectrometric isotope-dilution analyses. An irradiation in EBR-II, which is described below, is designed to provide such materials.

The irradiation is being performed in capsules which contain ~100-mg amounts each of uranium-233, -235, and -238, and of plutonium-239, -240, and -241. The length of the irradiation will be sufficient to achieve burnups of about 25% for U^{233} , U^{235} , U^{238} , and Pu^{241} , and burnups of about 5% for U^{238} and Pu^{240} . When the combined errors arising from determinations of the number of fissions and determinations of fission product concentration are considered, it is estimated that fission-yield values will be established with accuracies of $\pm 1\%$ for U^{233} , U^{235} , and Pu^{239} and of $\pm 5\%$ for U^{238} , Pu^{240} , and Pu^{241} .

To determine the effect, if any, of neutron energy on the fission yields, some capsules are being irradiated in the core and others in the blanket of EBR-II.

Because the heavy-element isotopic composition of nuclear fuels is of great importance to reactor design, operation, and fuel processing, a study of the transformations which have occurred as the result of such reactions as n, γ and $n, 2n$ will also be made. This study will be made on each of the uranium and plutonium isotopes as well as samples of thorium-232 and neptunium-237 which have been included in the irradiation.

This irradiation began in December 1966 and will terminate some time in 1969 or 1970.

D. Component Development

1. Sodium Technology and Development

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

A Statham thin-film strain-gauge pressure transducer having a range of 0-100 psia has been tested successfully up to 500°F. Accuracy over this temperature range is about 5% of full scale. Most of the error is introduced by a reproducible zero shift. At 600°F, a drift in the zero shift was noted, but sensitivity remains stable. The cause of the zero-shift drift remains to be determined; a fine-wire trimmer resistor in one of the bridge arms may be the cause and is being investigated.

As an entirely different approach, pressure transducers that operate on the Booth-Cromer transducer principle (null-point pressure indicator), with modifications, are being obtained from MSA Research Corp. When the completed transducers are received, they will be adapted for the expected FFTF conditions, tested in sodium, and the length of lead and insulator problems will be investigated.

During the next reporting period, backup techniques will be considered.

b. Method for Locating Failed Fuel. To find a method for determining which subassembly within a shutdown reactor core has one or more failed fuel pins, the feasibility of sensing the fission-gas release from the subassembly will be studied. To prove the principle, a scheme will be

demonstrated for efficiently disengaging noble gases from a continuously circulating liquid-sodium system at 1200°F. The equipment will include a gas-disengagement device, a sodium vapor trap, a viewing device, a radiation-monitoring device, and a gas-injection device.

Work during this reporting period has been aimed at determining that fission-gas separation from sodium, followed by detection or measurement, is a reasonable technical approach toward establishment of this method as contrasted to other methods of detection of failed fuel assemblies. The limited literature review thus far indicates that certain fission gases remain separate or can be separated from sodium.

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

An existing test tank that was used for previous connector work was disassembled, and the installation was redesigned to make it more efficient and reliable. A cold trap, more extensive instrumentation, better heaters, improved insulation, increased pipe sizes, etc., have been provided and worn out components are being repaired or replaced. Shop work is almost completed. New heaters and other materials are on order. Candidate gasket designs will be tested.

d. In-core Flowmeter. So that sodium flow through individual sub-assemblies or process tubes can be measured, technology will be established to assure that small magnetic, turbine, or other types of flowmeters can be obtained for use in the severe temperature and radiation environment. Out-of-pile and in-pile test programs will be established to assure satisfactory flowmeter performance. A prototype flowmeter of one or more candidate types will be fabricated and tested to demonstrate reliable performance.

A preliminary status report has been started; it will include a brief analysis and description of an eddy-current-type flowmeter, which looks very attractive and could circumvent some of the signal-generation problems if hard magnetic materials are found eventually to be completely unacceptable for in-core use in the FFTF.

The thermal testing of Alnico-V magnetic material is continuing. Typical cylindrical flowmeter magnets have been heated from room temperature to approximately 1000°F at a rate of 8°F/sec, then to 1300°F at a slower rate; the magnet is kept at 1300°F for approximately 30 min and cooled from

1300 to 900°F at a rate of 8°F/sec, from 900 to 500°F at a rate of 2°F/sec, and then more slowly to room temperature. (Heating and cooling rates were determined by a thermocouple attached directly on the magnet surface; magnetic flux density is measured at room temperature before and after each thermal cycle, at which time the material is also examined for evidence of cracking.) Eight different samples were cycled in this manner with very reproducible results: (1) the room-temperature flux density decreased to approximately 0.2 of its initial value and could not be restored by remagnetization; (2) there is no evidence of any material cracking; (3) further magnet exposure to at least three such cycles does not change the residual flux density at room temperature. This type of heat cycling appears to produce a "stabilized" magnet. When a "stabilized" magnet was further subjected to cooling rates of 4°F/sec and 7°F/min, the flux density was not affected; therefore, it is concluded that complete short-term stabilization has been achieved at room temperature. The reversible effects at other temperatures, as well as long-term stability, have yet to be determined.

Three additional Alnico-V magnet samples are being prepared for long-term temperature-stability tests. The magnets will be stabilized and run at 1200°F until an observable change (2%) in flux density is noted or until 2000 hr of operation have been achieved. Sample magnets of Alnico-VI and -VIII are being procured, but they will not be available in time for test with the three Alnico-V samples.

A study of techniques to measure magnetic properties continuously at high temperature has been initiated. These techniques will take into consideration the requirements imposed by postirradiation hot-cell examination of the magnets to be used during the in-pile testing.

Delivery of acceptable ceramic electrical insulators for fabricating a preliminary prototype magnetic flowmeter has been delayed.

The usefulness of EBR-II as a magnet irradiation facility is being studied. The neutron-induced gamma activities of the Alnico-V, -VI, and -VIII sample magnets are being calculated. Methods for running the irradiation tests at 1200°F are also being investigated, as are means for examining the irradiated magnets. The determination of the range of typical magnet length-to-diameter ratios for flowmeter applications has been initiated.

Because preliminary use of the sodium-flowmeter calibration loop at up to 900°F appears to be adequate for the initial testing of the prototype flowmeter, facility modifications will not begin until after all necessary loop drawings and materials are available.

e. Fuel-pin Thermocouple. Thermocouples are being developed to indicate fuel-pin temperatures. For this development, previous experience with fuel-pin thermocouples will be reviewed and evaluated, technology will be established to assure satisfactory thermocouple performance and design, thermocouple assemblies and materials will be extensively tested and evaluated for use in reactor fuel pins up to 2800°C, in-pile and out-of-pile tests will be conducted, and thermocouple specifications and fabrication techniques and instruction will be formulated so that competitive commercial bids can be obtained for the sensors developed.

The material and process specifications and quality-control procedures and standards developed and used to date in previous related instrument work have been compiled as a reference by which to develop new specifications and revise and upgrade the existing ones. The preliminary status report has been started. The compatibility of tantalum with UO_2 fuel up to 2800°C is being investigated. A preliminary test shows that tantalum is completely compatible with UO_2 up to 2400°C in a vacuum of about 1×10^{-3} Torr for at least 4 hr.

f. Components and Materials Evaluation Loop. Wiring of the various circuits into the Components and Materials Evaluation Loop (CAMEL) control panel was completed. Final connection to the loop heaters is in progress.

A circuit is being prepared to monitor the voltage drop across the pump tube. Any excessive voltage drop will trigger a relay which will disconnect the rectifier supplying the pump. This precaution is being added to prevent an inadvertent pump-tube burnout.

The modification circuit for the X-Y plotter and data-acquisition linkage was completed and installed. It is now possible to plot data manually upon the plotter with simultaneous punching of a data tape on the data-acquisition system. This feature extends the usefulness of the system because data now can be extracted from tapes holding many channels of information and transferred to a separate tape.

2. Liquid Sodium Coolant Chemistry

a. Dispersion of Carbon in Liquid Sodium. A potentially serious problem associated with the use of liquid sodium in nuclear reactors is the tendency of metals to carburize or decarburize. However, the mechanism by which liquid sodium acts as a medium for carbon transport is not understood. Experience has shown that the bulk of the carbon impurity in sodium is elemental and particulate. To enable study of the behavior of such carbon, attempts are being made to prepare dispersions of elemental carbon in liquid sodium. Because previous attempts using a chemical method (decomposition of Na_2CO_3 in liquid sodium; see Progress Report for November 1966, ANL-7279, p. 20) were unsuccessful, a physical method, namely, high-speed blending of carbon black in liquid sodium was tried.

Finely divided carbon black, of 90-Å particle size, was added incrementally to reactor-grade sodium contained in a stainless steel (SS) Waring Blendor. A total of ~85 ppm carbon was added. After each addition, the mixture was blended at 14,000 rpm at ~150°C for 30 min, and then sampled both with a filter (5 μ SS frit) and without a filter. Analysis of the samples for total carbon showed that dispersion of the carbon had not been achieved. The mixture was "aged" without stirring at ~120°C and sampled after 5 and 14 days, but no evidence for dispersion of the carbon was obtained.

It was concluded that these negative results might have been caused by failure of the sodium to wet the carbon black below 150°C. Since the effect of higher temperatures could not be tested in the Waring Blendor, the following was done. A portion of the mixture described above was poured into a heated stainless steel crucible, the mixture was heated to 570°C, and additional carbon black was added incrementally up to a concentration of ~1450 ppm carbon. After 9 hr of stirring, filtered and unfiltered samples of the mixture were withdrawn. Analysis of these samples indicated that the carbon black had not been suspended in the liquid sodium. However, decantation of the sodium revealed a lumpy, carbon-sodium paste at the bottom of the crucible. This observation strongly suggests that the carbon particles, individually or as agglomerates, were wet by the liquid sodium before settling. Future attempts to disperse carbon in liquid sodium will employ the preparation of such pastes as a first step.

3. Materials Evaluation

The following is a summary of work accomplished in recent months.

a. Physicochemical Mechanics of Metals. Chemical tests and electrical-property measurements were used to confirm and help explain metallographic and tensile-property investigations of how high-temperature sodium affects Type 304 stainless steel. Strength changes seem to correlate with the severe penetration of sodium into Type 304 stainless steel. It is believed that the sodium does not penetrate into the grain boundaries as an aggregate, but as some complex alloy that does not necessarily markedly affect the physical properties of the metal but that can, under some stress and temperature conditions, affect the mechanism of failure or crack propagation. Thus, for the structural design of fast reactors it could be very important that the effects of sodium, mechanical stress, temperature stress, and stress-cycle frequency be related to the way in which sodium penetration into the subsurface grain boundaries affects the mechanisms of crack propagation, fatigue, and failure in alloy steels.

Sodium penetration tests were conducted by exposing 304 stainless steel tube and rod samples to 1200°F for relatively brief times, cleaning the sample surface, etching it chemically, and observing the effects

either metallographically or chemically. To isolate the effects of high temperature from those of sodium, control samples were exposed to vacuum at the same 1200°F temperature. The various tests on both tube and rods show chemically and metallographically that sodium penetrates 0.006-0.010 in. into the grain boundaries within only 7 days. When the 0.018-in.-wall tube was etched, the sodium penetration led to its disintegration. There were inconclusive chemical indications that sodium penetrated as deep as 0.036 in. into a 5/8-in.-dia rod within 35.7 days. In either case, etchant reacts more strongly with samples that had been exposed to 1200°F sodium than with samples pretreated in a 1200°F vacuum. The longer the 304 stainless steel is exposed to 1200°F sodium, the more reactive the surface is with aluminum-sulphate etchant.

(i) Sodium Penetration into 304 Stainless Steel Tubing. Figure 17, which shows the depth to which aluminum-sulphate etchant penetrated the grain boundaries on the inside and outside surfaces of thin-wall tubing, illustrates that pretreatment with 1200°F sodium permitted drastically greater etchant penetration than did vacuum pretreatment at the same temperature. The sodium-pretreated sample completely disintegrated within the first 24 hr

Fig. 17. Depth of Penetration of Aluminum Sulphate Etchant into Grain Boundaries of Samples of Type 304 Stainless Steel Tubing (0.018-in. wall) That Had Been Pretreated in Several Ways

in the etchant. The closeness of the published-value³ curve to that for our unpretreated sample lends confidence in the test technique. Penetration was measured on photomicrographs.

(ii) Sodium Penetration into 304 Stainless Steel Rod. After samples of a 5/8-in.-dia rod were pretreated at 1200°F in sodium or a vacuum for 35.7 days, they and as-received (unpretreated) samples were etched with aluminum sulphate of pH 2-3. Etchant reaction with the rod surface indicated that the sodium had penetrated the grain boundaries. As surface layers of the 304 stainless steel were removed in steps down to 0.010-in. depth, the newly exposed surface reacted much more slowly with the aluminum sulphate. To save time, a dilute (3%) solution of sulphuric acid was used as the etchant on the surface as cutting proceeded in depths from 0.016 to 0.046 in. By means of color changes when the H₂SO₄ solution was precipitated with NH₄OH after each etching step, it was deduced that the sodium had penetrated 0.036 in. into the metal grain boundaries. At this depth, the surface of the sodium-exposed sample reacted chemically about the same as that of the vacuum-pretreated specimen at the same depth. Nevertheless, because it is difficult to separate and identify positively the temperature and sodium effects in the depth range from 0.016 to 0.036 in., the penetration of sodium to these depths is still open to question.

(iii) Sodium Effects on Tubing Resistivity and Strength. Tubing samples (0.018-in. wall) were fabricated into substandard-size tensile specimens. After these 304 stainless steel samples were exposed for 7 days to 1200°F vacuum or sodium at 900-1400°F, tensile properties and resistivity were measured in air in a furnace at the same temperatures; Fig. 18 shows the results. For comparison, room-temperature data are given for as-received samples that were not exposed to vacuum or sodium. Each point in Fig. 18 is the average of three resistivity measurements or two tensile tests.

Compared to the as-received samples at room temperature, the tensile properties at 1000°F of the sodium-exposed samples show 5-10% reduction in ultimate strength, 45% lower yield strength, and about 20% reduction in percent elongation. These results agree generally with other tensile data.⁴ The interesting feature is that the resistivity changes relate well with the changes of tensile property. Thus resistivity measurements offer promise as a uniquely convenient tool for determining the effect of high-temperature sodium on the grain-boundary constituents of structural metals. Such measurements might also be useful for evaluating the effects of irradiation on the integrity of alloys for sodium systems.

³Polar, J. P. (Climax Molybdenum Co.), A Guide to Corrosion Resistance, Caxton Printers, Ltd., Caldwell, Idaho (1961).

⁴Horton, K. E., et al., The Effects of Sodium on the Metallurgical and Mechanical Properties of Candidate LMFBF Alloys, presented at the International Atomic Energy Agency Symposium on Alkali Metal Coolants--Corrosion Studies and System Operation Experience, Vienna, Nov. 20-Dec. 2, 1966.

Fig. 18. Effects of Sodium on Changes in Resistivity and Mechanical Properties of Type 304 Stainless Steel Tubing

(iv) Additional Chemical Tests. The dimethylglyoxime test for nickel gives a stronger indication of nickel when 304 stainless steel is sensitized at 1200°F in the presence of sodium than in vacuum. Apparently, the chromium diffuses into the grain boundaries, forming a complex chromium carbide; the adjacent areas become depleted of chromium (and thus relatively richer in nickel) and therefore more susceptible to attack by the acid.

b. Bearing Materials in Sodium. Test data on sliding frictional coefficients in high-temperature sodium were obtained. The test samples consisted of a pair of Stellite-19B wear blocks and a Carboloy-98B shaft. Data were collected by a data-acquisition system, and were reduced and stored on tape by means of an X-Y plotter and a computer. The test was conducted with a modified Falex machine operating in a sodium pot. Shaft rotation was 20 rpm, total load on the jaws was fixed at 75 lb, and thermocouple output and motor current were measured at intervals of ascending and descending temperatures in the range from 400 to 1070°F.

The results of this test, when compared with the frictional-coefficient data from the inert-gas test, reveal significantly different behavior in sodium. The comparative data are shown in Table XVII.

TABLE XVII. Comparative Coefficients of Friction in Inert Gas and Sodium for Stellite-19B Wear Blocks and Carboloy-98B Shaft

Temperature (°F)	Average Coefficient of Friction	
	Inert Gas	Sodium
77	0.950	-
445	-	0.172
572	0.860	0.440
842	0.552	0.500
1022	0.312	0.602

The conclusions from a single test can only be tentative, but a number of interesting observations were made. The values for the friction coefficient are average values over a range since the motor torque oscillated over a range during the test. These oscillations are consistent with the adhesion theory of friction. The period for each oscillation was estimated to be one minute, i.e., one cycle for each 20 revolutions of the shaft.

The data in Table XVII show that at the lower temperatures the coefficient of friction was lower with the materials in sodium than in the inert gas. This signifies that the sodium behaved as a lubricant. However, the coefficient in sodium increased greatly when temperature was raised from 445 to 572°F, which is in the temperature range for wetting by sodium. Thus the greater friction presumably results from delubrication of the metal surface; the subsequent rise could possibly result from oxide reduction on the surface by the sodium. The opposite effect, oxide formation, could have caused the decrease in coefficient for the inert-gas case.

These conclusions are tentative and further investigations are required before definite conclusions can be drawn.

E. Fuel Development

1. Metal Fuels

a. Irradiation of Uranium-Plutonium Alloys. Sixteen fuel elements with U-Pu-Zr pins (U-15 w/o Pu-9 w/o Zr or U-15 w/o Pu-12 w/o Zr) severally jacketed in V-20 w/o Ti, Types 304 and 316 stainless steel, and Hastelloys-X and -X-280 were removed from Subassembly XA07 of EBR-II for transfer to Argonne National Laboratory, Illinois site (see Progress Report for December 1966, ANL-7286, pp. 22-24). They were operated at a maximum jacket temperature of 630°C and had attained estimated burnups

TABLE XVIII. Irradiation Conditions and Preliminary Postirradiation Results for U-Pu-Zr Alloy Fuel Elements Irradiated in EBR-II in Iron and Nickel-base Alloy Cladding

Specimen Number	Fuel Composition (w/o)	Cladding Alloy	Effective Density (%)	Cladding OD (in.)	Cladding Thickness (in.)	Max ^a kW/ft	Max Fuel Temp (°C)	Max Cladding Temp (°C)	% ΔL	Burnup (a/o)
ND28	U-15 Pu-9 Zr	304 SS	73.1	0.205	0.019	10.5	745	630	0.4	5.1
ND41	U-15 Pu-9 Zr	304 SS	73.8	0.205	0.019	10.5	735	625	0.9	4.9
ND32	U-15 Pu-9 Zr	316 SS	73.8	0.196	0.015	10.5	715	605	1.9	4.7
ND43	U-15 Pu-9 Zr	Hastelloy-X	74.5	0.196	0.015	10.5	725	615	-0.7	4.9
ND25	U-15 Pu-12Zr	304 SS	73.5	0.205	0.019	10.0	710	600	4.0	4.4
ND27	U-15 Pu-12Zr	304 SS	73.1	0.205	0.019	10.0	715	605	2.8	4.5
ND26	U-15 Pu-12Zr	316 SS	73.9	0.196	0.015	9.5	695	590	5.4	4.3
ND29	U-15 Pu-12Zr	316 SS	72.2	0.196	0.015	9.5	700	595	6.0	4.4
ND30	U-15 Pu-12Zr	316 SS	72.3	0.196	0.015	10.0	700	615	1.1	4.9
ND31	U-15 Pu-12Zr	316 SS	74.1	0.196	0.015	10.0	720	610	-	4.7
ND33	U-15 Pu-12Zr	Hastelloy-X	72.8	0.196	0.015	10.0	715	605	-	4.6
ND34	U-15 Pu-12Zr	Hastelloy-X	72.6	0.196	0.015	10.0	720	610	0	4.7
ND35	U-15 Pu-12Zr	Hastelloy-X	74.9	0.197	0.015	9.5	725	615	-1.5	4.9
ND37	U-15 Pu-12Zr	Hastelloy-X-280	63.8	0.207	0.015	9.5	720	610	2.3	4.9
ND39	U-15 Pu-12Zr	Hastelloy-X-280	65.4	0.207	0.015	9.5	720	610	2.3	4.8
ND44	U-15 Pu-12Zr	Hastelloy-X-280	66.4	0.208	0.015	9.5	710	600	1.5	4.7

TABLE XIX. Status of Metal Fuel Irradiations in Progress in EBR-II--Fast Reactor Development

Capsule or S/A No.	Specimen Number	Design Parameters					Operating Conditions			
		Fuel Composition (w/o)	Effective Density (%)	Cladding Composition (w/o)	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
XG06	ND23	U-15 Pu-10 Zr	63.9	V-20 Ti	0.209	0.016	10.0	540	5.2	12.5
XG05	ND24	U-15 Pu-10 Zr	63.6	V-20 Ti	0.208	0.015	10.0	535	4.9	11.8
XG05 ^a	NC17	U-15 Pu-10 Ti	66.7	V-20 Ti	0.204	0.016	10.0	540	5.1	11.3
XG06	NC23	U-15 Pu-10 Ti	63.3	V-20 Ti	0.209	0.016	10.0	540	5.3	11.7

^aBased on effective density.

2. Oxide and Carbide Fuels

Correction: In the Progress Report for December 1966, ANL-7286, p. 26, Table XII, the next to last column (X-ray Analysis: Major) should read "MC" instead of "none" in the last five rows.

a. Irradiations of Ceramic and Cermet Fuels. Five fuel elements at burnup levels between about 2 and 3 a/o were reported to have been removed from EBR-II (Subassembly XO09) for examination (see Progress Report for December 1966, ANL-7286, pp. 24-25). Two were of UO₂-20 w/o PuO₂ jacketed in Type 304 stainless steel, one in Hastelloy-X, and one in V-20 w/o Ti. Four elements fueled with UC-20 w/o PuC were, at the end of December, about ready for discharge at the 5 a/o burnup level. Two jackets were of Nb-1 w/o Zr and one each of stainless steel and Hastelloy-X. Similar mixed-carbide elements, intended for discharge at the 6 and 10 a/o levels were then, and still are, under continuing irradiation; and nine mixed-oxide elements intended for 5 a/o burnup are likewise under continuing irradiation. The continuing irradiations in EBR-II of mixed carbides are shown in Table XX, of mixed oxides in Table XXI, and of cermet fuels in Table XXII.

in the range 4.7 to 5.1 a/o. All 16 elements have been shipped, and 14 of them are being examined in the Metallurgy Alpha-Gamma Hot Cell.

Neutron radiography of the 16 irradiated capsules indicated that all fuel elements were intact. Fourteen of the capsules were then punctured, and no fission gas was detected. This result indicated that the jackets of the elements were intact. The fuel elements were then removed from their capsules. Dimensional measurements showed no significant changes in diameter as a result of irradiation. Lengths of the fuel pins were determined by neutron radiography and disclosed an average fuel length increase of 1.9%. The fuel pins were not seated on the bottom closures of the jackets, but were elevated an average of 0.20 in. These elements had been intentionally assembled without axial fuel restrainers to observe possible axial fuel movement.

The as-received capsules, which are 40 in. in length, were noticeably bowed. The average bow was 0.21 in. and could not be correlated to reactor position and amount of bow. It is speculated that relaxation of residual stresses in the capsule tubing at reactor operating temperatures may have caused the deformation. The fuel elements inside the capsules were also bowed an average of 0.12 in.

The neutron radiographs of the elements revealed one to two separations in most of the fuel pins within the cladding, varying from approximately 0.020 to 0.060 in. in width. The separations were also detected in the gross gamma scans of the fueled portion of the elements.

Diameter measurements of the elements showed no significant changes. Preliminary fission-gas recovery data indicates that approximately 40% of the fission gas had been released to the plenum. A calculation of the plenum pressures on two specimens reveals that at reactor operating temperatures the average gas pressure was 423 psi.

Transverse as well as longitudinal metallographic sections of the fuel elements failed to show penetration of the fuel into the cladding materials. The sections also showed that the original void volume that had existed as a sodium-filled annulus around the solid metal fuel pin at the start of the irradiation became redistributed throughout the fuel volume in the form of finely divided porosity. Table XVIII shows the irradiation conditions and preliminary postirradiation results for these elements. Capsules ND31 and ND33 will be returned to the EBR-II reactor for continuation of irradiation.

Four fuel elements continuing under irradiation in EBR-II include two each of U-Pu-Zr and U-Pu-Ti alloys in V-20 w/o Ti tubing. These rods are being irradiated at maximum jacket temperatures of 540°C and have reached a calculated 5.2 a/o burnup. Additional details on the design of the fuel rods and the irradiation conditions are summarized in Table XIX.

TABLE XX. Status of Irradiations of UC-20 w/o PuC Fuel in EBR-II

Capsule or S/A No.	Specimen Number	Design Parameters				Operating Conditions			
		Effective Density (%)	Cladding Composition	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
								a/o (U + Pu)	fiss/cc x 10 ^{-20(a)}
XG05	SMV-2	83.8	304 SS	0.296	0.021	21.6	645	4.9	13.6
XG05	HMV-5	80.0	Hastelloy-X	0.295	0.015	24.5	670	5.1	13.4
XG05	NMV-11	82.9	Nb-1 w/o Zr	0.284	0.012	22.3	645	5.1	14.1
XO08	NMP-2	81.6	Nb-1 w/o Zr	0.284	0.013	18.6	545	3.1	8.2
XO08	NMV-4	80.0	Nb-1 w/o Zr	0.283	0.013	27.9	635	4.4	11.7
XO08	NMV-7	80.0	Nb-1 w/o Zr	0.283	0.013	24.7	605	4.0	10.6
XO08	NMV-12	85.5	Nb-1 w/o Zr	0.284	0.014	27.7	635	4.4	12.5
XO08	HMV-1	80.0	Hastelloy-X	0.295	0.015	25.3	640	4.0	10.4
XO08	HMV-4	80.0	Hastelloy-X	0.295	0.016	27.9	670	4.3	11.5
XO08	HWMP-1	81.4	Hastelloy-X + W	0.296	0.020	17.9	555	3.0	8.1
XO08	HWMP-1	82.9	Hastelloy-X + W	0.295	0.018	27.4	685	4.4	12.0
XO15	NMP-1	81.6	Nb-1 w/o Zr	0.284	0.013	17.6	535	0.58	1.6
XO15	NMV-3	80.0	Nb-1 w/o Zr	0.284	0.013	27.1	625	0.86	2.3
XO15	HMV-2	80.0	Hastelloy-X	0.295	0.015	18.5	670	0.94	2.5
XO15	TVMV-1	83.7	V-20 w/o Ti	0.298	0.021	23.8	620	0.76	2.1

^aBased on effective density.TABLE XXI. Status of Irradiations of UO₂-20 w/o PuO₂ Fuel in EBR-II

Capsule or S/A No.	Specimen Number	Design Parameters				Operating Conditions			
		Effective Density (%)	Cladding Composition	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
								a/o (U + Pu)	fiss/cc x 10 ^{-20a}
XO11	SOV-3	83.1	304 SS	0.296	0.021	24.2	610	3.2	6.7
XO11	SOV-7	85.1	304 SS	0.296	0.021	26.1	630	3.1	6.6
XO11	SOV-1	79.9	304 SS	0.296	0.021	22.2	590	3.1	6.2
XO11	HOV-10	84.8	Hastelloy-X	0.295	0.016	26.4	615	3.1	6.4
XO11	HOV-15	79.8	Hastelloy-X	0.295	0.016	23.5	590	3.1	6.0
XO11	HOV-4	80.0	Hastelloy-X	0.295	0.016	25.1	600	3.2	6.3
XO11	TVOV-1	76.8	V-20w/oTi	0.298	0.022	23.2	575	3.1	5.9

^aBased on effective density.

TABLE XXII. Status of Type 304 Stainless Steel-clad Cermet Fuel Irradiations in Capsule XO11 EBR-II

Specimen Number	Design Parameters				Operating Conditions			
	Fuel Composition (w/o)	Effective Density (%)	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup to Date	
							a/o (U + Pu)	fiss/cc x 10 ^{-20a}
5P-9	SS-40PuO ₂	98.0	0.301	0.015	12.0	495	4.4	3.3
5P-12	SS-27PuO ₂	98.7	0.294	0.015	7.9	450	4.4	2.2
5U-14	SS-27UO ₂	97.9	0.298	0.013	6.2	435	3.3	1.7

^aBased on effective density.

Elements received for examination were mixed oxides SOV-5 and SOV-6, and mixed carbides SMV-1, SMP-1 and VMV-1. A summary description of these five elements and their irradiation is shown in Table XXIII.

TABLE XXIII. ANL Ceramic-fuel Irradiations Removed from EBR-II Experimental Subassembly X009 for Examination

Specimen Number	Design Parameters					Operating Conditions			
	Fuel Composition (w/o)	Effective Density (%)	Cladding Composition	Cladding OD (in.)	Cladding Thickness (in.)	Max kW/ft	Max Cladding Temp (°C)	Burnup at Time of Discharge	
								a/o (U + Pu)	fiss/cc x 10 ^{-20a}
SOV-5	UO ₂ -20 PuO ₂	83.4	304 SS	0.296	0.021	18.2	555	2.7	5.6
SOV-6	UO ₂ -20 PuO ₂	83.3	304 SS	0.296	0.021	19.1	565	2.9	6.0
SMV-1	UC-20 PuC	80.0	316 SS	0.306	0.025	24.2	570	2.7	7.2
SMP-1	UC-20 PuC	81.4	316 SS	0.306	0.024	18.2	640	2.1	5.6
VMV-1	UC-20 PuC	85.9	Vanadium	0.301	0.024	21.6	640	3.1	8.9

^aBased on effective density.

Neutron radiography of the five capsules and removal of the elements from the capsules were completed during the month. Neutron radiographs disclosed the elements to be intact and no indication of jacket failure. Subsequent monitoring of the capsule-plenum gases confirmed the integrity of the element jackets.

Comparison of fuel-column dimensions on radiographs before and after irradiation disclosed a small axial growth, about 1%, in the mixed oxides, but no indication of axial change in the mixed carbides. Axial center voids were apparent in both mixed-oxide elements. The voids were approximately 0.0625 in. in diameter, extending uniformly to about 0.0625 in. from the top and 0.25 in. from the bottom of each fuel column. Appearance of these voids was expected as a result of densification of material at the hot center of the fuel.

Examination of the elements is to continue with gamma scanning and collection of plenum gases, and, ultimately, sectioning for microscopic examinations and chemical analyses of fuel and cladding materials.

b. Compatibility of (U, Pu) Carbides with Potential Cladding Materials. Compatibility tests previously reported (see Progress Report for October 1966, ANL-7267, pp. 38-39) included some showing that V-15 w/o Cr-5 w/o Ti was unattacked by (U, Pu) carbide (stoichiometric or nonstoichiometric) at 800°C, but that V-20 w/o Ti showed grain-boundary attack for a distance of 95 μ after 1000 hr at the same test temperature. Accordingly, couples of (U, Pu) carbide with V-20 w/o Ti were tested at 700°C and were reported (see Progress Report for November 1966, ANL-7279, pp. 26-30) to be in the process of preparation for metallographic examination in order to assess the effect of lowering the temperature of testing.

The compatibility tests of vanadium-20 w/o titanium at 700°C for 1000 hr have been completed and have shown a more extensive reaction zone from contact with hyperstoichiometric (5.25 w/o equivalent C) than with stoichiometric (4.83 w/o equivalent C) uranium-20 w/o plutonium carbide. In addition to the main reaction zone, there appears to be some grain-boundary penetration into the jacket material. The extent of the reactions are shown in Table XXIV. Vanadium alloys with a lower titanium content should be available for comparative tests in the near future.

TABLE XXIV. Extent of Reaction Observed Metallographically between (U, Pu) Carbides and Vanadium-20 w/o Titanium after 1000 hr at 700°C

Type of (U, Pu) Carbide	Reaction Zone (μ)		Grain-boundary Penetration (Distance from Interface) (μ)	
	Average	Maximum	Average	Maximum
Stoichiometric	10	35	50	80
Hyperstoichiometric	18	55	65	100

Other jacket-carbide couples reported in November (ANL-7279) to be in the process of preparation for examination were Type 304 stainless steel in contact with a carbide in which $(U, Pu)_2C_3$ was the major phase. (The equivalent carbon content was 6.75 w/o.) These couples had been heated to 800°C.

As reported in still an earlier test (800°C for 1000 hr) with Type 304 stainless steel (see Progress Report for April 1966, ANL-7204, p. 17), a differently etched band was found adjacent to the interface in the case of the hyperstoichiometric carbide composition, but was not found in the case of the stoichiometric composition. Efforts to show a correlation between this band and an increased carbon content in the steel have not been successful. Results of three recently completed tests involving (U, Pu) carbides and Type 304 stainless steel indicate no correlation between this band and the presence of $(U, Pu)_2C_3$. Compatibility couples containing both the stoichiometric and the hyperstoichiometric compositions and heat-treated at 800°C for 4000 hr show no significant differences in the stainless steel as observed metallographically. However, the previously mentioned band was present in both samples to a depth of 15 to 30 μ , comparable to the depth after 1000 hr. A test of Type 304 stainless steel for 1000 hr at 800°C with a (U, Pu) carbide composition containing 6.75 w/o equivalent carbon [approximately 90% $(U, Pu)_2C_3$] showed no effect on the stainless steel other than that observed with compositions with lower carbon contents.

3. Fuel Cladding and Structure

a. Development of Refractory-metal Alloys for Service in Oxygen-contaminated Sodium. Vanadium-5 w/o chromium was reported (see Progress Report for November 1966, ANL-7279, pp. 30-32) to be showing improved resistance to corrosion in flowing sodium at 650°C (oxygen ≤ 10 wt ppm) of an order of magnitude better than that of V-20 w/o Ti.

The corrosion behavior of a few selected vanadium alloys is being investigated at 550°C. Tests at two oxygen levels (coldtrapped at 120°C and at 170°C) are being run in two similar oxygen-refreshed autoclave systems. The oxygen levels were chosen to bracket the expected equilibrium concentration in the sodium for the formation of a vanadium oxide phase.

At the higher oxygen concentration (as previously noted at 650°C) all of the alloys lost corrosion product (Table XXV). A plot of these losses versus time shows nearly linear kinetics with slopes (corrosion rates) of about 60 mdd (mg per dm²-day) for the first three samples and 13 for the V-20 w/o Ti. The rate for V-20 w/o Ti at 650°C is about 100 mdd (under similar oxygen conditions), indicating the strong effect of temperature on the corrosion process when a vanadium oxide corrosion product is formed. The large difference in weight loss between V-20 w/o Ti and the other alloys was unexpected, based on one-week exposures at 650°C.

TABLE XXV. Corrosion of Vanadium Alloys
in Cold-trapped Sodium^a at 550°C

Alloy or Element (w/o)	Weight Loss (mg/cm ²) at Indicated Time				
	1 week	2 weeks	3 weeks	6 weeks	8 weeks
V-5 Ti-15 Cr (cold rolled)	4.25	9.80	18.8	27.0	34.7
V-5 Ti-15 Cr (1250°C HT)	4.09	9.55	19.2	27.1	35.1
V-5 Cr (annealed)	3.73	9.24	18.6	26.0	31.8
V-20 Ti (annealed)	0.75	1.80	3.57	5.52	7.36
Oxygen Level ^b (wt ppm) at Indicated Time					
Sodium	13.7	13.1	11.5	11.0	12.4

^aCold trap at 170°C.

^bMercury amalgamation technique.

At the lower oxygen concentration only weight gains have been recorded to date (Table XXVI). The scatter of values (due to the very small weight changes) is too great for reliable estimation of the kinetics of the process. No exact comparison with 650°C experience is possible for the longer times, but at one week the 550°C samples gained about half as much weight as the 650°C specimens. No dark corrosion product has been noted in this test although a brown tint on the corroded specimen becomes obvious on comparison with a freshly ground sample.

TABLE XXVI. Corrosion of Vanadium Alloys
in Cold-trapped Sodium^a at 550°C

Alloy or Element (w/o)	Weight Gain (mg/cm ²) at Indicated Time			
	1 week	2 weeks	4 weeks	6 weeks
V-5 Ti-15 Cr (cold rolled)	0.1 ⁵	0.2 ⁷	0.3 ⁵	0.4 ²
V-5 Ti-15 Cr (1250°C HT)	0.1 ²	0.2 ³	0.3 ¹	0.4 ³
V-5 Cr (annealed)	0.08	0.2 ³	0.3 ¹	0.5 ⁹
V-20 Ti (annealed)	0.2 ⁵	0.2 ⁹	0.3 ⁸	0.5 ⁵
Oxygen Level ^b (wt ppm) at Indicated Time				
Sodium	-	9.1	8.7	8.6

^aCold trap at 120°C.

^bMercury amalgamation technique.

The microstructure of V-20 w/o Ti samples removed at one and four weeks were quite similar for both tests at 550°C in spite of the large differences in weight change. At one week the hardened zone was 10 to 12 μ deep, and at four weeks 22 to 25 μ .

b. In-reactor Corrosion of Jacket and Structural Materials by Sodium. The final judgment as to the adequacy of a proposed reactor material must be based upon the performance of prototype components in an operating reactor. A necessary intermediate step between ex-reactor materials study and development and in-reactor component testing is the study of the behavior of materials per se (not as particular structures) in the reactor environment.

Accordingly, as described in the Metallurgy Division Annual Report for 1965 (ANL-7155, p. 77) a cooperative program between the Engineering Irradiation and Applied Corrosion Groups is under way. Unencapsulated samples are being exposed to flowing sodium in Metallurgy Division facilities in the core of EBR-II. Sample materials include vanadium, vanadium alloys, stainless steel, and nickel-base alloys.

The first group of 12 samples (contained in three pins) have been removed after exposure in the EBR-II core for about 14 months (not all this time was at full power). Full-power sodium temperatures were either 510 or 530°C. Examination of the samples is now in progress. Preliminary results are that both Type 304 stainless steel and nickel-base samples show essentially no weight changes, and that V-20 w/o Ti and V-40 w/o Ti show small weight losses but no detectable (to 0.1 mm) dimensional changes. An unalloyed vanadium sample was severely attacked (approximately 50% weight loss).

c. Jacket Development. Tests of V-15 w/o Cr-5 w/o Ti and of V-5 w/o Cr (see Progress Report for December 1966, ANL-7286, pp. 30-31) indicated that the vanadium-base ternary alloy and a V-Cr binary (V-10 w/o Cr was recommended for trial because of its higher strength) merited a scale-up of fabrication. Ingots of both alloys had been consolidated (by methods developed for V-15 w/o Ti-7.5 w/o Cr), machined, canned, and extruded. Billets for sheet bar and rod, and for the preparation of a tube-blank billet were re-extruded.

During January, determinations of the effects of thermal treatments upon the microstructure, hardness, and fabricability of these extruded products were in progress. The extruded structures are not fabricable at room temperature, but early results indicate that cold workability can be regained by exposing the extruded material to a specific thermal treatment.

The two alloys exhibited a marked difference in grain-growth behavior. The 15 w/o Cr ternary alloy revealed only a slight grain growth

after a 1200°C, one-hour thermal treatment (compared with the as-extruded grain size). The change in grain size was quite pronounced for the V-10 w/o Cr alloy similarly treated.

4. Fuel Reprocessing

a. Processes for Fast Reactor Fuels. Pyrochemical processing of fast reactor fuels of the ceramic type (e.g., oxide or carbide) clad with stainless steel is being investigated. A conceptual flowsheet currently under investigation has been described previously (see Progress Report for September 1966, ANL-7255, pp. 30-31). The process utilizes liquid metal-molten salt extractions and salt-transport separations⁵ for the separation of fissile and fertile constituents of the fuel from the fission products. Recent laboratory investigations of the current flowsheet are discussed below.

The liquid donor solvent (Cu-33 w/o Mg alloy) in the salt-transport separations step contains the plutonium, uranium, and noble and refractory metal fission products. In this separations step, the plutonium, which is in solution, is selectively transferred through a molten MgCl_2 -based salt to a liquid acceptor alloy. The uranium and the noble and refractory metal fission products remain in the donor solvent. The uranium is present as a precipitated metal phase, and can be recovered by decanting the supernatant copper-magnesium alloy and washing with magnesium to remove residual copper. The removal of the noble and refractory metal fission products is dependent upon their solubility in the copper-magnesium alloy and the extent of their coprecipitation with the uranium.⁶

The solubility of various fission product elements (molybdenum, niobium, cerium, and ruthenium) in liquid Cu-33 w/o Mg at temperatures between about 600 and 850°C were determined. The solubilities were as follows:

<u>Fission Product Element</u>	<u>Solubility (w/o)</u>
Molybdenum	$\sim 10^{-3}$ (~600 to 850°C)
Niobium	$\leq 2 \times 10^{-3}$ (~600 to 850°C) ^a
Cerium	~ 10 (600 to 800°C)
Ruthenium	0.6 (at 600°C) to 1.5 (at 850°C)

^aThe limit of detection.

⁵Salt-transport separations are based upon the selective transfer of fissile and fertile materials from one liquid-metal solution (donor) to another (acceptor) by cycling a molten salt phase which acts as a carrier between the two metal solutions. Noble and refractory metal fission products remain in the donor alloy.

⁶If higher decontamination of uranium is required, the uranium can be purified by a salt-transport step. In either case, the metallic uranium product is retorted to remove the adhering solvent metals.

The results indicate the following:

(i) At process temperatures (~600 to 850°C), molybdenum and niobium will not be in solution in the liquid donor alloy. Since these fission products may not be separated from the precipitated uranium by decantation of the donor alloy, other methods for their removal are being investigated.

(ii) At process temperatures (~600 to 850°C), cerium and ruthenium will be in solution in the liquid donor alloy. Thus, these two fission products will be removed in the donor alloy during the decantation step, provided neither fission product coprecipitates with uranium.

In work related to the salt-transport step, the distribution coefficient of neptunium ($K_D = w/o \text{ Np in salt} / w/o \text{ Np in metal}$) between molten 50 m/o $MgCl_2$ -25 m/o $NaCl$ -25 m/o $CaCl_2$ and liquid Zn-10 w/o Mg at 700°C was determined to be 2.60×10^{-2} .

F. Design Concept Analyses and Advanced Systems Evaluation

1. 1000-MWe Study

Contract negotiation meetings were held with Combustion Engineering Inc. on January 12 and the Babcock & Wilcox Co. on January 19. New AEC background patent and technical data requirements have been transmitted to the proposed contractors for their review and comment.

G. General Research and Development

1. Fast Reactor Core-parameter Study

Work is continuing on the neutronic analysis of cores having a height-to-diameter ratio (H/D) of 0.1, on the sodium-void calculations, and on the effects of various blanket thicknesses. After this work is completed, analyses of radially zoned cores will be started.

A computer code has been developed to calculate the core composition (fuel, structure, coolant, and control) for a number of circular cores having a wide range of core power density and geometry. This code, which is operational and has been extended to calculate nominal coolant velocities in the core subassemblies at midlife, is being revised to perform the complete steady-state thermal analysis--including calculation of temperature conditions in the core at the start-of-life, the temperature conditions in the radial blankets at the end-of-life, and the necessary coolant orificing to remove some of the clad and coolant temperature disparity in the various reactor regions.

III. GENERAL REACTOR TECHNOLOGY

A. Applied and Reactor Physics Development

1. Reactor Physics Laboratory On-line Computer System

For some time a DDP-24 computer has been used for on-line data reduction in various nuclear reactor experiments. Handling data in this way increases the efficiency of the experimental program because the rapid availability of the reduced data allows immediate assessment of the significance and validity of the results. With this on-line capability, an experiment can be instantly altered or modified. In addition, the computer performs the functions of a number of different special-purpose instruments that would otherwise be required.

The present system has an 8192-word memory and is equipped with paper-tape input and output, card reader, typewriter, digital graph plotter, line printer, oscilloscope display, and two 24-bit parallel input channels.

Fig. 19. Block Diagram of Computer System

As depicted in the block diagram of Fig. 19, each parallel channel is connected to a 24-bit binary scaler used for on-line data collection. Each binary stage of this scaler has input and output connections brought out to an array of jacks on the front panel, so any scaler configuration may be obtained by appropriate wiring of a plugboard. If desired, the 24 stages may be connected as two or more separate scalars, so that two or more quantities may be counted. In that case, the input work is unpacked by the computer to separate the quantities. A resolving time capability of $0.1 \mu\text{sec}$ is available.

Scaler read-in is initiated by an externally supplied pulse. When a read pulse appears, the input gates are closed, and counting halts. A signal is then sent to the computer, indicating that the data are ready to be read. On completion of the read operation, the scalars are reset to zero. Then counting resumes. Counting is interrupted for less than $25 \mu\text{sec}$ if the computer is ready for the data at the time the read pulse occurs. By throwing a switch, the two scalars can be made to work independently or together. In the latter case, counting in both scalars will halt simultaneously, and the computer will be told that both scalars are ready to read. Counting will resume simultaneously only after both scalars have been read and reset.

The system is frequently used for the measurement of reactor periods. Pulses from a neutron counter (or from an ion chamber, current amplifier, and voltage-to-frequency converter) are fed to the scaler input. The reactor operator is provided with a switch connected to one of the computer's external sense lines. When the operator is ready for the period measurement, he throws the switch. The scaler data are then repetitively read at selected time intervals. Meanwhile the computer is accumulating the sums required for a least-squares fit to an exponential rise. When the measurement is over, the operator throws the switch back. The computer stops reading data and immediately begins to print out the results. These results can include the period of the exponential and associated error (assuming Poisson-distributed counts), the reactivity and associated error (using a set of delayed neutron parameters that has been read in before the run), and the least-squares sum. Optionally, the output also includes the observed and fitted counts, standard deviation, and residual for each data point.

This system is also routinely used for calibrating control rods. Here the rod is steadily moved in the direction of decreasing reactivity. The resulting flux record is converted to reactivity as received. Rod motion is monitored by counting pulses produced by a device on the rod-position-indicator shaft. After the run is complete, a graph of the rod worth is automatically generated.

2. Spectrum Autocorrelation

The use of spectrum autocorrelation (see Progress Report for December 1966, ANL-7286, pp. 38-39) to determine error distribution for cross sections suggests that the same technique be used to find the parameters of isolated resonances. The information which can be extracted in this way is, however, disappointingly meager. The reason is simple: In the narrow resonance approximation the flux in the neighborhood of the resonance is, ignoring the factor of $1/E$, given by

$$F = (1 + X^2) / (1 + \beta + X^2),$$

where

$$F(X) = \text{flux};$$

$$X = \text{reduced energy} = 2(E - E_r) / \Gamma;$$

$$\beta = \Sigma_0 / (\Sigma_M + \Sigma_p), \text{ assumed small};$$

$$\Sigma_M = \text{moderator cross section (macroscopic)};$$

$$\Sigma_0 = \text{resonance peak cross section (macroscopic)};$$

$$\Sigma_p = \text{resonant nuclide potential scattering (macroscopic)};$$

E_r = resonance energy;

E = neutron energy;

Γ = resonance width.

The autocorrelation function computed for an interval of several widths is just proportional to a constant minus $\pi\beta\Gamma$. In most instances $\Sigma_0\Gamma$ (which is the significant part of $\beta\Gamma$) is more readily measured by conventional techniques. To extract more information, correlations of higher order have to be used.

3. Effect of Epithermal Spectra on Thermal Cross Sections

A study is being made of the effect of the epithermal spectrum on thermal cross sections. The integral transport theory thermalization code THERMØS⁷ presently employs an asymptotic flux ϕ_{asy} proportional to $1/E$ in calculating the slowing-down sources into the thermal region. By assuming that $\phi_{asy} \sim E^N$, where N is an arbitrary number, various slowing-down sources may be computed within THERMØS, and revised thermal cross sections may be generated.

In addition, realistic epithermal spectra, as generated in fast reactor codes such as MC², will also be used to provide source terms for the thermal region. For fast reactors in which a steam coolant is used, the effects of revised thermal-fission cross sections on k_{eff} and B^2 calculations in MC² will also be investigated.

4. ZPR-7 Physics Analysis

Work on the analysis of the Hi-C critical experiments has been completed, and the first draft of ANL-7270, "Analysis of High Conversion Critical Experiments," is now ready for typing. Recent work on Hi-C analysis has included calculation of gold and indium cadmium ratios and considerations of the effects of cross-section changes on reactivity.

Calculations of the gold and indium cadmium ratios were carried out using the B512/RP code.⁸ Agreement between theory and experiment is reasonably good. The agreement is better for indium than for gold, and better for the more loosely packed lattices than for the highly undermoderated ones. The approximations made in the calculations are such that one could expect better agreement for the looser lattices.

The changes in reactivity of Hi-C lattices produced by changing various important cross sections by 10% were computed. The results

⁷Toppel, B. J., and Baksys, I., The Argonne-revised THERMØS Code, ANL-7023 (1965).

⁸Pennington, E. M., and Boettinger, W. L., Foil Activation Programs, B512/RP, ANL-6822 (1964).

demonstrated the great sensitivity of the Hi-C lattices to changes of non-thermal cross sections in general. Sensitivity to the U^{238} capture cross section is especially pronounced. A 10% increase in the capture cross section of U^{238} in group 2 of a four-group scheme decreases reactivity by about 0.4% for the loosest lattice studied experimentally and 2.5% for the tightest one. A similar change in $\sigma_{n\gamma}$ for U^{238} in group 3 results in reactivity changes of -1.4% and -2.8% in the loosest and tightest lattices, respectively. It was mainly the large differences in $\sigma_{n\gamma}$ for U^{238} in group 2 which caused the large reactivity changes when U^{238} cross sections obtained from Hanford were substituted for those in the original GAM-I library.

5. Reactor Computation System

A set of problems have been run to illustrate the dependence of RABBLE computation time upon the geometry option, the number of regions, the number of resonances, and the width of a fine group. Also, these problems give an indication of the dependence of the accuracy of a computation upon the space and the energy mesh, and for a slab cell of the angular distribution of the interface currents.

The problems considered a cell with two compositions; the inner composition was pure U^{238} , and the outer composition was a mixture of U^{238} , sodium, and iron. The volumes of the compositions were in the ratio of 3 to 13. The dimensions of the inner composition were chosen such as to have the same ratio of surface to volume for the cylindrical and slab cells, so that we would expect these cells to be equivalent. The energy range of the computation was from 1.4 to 1.0 keV, and 25 resonances of U^{238} were included.

The results of these computations are given in Table XXVII. In problem 4, the extra 25 resonances were made extremely weak so as to

TABLE XXVII. Results of Sample Calculations for the Effect of the Options on Accuracy and Computation Time

Prob.	NOPT	Number of Resonances	Number of Regions	Absorption Probability	Effective Absorption Cross Section ^b		Run Time (sec)
					Comp. 1	Comp. 2	
1	0	25	1	0.26233	0.8045	-	67
2	1	25	8	0.31149	0.6527	0.8275	135
3	1	25	12	0.31149	0.6532	0.8273	172
4	1	50	8	0.31149	0.6527	0.8275	176
5	1	25	8	0.31109	0.6527	0.8277	251
6	2	25	8	0.31237	0.6611	0.8208	113
7	2	25	12	0.31260	0.6626	0.8197	134
8	3	25	8	0.31036	0.6488	0.8304	129
9	3	25	12	0.31023	0.6483	0.8307	161

affect neither the absorption probability nor the effective cross sections. From Table XXVII, we see that the homogeneous option ($N_{OPT}=0$) requires appreciably smaller running times than the three heterogeneous options. The slab option with cosine currents ($N_{OPT}=2$) is the next least time-consuming, and the other two heterogeneous options require about the same running time. From these results, the running time appears to be roughly proportional to the product of the number of fine groups and the number of resonances.

For the heterogeneous problems, the accuracy of the computation was not strongly dependent on the number of regions into which the cell is subdivided. Increasing the number of regions from eight to twelve had a negligible effect on the values of the computed quantities. This is not surprising in that integral transport methods that do not allow space dependence of the source within a composition frequently yield adequate results. Therefore an excessive number of regions is not necessary for precise calculations. It is also seen that the angular dependence of the interface currents does not have a large effect on the computed quantities for the slab option. The assumption of cosine interface currents causes the absorption probability to be overestimated by less than 1% and causes errors in effective absorption cross sections of between 1-2%.

The material composition for the homogeneous cell was obtained by homogenizing the two-composition cell. It is interesting that the homogeneous problem significantly underestimates the absorption probability. The explanation for this effect is probably that in homogenizing the cell relatively high scattering material is introduced in the inner composition, which allows neutrons to be scattered past resonances.

B. Reactor Fuels and Materials Development

1. Fuels and Cladding

a. Irradiation of High-temperature Materials. Eight specimens of US were reported to be under irradiation in MTR at maximum jacket temperatures of 380 to 750°C and burnups of 4.5 to 7.4 a/o U (see Progress Report for December 1966, ANL-7286, p. 45). Seven specimens of UC-20 w/o PuC at maximum jacket temperatures of 470 to 715°C were then at burnups of 2.0 to 7.1 a/o (U+Pu), and one specimen of PuC had attained a burnup of 2.0 a/o Pu at a maximum jacket temperature of 700°C. All jackets are of Nb-1 w/o Zr. These irradiations, which are being made in instrumented capsules as part of a high-temperature materials program, are continuing under conditions summarized in Table XXVIII.

TABLE XXVIII. Status of Irradiations of Ceramic Fuel, Clad in Nb-1 w/o Zr, in MTR

Capsule or S/A No.	Specimen Number	Design Parameters				Operating Conditions			
		Fuel Composition (w/o)	Effective Density (%)	Cladding OD (in.)	Cladding Thickness (in.)	Power Density (kW/cc) ^a	Max Cladding Temp (°C)	Burnup to Date	
								a/o (U + Pu)	fiss/cc x 10 ⁻²⁰ ^a
56-11	MV-2	UC-20 PuC	79	0.281	0.012	1.2	470	6.8	17.6
56-8	MV-3	UC-20 PuC	81	0.281	0.012	1.2	715	7.3	19.4
56-8	MV-5	UC-20 PuC	80	0.281	0.012	1.2	705	7.0	18.4
56-11	MV-6	UC-20 PuC	80	0.281	0.012	1.2	480	7.3	19.2
56-13	Z-4	UC-20 PuC	79	0.174	0.015	1.2	665	2.2	5.7
56-13	Z-5	UC-20 PuC	79	0.174	0.015	1.2	585	2.2	5.7
56-13	Z-7	UC-20 PuC	79	0.174	0.015	1.2	570	2.2	5.7
56-13	C-45	PuC	84	0.174	0.009	1.4	700	2.2	6.0
56-8	S-7	US	80	0.281	0.012	1.0	535	5.6	10.8
56-8	S-8	US	89	0.281	0.012	1.0	725	7.6	16.4
56-8	S-9	US	76	0.281	0.012	1.0	750	7.6	14.0
56-8	S-10	US	91	0.281	0.012	1.0	690	7.6	16.8
56-11	S-15	US	82	0.281	0.012	1.0	380	4.7	9.4
56-11	S-16	US	90	0.281	0.012	1.0	510	6.6	14.4
56-11	S-17	US	88	0.281	0.012	1.0	500	5.3	11.3
56-11	S-18	US	77	0.281	0.012	1.0	610	7.1	13.3

^aBased on effective density.

A metallographic examination was conducted of a control specimen of vibratorily compacted powders of UC-20 w/o PuC_{1.02} that has been heat-treated at 1300°C for 70 hr. The temperature was selected to simulate approximately the maximum centerline temperature of specimens when they are irradiated at maximum jacket surface temperatures ranging from 650 to 1000°C. Figure 20 shows the photomicrographs of (a) the heat-treated control specimen and (b) a vibratorily compacted UC-20 w/o PuC_{1.33} specimen irradiated to a 5.82 a/o burnup of the uranium and plutonium atoms

(a) Unirradiated UC-20 w/o PuC_{1.02} Specimen
Heat-treated at 1300°C for 70 hr

(b) UC-20 w/o PuC_{1.33} Specimen Irradiated to
5.82 a/o Burnup of Heavy Atoms (1.54 x
10²¹ fiss/cc)

Fig. 20. Photomicrographs of Vibratorily Compacted UC-20 w/o PuC Fuel Specimens

(1.54×10^{21} fiss/cc). Little sintering of the $\text{PuC}_{1.02}$ particles occurred in the unirradiated specimen during the heat treatment at 1300°C , and some $\text{PuC}_{1.02}$ particles fell out during vibratory grinding and polishing. This is in contrast with the extensive swelling and sintering of the $\text{PuC}_{1.33}$ particles, which occurred in the irradiated specimen. The PuC phase formed an almost continuous network around the coarse UC particles. The depletion of UC_2 platelets at the surface of the UC_{1+x} particles and internal cracks observed with irradiated specimens were not observed with the unirradiated specimen. However, it was not possible to establish whether the depletion of UC_2 platelets is due to a gas-phase reaction, solid-solution formation, or an irradiation-induced phenomenon.

Electron-microscopic studies were conducted with irradiated specimens of vibratorily compacted powders of a hyperstoichiometric carbide (UC-20 w/o $\text{PuC}_{1.33}$) and a hypostoichiometric carbide (UC-20 w/o $\text{PuC}_{0.89}$). Microstructural changes observed in an irradiated UC-20 w/o $\text{PuC}_{1.33}$ (5.82 a/o burnup) were presented in an earlier report. The analyzed burnup of the UC-20 w/o $\text{PuC}_{0.89}$ specimen was 6.02 a/o of uranium and plutonium atoms (1.60×10^{21} fiss/cc). This corresponds to a burnup of 30.1 a/o in the $\text{PuC}_{0.89}$ phase, since the uranium was not enriched. The fission-gas bubbles observed in the $\text{PuC}_{0.89}$ were generally larger and more interconnected than those in $\text{PuC}_{1.33}$. This gave rise to the greater fission-gas release by a factor of about ten in the $\text{PuC}_{0.89}$ specimen. Figures 21 (a) and 21 (b) are electron micrographs showing fission-gas bubbles of various sizes. Figure 22 compares the distribution of fission-gas bubbles in the $\text{PuC}_{0.89}$ and the $\text{PuC}_{1.33}$, determined by use of both optical and electron microscopy. Spherically shaped second-phase particles of several sizes as shown in Fig. 23 were observed in $\text{PuC}_{1.33}$. These second-phase particles

—| 5 μ |—

(a) Typical Fission-gas Bubbles

—| 2 μ |—

(b) Fission-gas Bubble with Small Bubbles on Surface

Fig. 21. Electron Micrographs Showing Fission-gas Bubbles in $\text{PuC}_{0.8}$ with 30% Burnup

Fig. 22. Comparison of Size Distributions of Fission-gas Bubbles in $\text{PuC}_{1.33}$ and $\text{PuC}_{0.89}$

Fig. 23. Second-phase Particles of Several Sizes in $\text{PuC}_{1.33}$ Irradiated to 29 a/o Burnup (1.52×10^{21} fissions/cc)

were assumed to be associated with solid fission products in the $\text{PuC}_{1.33}$. The second-phase particles were far less frequently observed in the $\text{PuC}_{0.89}$ than in the $\text{PuC}_{1.33}$. It will be explored whether or not the difference in the distribution of the second-phase particles is related to the difference in the distribution of fission-gas bubbles in the $\text{PuC}_{0.89}$ and in the $\text{PuC}_{1.33}$, and in turn is related to the difference in the swelling and fission-gas-release behaviors of the two fuels.

A shielded microprobe is to be installed during 1967 to identify the various solid fission product phases, and to establish whether or not any solid solution formation occurred across the UC-PuC phase boundaries.

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

b. Studies of Corrosion in Lithium at Elevated Temperatures. The phenomenon of inhibition of corrosion by lithium in contact with tantalum was reported previously (see Progress Reports for September and November 1966, ANL-7255, pp. 45-46, and ANL-7279, p. 51). A protective layer that is thermodynamically stable and is effective as a diffusion barrier was shown to form on tantalum in silicon-containing lithium at 1200°C.

TABLE XXIX. Status of Irradiations of Metal Fuel, in Capsule No. CP-50 and Clad in 0.015-in.-thick, 0.198-in.-OD V-20 w/o Ti, in CP-5

Specimen Number	Design Parameters		Operating Conditions			
	Fuel Composition (w/o)	Effective Density (%)	Max (kW/ft)	Max Cladding Temp (°C)	Burnup to Date	
					a/o (U+Pu)	fiss/cc x 10 ^{-20a}
1N16	Th-20 U	74.7	8.7	590	15.9	7.6
4N19	Th-20 U	74.0	8.7	560	15.9	7.6
2N17	Th-10 Pu-10 U	74.1	9.2	590	18.5	8.6
5N20	Th-10 Pu-10 U	73.0	9.2	590	18.5	8.6
3N18	Th-10 Pu-20 U	73.5	8.7	560	10.8	7.8
6N21	Th-10 Pu-20 U	73.5	8.7	560	10.8	7.8

^aBased on effective density.

The reaction zone consists of two regions: an outer layer, Ta₂Si, is 5 μ thick, and under this reaction zone there is a diffusion region that extends 1500 μ to the inner base metal.

Chemical and spectrochemical analyses reveal that the silicon content of the latter region decreases from 340 to 30 ppm silicon with increasing depth of diffusion penetration.

Initial results show that the best protective concentration of silicon in lithium is about 1.50 a/o under the above test conditions. There appears to be no additional protective value if the additive is greater than 1.50 a/o. The study of the Li-Si-Ta system is continuing.

c. Thermal Stability of Plutonium Ceramics. Pellets of PuO₂ for vaporization tests were reported (see Progress Report for October 1966, ANL-7267, p. 57) to have been supported in wire made of rhenium, which is more compatible with PuO₂ than is molybdenum or tungsten. Pellets of U-Pu oxide attained 80% of full density when sintered at 1550°C in argon containing 6 v/o of hydrogen. Pellets heated for 2 hr at 1200°C, or even at 750°C, in helium containing 2 v/o of oxygen, would crack to the point of disintegration. In vacuum evaporation tests at 2000°K the weight loss of the rhenium wire was only 3 to 4% of that of the pellet.

Since October, experiments on the evaporation of PuO₂ were continued, with emphasis on the determination of the kinetics of evaporation and on the evaluation of the oxygen-to-metal (O/M) ratios in specimens after evaporation. Efforts were also made to find the most suitable crucible material for containing the plutonia.

A PuO_2 pellet was successfully fired in an iridium crucible in vacuo at 2116°K . The weight loss of the iridium crucible was less than 1% of the total 52.6-mg loss, and the only evidence of reaction was slight discoloration of the bottom of the pellet. Among the refractory metals iridium, therefore, appears to be the most satisfactory crucible material for PuO_2 .

Oxygen-to-metal ratios were determined by equilibrating specimens in a 10/1 CO/CO_2 gas mixture at 1123°K , as described by Markin et al.⁹ The results of these determinations were identical with those obtained previously by heat treatment of a specimen in a helium-1 v/o oxygen atmosphere at 1473°K .

Numerous evaporation runs were made in a vacuum with PuO_2 pellets at temperatures from 1930 to 2115°K . The wire coil used to contain the pellets was made of rhenium, which had adequate compatibility in view of the smallness of the area of interface between the wire and the specimen. A run was continued until the evaporation rate became linear, indicating apparent congruent evaporation. In order to establish the congruent composition, the O/M ratio of the specimen was determined after each run, but the composition of the specimen changed with time even though the weight-loss rate appeared constant.

Table XXX shows the variation of the O/M ratio as a function of time and weight loss in specimens evaporated at about 2115°K . It is evident that most of the oxygen loss

TABLE XXX. O/M Ratios in PuO_2
Specimens after Evaporation

Run No.	T ($^\circ\text{K}$)	Time (min)	Weight Loss (%)	O/M
86	2114	10	0.94	1.929
87	2104	35	1.85	1.908
84	2110	60	3.50	1.896
85	2114	110	5.85	1.867

occurs at the very beginning of an evaporation run. Although the experimental weight loss-versus-time data indicate that the overall rate becomes linear after about 20 min, Table XXX shows that the O/M ratio is still decreasing at an evaporation time of 110 min, involving a total weight loss of almost 6%.

Further attempts will be made to determine whether a congruently evaporating composition can be obtained and to make vapor-pressure measurements in a rhenium Knudsen cell.

2. Radiation Damage to Structural Materials

1. Effect of Fast-neutron Irradiation on Jacket Materials. One-hundred-and-three tube-burst specimens and 321 tensile-type specimens

⁹Marken, T. L., Walter, A. J., and Bones, R. J., The Determination of Oxygen/Metal Ratios for Uranium, Plutonium and (U · Pu) Oxides, AERE-R 4608 (1964).

were reported (see Progress Report for December 1966, ANL-7286, pp. 45-46) to have received 4×10^{20} to 1.6×10^{22} n/cm² of exposure, which was continuing at between 500 and 650°C. Forty-eight tensile-type specimens, made severally of V-20 w/o Ti, Hastelloy-X, and Type 304 stainless steel, had then accumulated maximum exposures of 3.8×10^{22} n/cm² and had been removed from EBR-II for shipment to the Illinois site. A characterization of the EBR-II neutron spectra was also initiated to make the results being obtained more meaningful.

Sixty-three tube-burst specimens and 112 tensile-type specimens (V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr, Hastelloy-X, and Type 304 stainless steel) were inserted in EBR-II during December for irradiation at 650°C to 10^{23} n/cm².

In the present reporting period, the effects of elevated temperature coupled with fast-neutron irradiation on the creep, tensile, creep-rupture, and tube-burst properties of V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr, Hastelloy-X, Inconel 625, and AISI Type 304 stainless steel are being evaluated in the temperature and exposure ranges of 400 to 750°C and 10^{21} to 10^{23} n/cm², respectively. Thirty-five capsules, containing 103 tube-burst specimens and 420 tensile-type specimens are now being irradiated in EBR-II subassemblies XA08, XO10, XO16, XO18, and XO20 at temperatures between 500 and 650°C. These specimens have accumulated at the end of this report period maximum total neutron exposures ranging from 5.3×10^{20} to 3.7×10^{22} n/cm². The status and identification of these capsules are given in Table XXXI.

TABLE XXXI. Status of Cladding Materials Irradiations in EBR-II

S/A No.	Capsule Number	Design Parameters				Operating Conditions	
		Cladding Composition	Type of Specimen	No. of Specimens	Specimen Environment	Max Specimen Temp (°C)	Exposure to Date (n/cm ²)
XA08	AS-1	V-20 w/o Ti	Tube-burst	12	Argon-helium	540	3.4×10^{22}
XA08	AS-2	V-20 w/o Ti	Tube-burst	12	Argon-helium	540	3.4×10^{22}
XA08	AS-3	Hastelloy-X	Tube-burst	12	Argon-helium	540	3.6×10^{22}
XA08	AS-4	Hastelloy-X	Tube-burst	12	Argon-helium	540	3.6×10^{22}
XA08	AS-5	304 SS	Tube-burst	12	Argon-helium	540	3.6×10^{22}
XA08	AS-6	V-20 w/o Ti	Tensile	16	Argon-helium	580	3.7×10^{22}
XA08	AS-7	Hastelloy-X	Tensile	16	Argon-helium	580	3.4×10^{22}
XA08	AS-8	304 SS	Tensile	16	Argon-helium	580	3.4×10^{22}
XA08	AS-12	V-20 w/o Ti	Tensile	16	Argon-helium	580	3.3×10^{22}
XO10	AS-16	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.2×10^{22}
XO10	AS-17	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.3×10^{22}
XO10	AS-18	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.2×10^{22}
XO10	AS-19	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.2×10^{22}
XO10	AS-20	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.4×10^{22}
XO10	AS-21	V-20 w/o Ti	Tensile	13	Argon-helium	500	1.6×10^{22}
XO10	AS-22	Hastelloy-X	Tensile	13	Argon-helium	500	1.4×10^{22}
XO10	AS-23	304 SS	Tensile	13	Argon-helium	500	1.6×10^{22}
XO10	AS-24	304 SS	Tensile	13	Argon-helium	500	1.4×10^{22}
XO10	AS-25	304 SS	Tensile	13	Argon-helium	500	1.3×10^{22}
XO10	AS-26	304 SS	Tensile	13	Argon-helium	500	1.3×10^{22}
XO16	AS-29	304 SS	Tensile	9	Sodium	650	9.1×10^{20}
XO16	AS-30	V-15 w/o Ti-7.5 w/o Cr	Tensile	9	Sodium	650	8.6×10^{20}
XO16	AS-31	V-20 w/o Ti	Tensile	9	Sodium	650	8.2×10^{20}
XO16	AS-32	Hastelloy-X	Tensile	9	Sodium	650	9.3×10^{20}
XO16	AS-33	V-15 w/o Ti-7.5 w/o Cr	Tensile	9	Sodium	650	8.9×10^{20}
XO16	AS-50	V-15 w/o Ti-7.5 w/o Cr	Tensile	9	Sodium	650	8.5×10^{20}
XO16	AS-51	V-20 w/o Ti	Tensile	9	Sodium	650	8.1×10^{20}
XO16	AS-52	Hastelloy-X	Tensile	9	Sodium	650	9.1×10^{20}
XO16	AS-53	V-20 w/o Ti	Tensile	9	Sodium	650	8.3×10^{20}
XO18	AS-56	V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr	Tensile, Tube-burst	75	NaK	650	5.3×10^{21}
XO18	AS-57	304 SS, Hastelloy-X	Tensile, Tube-burst	75	NaK	650	5.3×10^{21}
XO18	AS-58	V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr	Tensile, Tube-burst	25	NaK	650	5.3×10^{21}
XO20	AS-59	V-20 w/o Ti	Tensile	9	Sodium	650	5.3×10^{20}
XO20	AS-60	V-20 w/o Ti	Tensile	9	Sodium	650	5.3×10^{20}

Seventeen capsules from subassembly XO13, containing 153 tensile-type specimens of Hastelloy-X, V-20 w/o Ti, V-15 w/o Ti-7.5 w/o Cr, and AISI Type 304 stainless steel, were received at the Illinois site. Disassembly, removal and cleaning of specimens, and temperature-monitor evaluation were initiated. Arrangements were made to ship the remaining capsules from subassembly XO13 and the six capsules from subassemblies XA07 and XO09 during the latter part of February. Nine capsules, containing 81 tensile-type specimens, and two capsules containing 18 tensile-type specimens were inserted in EBR-II subassemblies XO16 and XO20, respectively. The target exposures are 1×10^{22} n/cm² for subassembly XO16 and 5×10^{22} n/cm² for subassembly XO20.

3. Techniques of Fabrication and Testing

a. Ultrasonic Instrument and Transducer Development. Calculations of the acoustic impedances for three porous tungsten samples with particle sizes of 100 or 400 μ revealed (see Progress Report for November 1966, ANL-7279, p. 58) that a theoretical density below 70% would be required to achieve an acoustic impedance match with lead zirconate-titanate transducers. For Clevite's PZT-4 composition ($Z = 34 \times 10^5$ g cm⁻² sec⁻¹), an impedance match occurs at a density of about 67%.

The Coating and Jacketing Group, attempting to manufacture a 67% dense sample with 100 μ particles, produced one sample with a theoretical density of 70.8%. An acoustic impedance value of 42.5×10^5 g cm⁻² sec⁻¹ was calculated for this sample.

Methods of joining a porous tungsten backing to a silver-plated transducer are being evaluated. One method involves the deposition of a thin film of silver on the tungsten surface prior to the application of indium solder at the joint. Silver-loaded epoxy has also been applied at the joint with success. Bonding-layer thickness and its effect on the transducer characteristics have yet to be determined.

Further porous tungsten samples have been received from Sintered Specialties. These samples have theoretical densities around 60, 70, and 80%. The particle size of these samples is unknown. Evaluation of these samples is in progress.

b. Development of a Neutron-image Intensification System. Gamma-irradiation studies of an image orthicon television camera tube indicated (see ANL-7279, pp. 58-59) a loss in sensitivity of about 320X at an exposure of 10^8 R. Effects other than mere loss in light transmission through a darkening glass are indicated at this level by a great deal of noise.

The television image of a europium-activated Li^6I neutron scintillator was reported to be observable in an image orthicon system for incident thermal-neutron intensities as low as $10^6 \text{ n/cm}^2\text{-sec}$, as compared with $4 \times 10^6 \text{ n/cm}^2\text{-sec}$ for a Li^6F , ZnS(Ag) neutron scintillator.

Since the excellent sensitivity of the neutron-image intensifier suggests that it may be useful with relatively low-intensity neutron sources, some effort has been made to analyze the performance of several neutron-imaging detectors under conditions of low neutron intensity. Table XXXII summarizes data about many neutron-image detectors. These include two direct-exposure photographic methods, a fast metal-screen method (the double-screen rhodium-gadolinium combination), and a scintillator method (based on the Li^6F , ZnS(Ag) scintillator described by Wang *et al.*¹⁰ Two unusual, relatively new methods are also included; these are the LiF thermoluminescent technique¹¹ and the track-etch plastic-film method.¹² The table includes new data on the thermoluminescent technique¹³ and on

TABLE XXXII. Comparison of Neutron-imaging Methods

Method	Neutron Exposure Required to Equal Detector Response to 1 mR Co^{60} Gamma Radiation (n/cm^2)	Thermal- neutron Intensity Needed for Image ($\text{n/cm}^2\text{-sec}$)	Thermal- neutron Exposure Needed for Image (n/cm^2)	Detector Resolution (High Contrast Conditions) (mm)	Image Contrast Sensitivity (%)
Direct Exposure-- Metal Screens	10^5	-	5×10^6	0.025	2
Direct Exposure-- Scintillator	10^4	-	$10^4\text{--}10^5$	0.05	6
Neutron-image Intensifier	10^4	3×10^5	10^4	0.35	6
LiF Thermolumi- nescent Detector	5×10^3	-	10^9	0.125	10
U-Foil-Cellulose Nitrate Track Etch Detector	-	-	2×10^9	<0.025	10
Transfer Exposure					
Dysprosium	-	5×10^3	5×10^7	0.05	1
Gold	-	5×10^3	10^9	0.025	1
Indium	-	5×10^4	2×10^7	0.05	1

¹⁰Wang, S. P., Shull, C. G., and Phillips, W. C., Photography of Neutron Diffraction Patterns, Rev. Sci. Instr. **33**, 844 (1962).

¹¹Kasner, J., Berger, H., and Kraska, I. R., LiF Thermoluminescence for Neutron Image Storage, Nuclear Applications **2**, 252 (1966).

¹²Fleischer, R. L., Price, P. B., and Walker, R. M., Tracks of Charged Particles in Solids, Science **149**, 383 (1965).

¹³Kasner, J., Berger, H., and Kraska, I. R., A Thermoluminescent Image Detection Method for Neutron Radiography, to be published.

the track-etch method with a detector combination of an enriched uranium foil-cellulose nitrate plastic film.¹⁴ Both these techniques are comparable to direct-exposure methods in that one can accumulate image information with them for very long periods. Therefore, no minimum neutron intensity is listed for any of these techniques.

Several common transfer methods are also listed, involving dysprosium, gold, or indium. The neutron intensity listed for these transfer methods is the intensity one would need to obtain a reasonable, fast X-ray film density following a one-half-life neutron exposure and a three-half-life transfer period for the decay of the radioactive screen on the film. The neutron intensities listed, therefore, represent about the minimum intensity that can be used with the indicated transfer materials.

All the techniques are listed in terms of improving relative n- γ response. The track-etch and transfer methods, all essentially insensitive to gamma radiation, provide the best results in this area.

The neutron exposure required for an image is one that will yield the indicated resolution. A longer exposure might be required in many cases to achieve the indicated contrast sensitivity. The neutron-image intensifier does respond to thermal-neutron intensities lower than that indicated, but the indicated level of intensity is required if the indicated resolution is to be achieved by direct or television viewing of the neutron image.

The performance of the neutron-image intensifier is essentially comparable to that of the neutron scintillator technique, as one might expect. Significant improvements in relative n- γ response, and resolution and contrast capability appear possible only if a different approach to developing the intensifier input target is tried. Several possibilities are now under consideration.

For the particular problem of detecting low-level neutron intensities, the dysprosium transfer technique looks promising if the available neutron intensity is at least 5×10^3 n/cm²-sec. If the intensity is lower, then something other than a transfer method would have to be used. The plastic film, track-etch method is promising in terms of minimal gamma response and resolution, but does have some disadvantages in terms of total exposure required and contrast.

The neutron-image intensifier would offer more promise as a low-level neutron-imaging detector if the image information could easily be stored and if the relative n- γ response could be improved. Both these improvements appear possible.

¹⁴Berger, H., and Kraska, I. R., A Track-etch, Plastic Film Technique for Neutron Imaging, to be published.

c. Thermal Conductivity of Irradiated Fuel as a Function of Burn-up and Temperature. Arrangements have been made (see Progress Report for November 1966, ANL-7279, p. 59) to measure the thermal conductivity of irradiated U-5 w/o Fs at temperatures to 600°C. The vacuum system and furnace have been transferred to the cave area, and the set-up is being reassembled. Some delay is expected in the completion of this assembly work as it is necessary, before we proceed further, to repair the hot-cell hoist.

d. Tungsten-base Alloy Tubing. The Argonne National Laboratory three-roll tube-reducer was reported (see Progress Report for December 1966, ANL-7286, pp. 47-48) as being modified for warm-working operations with the objective of eliminating minor surface striations occurring on tube-blanks of tungsten and tungsten alloys extruded by the filled-billet technique, thus avoiding the need for uneconomic drawing and annealing sequences. The inner and outer surfaces of extruded W-Re-Mo alloys had been so improved by tube reduction that application of the method to tungsten-rhenium alloys and to tungsten (both more recalcitrant metals) reportedly was being planned. In addition, improved drawing lubricants and swaging techniques were reported to have resulted in more successful final drawing operations, both technically and economically.

Initial schedules during January called for ductile-core drawing W-Re alloys at 625 to 725°C with recrystallization anneals after two passes, each of 10% reduction in area. Heavier reductions are now possible in all tube sizes. The microstructure of a tube section drawn over a molybdenum core to a 47% reduction of area at 700°C and annealed at 1525°C for one hour is shown in Fig. 24. The annealed hardness of the tubing before reduction

As-drawn--47% R of A
Hardness: 52 R_C

Annealed: 1525°C--One hour
Hardness: 44 R_C

150X

150X

Fig. 24. As-drawn (47% Reduction of Area) and Recrystallized Structure of W-25 w/o Re Alloy Tubing

was 45 R_c ; after reduction (47%) 52 R_c ; and after annealing, at 1525°C, 44 R_c . Area reductions in excess of 60% have been successfully accomplished on small-diameter (<0.140-in.) tubing. The annealing temperature of 1525°C is believed to result in an optimum combination of softening without excessive grain growth.

C. Engineering Development

1. Heat Transfer, Fluid Flow, and Mechanics of Materials

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

Sodium has been circulated intermittently in the loop at temperatures to 500°F and a flowrate of 15 ft/sec during calibrations of the pressure transducers. The differential transducers are still operating erratically, so that higher-temperature operation using only the absolute transducers will be made for the present. Preparations for flowmeter calibrations using both pressure-drop and calorimetric methods are underway.

b. Heater Experiments

(i) Electron-bombardment Heater (EBH) Experiment. The original test facility has been shut down, and the heater and pool boiler removed due to a tank failure; this test is terminated following very satisfactory operation beyond original design limits. Further EBH tests will be carried out utilizing the new EBH test facility now in the procurement stage.

c. Heat Transfer in Double-pipe Heat Exchangers

(i) Countercurrent Turbulent Liquid-metal Flow. Installation of the second heat-exchanger test section and loop cleanup have been completed. Operation will begin shortly.

The first heat-exchanger test section has been dismantled and carefully inspected for possible causes of the anomalous experimental results obtained in the initial series of experiments. These results showed that experimentally determined heat-exchanger efficiencies and fully developed overall heat-transfer coefficients were functions of the fluid inlet-temperature difference (see Progress Report for September 1966, ANL-7255, p. 55). According to analytical predictions (e.g., by use of the traditional heat-exchanger design equations or the more exact analysis-from-fundamentals), efficiencies and heat-transfer coefficients should be independent

of the inlet-temperature difference. It was found that the insulating air space in the tube wall between the two fluids in the calming-length portion of the first test section was filled with mercury. Although this malfunction provided an extraneous source of heat transfer between the fluids, analysis based on the traditional heat-exchanger design equation showed that it could not explain the anomalous results.

Reformulation and extensions of the analysis-from-fundamentals for the "degenerate" case of equal-heat-capacity mass flow-rates on both sides of the exchanger have now resulted in a general analytical solution for plug, laminar, or turbulent flow for the generalized symmetrical duct (annular within an annulus) heat exchanger. When compared with previous analyses-from-fundamentals, the mathematics associated with this special but important case has additional features which appear unique in applied mathematics.

(ii) Nonsymmetrical Geometries. Loop construction continues. Several designs of the first test section have been considered; a "best" design has been selected. The purpose of the first series of experiments is to explore the effect on overall heat-transfer rates of eccentricity of the annular space in double-pipe heat exchangers using sodium.

(iii) Liquid-metal-heated Steam Generators. The formulation of a mathematical model of a simple, double-pipe, liquid-metal-heated steam generator has begun. The model consists of a concentric-tube exchanger with liquid metal flowing through the tube side. Liquid water enters the annular space and exits as steam. Heat transfer from the liquid metal will be formulated from fundamentals rather than on the traditional use of heat-transfer coefficients. Initially, heat flow through the tube wall will be assumed to be proportional to the temperature difference between the inside surface of the tube and the local saturation temperature of the steam. The local steam saturation temperature is "coupled" to the heat transfer by use of the vapor pressure-temperature relation together with an application of the one-dimensional equations of fluid motion to the vaporizing liquid in the annular space. The purposes of the formulation are to assist in the planning of future experiments and to provide a "more exact" prediction method for comparison with traditional design methods.

d. Heat Transfer in Liquid-metal-cooled Reactor Channels. A first draft of a paper describing recent research related to the significance of axial heat diffusion in liquid-metal-cooled ducts with nonuniform flux distributions has been completed.

e. Studies of Structural Dynamics of Core. Work continued on the survey of vibration of reactor core components; the survey report is being drafted for review during the next reporting period.

The small flow loop was completed; maximum flow obtained through the 2.25-in.-ID test section was 45 gpm, not enough to induce observable oscillations in the several brass rods tested. However, 1/2-in.-dia nylon rods, 30 in. long, did demonstrate observable vibrations.

With a flow of 45 gpm, the frequency spectrum of the displacement of the nylon test rod was plotted. The plot showed a dominant component at a frequency of 12.7 cps (this can be compared with natural frequencies of 10.1 cps in static water and 18.3 cps in air). These changes in frequency, when vibrating in static water and in flowing water, compare favorably with trends exhibited in the experimental data of Burgreen.¹⁵ At half the flowrate (approximately 22 gpm), the nylon rod vibrated at a reduced amplitude with a frequency of approximately 12.7 cps.

Both displacement and acceleration measurement techniques were developed.

(i) Displacement Measurement. The Photocon capacitance transducer can measure 1×10^{-4} in. of displacement of a vibrating element in air without touching the specimen; however, the transducer would not work in the highly conductive domestic water used in the test loop. So a probe having an outer Teflon shield and a central brass rod was developed; with this probe, the transducer can sense the motion of a rod under flow conditions with the dynamic range of the normal probes in air. A micrometer has been modified for calibration of this device in the loop. In future tests the probe will be streamlined to reduce turbulence.

(ii) Acceleration Measurement. An Endevco piezoelectric accelerometer was mounted in the rod. No difficulties were encountered with sealing the accelerometer or its cable to prevent water leaks or electrical grounds. However, the vibration was of too low a magnitude for the sensitivity of the accelerometer. Future tests with higher flows will allow use of this instrumentation.

A new loop is being constructed; completion is scheduled during the next month. This loop features 2-in.-ID outlet piping with a minimum of flow restrictions; a pump with a 20-hp motor is expected to deliver 200 gpm, giving flow velocities high enough to induce measurable vibrations in metal rods. A Potter flowmeter has been ordered.

2. Instrumentation and Control

An instrumentation research and development program was initiated. The principal objective is to develop in-core sensors needed to diagnose reactor performance and detect abnormalities; initial work concentrates on instrumentation important for reliable operation of large liquid-metal

¹⁵Burgreen, D., Byrnes, J. J., and Benforado, D. M., Vibration of Rods Induced by Water in Parallel Flow, Trans. ASME (1958), pp. 991-1103.

fast reactors. Methods of detecting local boiling in the coolant are being investigated. One such method, whose effectiveness will be evaluated in a high-temperature sodium environment, utilizes the acoustic-frequency spectrum, both audio- and ultra-sonic. Experimental and materials planning and a literature survey have begun.

D. Chemistry and Chemical Separations

1. Fluoride Volatility Processes

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

(i) Reaction of U_3O_8 with BrF_3 . The reaction of BrF_3 with U_3O_8 is of interest in the volatility processing of reactor fuels since the reaction proceeds at high rates at temperatures lower than those required for the BrF_5 - U_3O_8 reaction. A series of laboratory-scale experiments has been performed to study the kinetics of the BrF_3 - U_3O_8 reaction in the temperature range from 100 to 400°C and at BrF_3 partial pressures of 18 to 44 Torr. The reaction-rate data were treated using a diminishing-sphere model. The following equation was derived to represent the temperature and partial pressure dependence of the rate constant k for the U_3O_8 - BrF_3 reaction:

$$\log k (\text{min}^{-1}) = 1.3 \log P(\text{Torr}) - 3.691 - \frac{153.6}{T(^{\circ}\text{K})}.$$

In the temperature range from 200 to 250°C, the rate of reaction of U_3O_8 with BrF_3 was approximately 10 times the rate of reaction of U_3O_8 with BrF_5 . An activation energy of 0.7 kcal/mole was calculated from the data. This value compares favorably with a value of 0.9 kcal/mole reported by Iwasaki and Sakurai.¹⁶

Iwasaki and Sakurai, whose experiments were carried out under conditions that were roughly the same as ours, reported a minimum in the reaction rate constant at a temperature of about 230°C. At temperatures above 230°C the reaction rate increased slowly with an increase in temperature. This behavior was not observed in this study; no minimum in the rate constant was observed.

X-ray diffraction analyses of solid residues remaining after partial conversion of U_3O_8 to UF_6 showed no evidence for the formation of UO_2F_2 as an intermediate compound in the fluorination reaction.

¹⁶Iwasaki, M., and Sakurai, T., J. Nucl. Sci. Technol. (Tokyo) 2(6), 225-228 (June 1965).

(ii) Purification of Plutonium Streams. In the flowsheet being studied for the volatility processing of oxide fuels, separation of plutonium is achieved by fluorination of PuF_4 to PuF_6 with fluorine. This PuF_6 product stream will contain, in addition to plutonium, uranium, and neptunium, certain fission products which form volatile fluorides, and therefore the product stream will require further processing for the production of a purified plutonium product. Among the fission products which may accompany the plutonium are antimony, niobium, ruthenium, molybdenum, tellurium, and technetium. A laboratory-scale experimental program is under way to explore several processing schemes for purification of PuF_6 process streams.

One method under study involves the selective reduction of PuF_6 to PuF_4 with BrF_3 . The procedure involves contacting the mixture of PuF_6 , actinide hexafluorides, and fission product fluorides with BrF_3 for one hour at 80°C . The reaction vessel is then evacuated to volatilize the fission products and bromine fluorides as the temperature of the vessel is increased to 150°C . Remaining traces of uranium, neptunium, and fission products are removed from the PuF_4 product by contact with BrF_5 gas at 300°C . An experiment has been performed using this procedure; the results will be reported when they become available.

An alternative technique for purifying the PuF_6 stream involves sorption of PuF_6 on NaF at 100°C . Plutonium hexafluoride forms a stable complex with NaF (presumably $3\text{NaF} \cdot \text{PuF}_4$). Under the sorption conditions, UF_6 , NpF_6 , technetium, and small amounts of ruthenium, niobium, and antimony are also expected to be sorbed. Removal of uranium, neptunium, and technetium from the NaF bed is achieved by passing fluorine through the bed at 400°C . The resulting NaF bed, containing plutonium, is leached with anhydrous HF at room temperature. Sodium fluoride is soluble in liquid HF (30 g NaF /100 g HF at 11°C). Plutonium tetrafluoride, which is insoluble, appears as a precipitate and is recovered by filtration of the solution.

Experiments are also being performed to determine the extent of separation that can be achieved by sublimation of a mixture of UF_6 (as a stand-in for PuF_6) and SbF_5 , NbF_5 , and RuF_5 . Uranium hexafluoride mixed with the fluorides was sublimed in a stream of nitrogen carrier gas from a vessel at 0°C to a trap maintained at -78°C . The separation from ruthenium was less than that expected from knowledge of the relative volatilities of the fluorides (assuming that the volatile compound is RuF_5), while the separation from niobium and antimony was greater than expected. The decontamination factors from these compounds ranged from 50 to 700. Further experiments are planned to define this behavior more exactly.

(iii) Neptunium Fluoride Chemistry. Laboratory-scale studies are being carried out to determine the behavior of neptunium when oxidized nuclear fuel is reacted with BrF_5 . It has been determined that the reaction between NpF_4 and BrF_5 produces volatile NpF_6 (see Progress Report for October 1966, ANL-7267, p. 67). This study has been extended to measure the rates of fluorination of NpF_4 with BrF_3 and with BrF_5 at 350°C . The fluorinating agents, diluted with nitrogen (6 to 13 v/o BrF_3 ; 33 to 35 v/o BrF_5), were passed over NpF_4 in a boat reactor. The NpF_4 - BrF_3 reaction proceeds more rapidly than the NpF_4 - BrF_5 reaction, and, in this respect, the behavior of neptunium is similar to that of uranium (see Progress Report for March 1966, ANL-7193, pp. 75-76). The rate constants obtained were 0.0051 min^{-1} for the BrF_3 reaction and 0.0013 min^{-1} for the BrF_5 reaction.

Previous experiments have shown that NpF_6 reacts readily with bromine at 30°C to produce solid NpF_4 and BrF_3 (see Progress Report for November 1966, ANL-7279, p. 69). Since UF_6 is not reduced by bromine, the selective reduction of NpF_6 to NpF_4 offers an attractive technique for separating uranium from neptunium in NpF_6 - UF_6 mixtures. A series of experiments was performed to evaluate this reaction as a separation method. Mixtures of UF_6 and NpF_6 were contacted with excess bromine at 30°C . The results indicated that a satisfactory separation of uranium from neptunium can be achieved, but the reaction proceeds slowly at 30°C . Further tests at higher temperatures ($\sim 80^\circ\text{C}$) are planned.

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

(i) Engineering-scale Alpha Facility. The major objectives of experiments under way in the engineering-scale alpha facility are to demonstrate handling of PuF_6 in engineering-scale equipment and to determine the feasibility of fluoride volatility flowsheets. Recent experiments previously reported have involved the fluorination of PuF_4 in a fluidized bed of alumina to simulate the plutonium recovery step of the interhalogen flowsheet (see Fig. 10 in Progress Report for December 1966, ANL-7286, p. 53). Fission product compounds were not present in these runs. A recent experiment, Run Pu-15, was performed using feed material generated in experiments carried out in the laboratory-scale fluid-bed facility. The fluid bed remaining after the uranium had been recovered by fluorination of oxidized UO_2 - PuO_2 -fission product pellets was charged to the engineering-scale fluorinator. The total charge consisted of 1100 g of material from the laboratory-scale facility and 5500 g of alumina bed material which was the final bed after Run Pu-14 in the engineering-scale facility (see ANL-7286, pp. 54-55). Fluorination with 80 v/o fluorine in nitrogen was carried out as follows: 10 hr as the bed temperature was increased from 300 to 550°C , 5 hr at 550°C , followed by an additional clean-up period of 12.2 hr at 550°C . Preliminary analyses of the final dumped bed showed a plutonium concentration

of 0.017 w/o. This value is somewhat higher than the value (0.011 w/o) obtained in Run Pu-14 carried out under similar conditions. Both values represent acceptably low plutonium losses.

A complete plutonium material balance has been calculated for Runs Pu-6 to Pu-13 during which approximately 500 g of plutonium was processed. These runs involved either the fluorination of PuF_4 powder with fluorine or the transfer of PuF_6 from one process vessel to another. Analyses of all process streams and samples showed an accountability of 98% of the plutonium.

Initial experiments in the alpha facility with plutonium-containing materials involved the direct fluorination of UO_2 - PuO_2 -fission product pellets with fluorine to form UF_6 and PuF_6 . The mixed hexafluorides served as the feed material for thermal decomposition experiments. In these experiments, PuF_6 was reduced to solid PuF_4 while the UF_6 passed unaffected through the 300°C-thermal-decomposition reactor.

Chemical analyses of interim and product streams have provided preliminary information on the behavior of molybdenum (present in the fuel at fission product concentration) in these process steps. The bulk of the molybdenum was volatilized during the fluorination and collected with the mixed hexafluoride in the condenser. Very little molybdenum was found in the fluid bed after fluorination. In the subsequent step, most of the molybdenum was found in the UF_6 product stream from the thermal-decomposition experiments. Apparently, the molybdenum compound passed unaffected through the 300°C-thermal-decomposition reactor. Approximately 20% of the molybdenum remained in the mixed hexafluoride condenser after the fluorination step and was not fed to the thermal-decomposition step. This molybdenum was recovered during the clean-up fluorination with fluorine at elevated temperatures (300°C). This finding suggests two possible modes of behavior: (1) a fraction of the volatile molybdenum fluoride was reduced to a nonvolatile form in the condenser and this material was then refluorinated to a volatile compound by fluorine at high temperatures, and (2) a high-boiling compound, perhaps an oxyfluoride or MoF_5 , was formed during the fluorination step and was volatilized only when the condenser was heated to 300°C during the clean-up fluorination.

(ii) Process Development Studies for Uranium Dioxide Fuels. Engineering-scale studies are being performed in a 3-in.-dia fluid-bed reactor facility to determine the effects of important process variables on the fluorination of UO_2 fuels with BrF_5 . The current program involves a series of eight statistically designed experiments to measure the effects of six independent variables on UF_6 production rate, BrF_5 utilization, off-gas composition, and uranium removal. The variables under study are (1) bed temperature, (2) BrF_5 concentration, (3) fluidization gas velocity, (4) quantity of UO_2 charge, (5) particle size of the inert alumina bed material, and

(6) the presence or absence of inactive fission products. The operating conditions for the oxidation step (temperature, oxygen concentration, and gas velocity) will be constant for all eight experiments. These runs are under way and the results will be reported as they become available.

IV. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

1. General

Construction Package No. 1 has been approved by the Laboratory and the AEC. Drawings and specifications have been reproduced and transmitted to the Chicago Operations Office.

The architectural layout drawings have been reviewed, and the comments have been forwarded to Burns & Roe. A meeting was held on January 20, 1967 to complete comments on Title I electrical systems.

At a meeting of personnel from ANL and Burns & Roe with H. O. Ireland, a soils consultant, it was decided that the reactor containment building will have a mat foundation.

The consultant responsible for seismic evaluations has started Phase I work, which is the development of geological and seismic-exposure information. At a meeting with the consultant on January 19, 1967, information on soils data, safety analysis, and preliminary drawings of AARR were transmitted to him for his review.

The final design criteria for the instrumentation and control systems have been transmitted to Burns & Roe, who have been directed to begin the detailed design.

The revised Preliminary Safety Analysis Report (PSAR), Supplement No. 2, has been completed and was transmitted to the AEC on January 20, 1967. A review of Supplement 1 was held with DRL on January 25, 1967.

2. Strain Energy of the AARR Pressure Vessel

The strain energy of the AARR pressure vessel has been recalculated. In the previous computation (see Progress Report for November 1966, ANL-7279, pp. 84-85) it was assumed that the vessel wall was 3-1/2 in. thick uniformly in the main shell. A recent study by Combustion Engineering indicates that it is possible to reduce the main shell thickness, outside the belt region, to 2-7/8 in. It is, therefore, necessary to reinvestigate the strain energy of the pressure vessel and its containment potential. For the purpose of analysis, we assume that the vessel wall is 2-7/8 in. thick; the extra wall thickness in the belt region is considered as the reinforcing material. It is believed that the extra thickness provided in the belt region will preclude premature failure at the nozzle penetrations and that the vessel will undergo a gross ductile deformation without rupture up to the

ultimate strain of the material. The strain energy of the vessel at the ultimate strain is about 340 MW-sec, and that for one-third of the ultimate strain is about 66 MW-sec.

3. Effect of a Nuclear Excursion Followed by a Metal-Water Reaction

A model has been developed to calculate the pressure transient in a nuclear excursion followed by a metal-water reaction. The assumptions of the model are:

- (1) The effect of the nuclear excursion is to heat the aluminum in the core instantaneously.
- (2) When the aluminum reaches 1750°C, it ignites in steam.¹⁷
- (3) After the ignition has been initiated, the return of the water, which was forced outward by the formation of vapor bubbles, collapses the bubbles and causes the core to fragment.
- (4) The core disassembles uniformly into spherical particles giving a sixfold increase in the aluminum area for further reaction with steam.

If the reactivity excursion model in the preliminary safety analysis of the AARR is taken as the maximum credible accident, the energy released from the nuclear excursion is about 163 MW-sec, and that from metal-water reaction is about 1300 MW-sec.¹⁸ By use of the first law of thermodynamics, the heat is available for the formation of the steam from water initially at 340°K (neglecting the original volume of the water used to form steam and the work of compression of the water) is

$$Q + n_a(H_{2050}^{\circ} - H_T^{\circ}) Al_2O_3 + n_h(H_{2050}^{\circ} - H_T^{\circ}) H_2 = n_s \Delta H_v + n_s(H_T^{\circ} - H_{340}^{\circ}) H_2O - Pv + W, \quad (1)$$

where

n_a , n_h , n_s = number of moles of Al_2O_3 , hydrogen, and steam, respectively;

Q = the sum of the heat released from the nuclear excursion and the metal-water reaction;

¹⁷Survey of Metal-Water Reaction Data Relevant to AARR, Appendix B, PSAR (AARR).

¹⁸Hypothetical Rapid Insertions of Reactivity to Super-prompt Criticality, Section VIII-B, PSAR (AARR).

$H_{2050}^{\circ} - H_T^{\circ}$ = enthalpy decrement at temperature $T^{\circ}\text{K}$ relative to 2050°K ;

$H_T^{\circ} - H_{340}^{\circ}$ = enthalpy increment at temperature $T^{\circ}\text{K}$ relative to 340°K ;

ΔH_v = the heat of vaporization of water;

P, v = the final pressure and volume of the steam and hydrogen mixture, respectively;

W = work done on the primary system.

The equation of state of the steam and hydrogen mixture is

$$Pv = (Z_s n_s + Z_h n_h) RT, \quad (2)$$

where

Z_s, Z_h = the compressibility factors of steam and hydrogen, respectively.

The equation of state of the pressure vessel (see Progress Report for November 1966, ANL-7279, pp. 84-85) is

$$v_v = CP^2, \quad (3)$$

where v_v is the incremental volume of the deformed vessel and C is a constant. If the pressure vessel is made of Inconel-600 and the main shell is 2-7/8 in. thick, this constant is 8.47×10^{-10} . Accordingly, the strain energy, E_s , of the vessel is given by

$$\int_0^P P dv_v + 0.667 P v_v. \quad (4)$$

Since the energy released in the nuclear excursion and metal-water reaction is very rapid, for the sake of conservatism no credit is taken for the strain work on the piping or for the flow through the pressure relief system. Therefore,

$$W = E_s = 0.667 P v_v \quad (5)$$

and

$$v_v = v. \quad (6)$$

By the use of thermodynamic data¹⁹ and compressibility factors,²⁰ these equations were solved for various values of temperature. The values of strain energy as a function of the final steam temperature are plotted in Fig. 25. The maximum possible strain energy imparted to the pressure vessel from the nuclear excursion and a rapid metal-water reaction

(1300 MW-sec) is about 190 MW-sec, which is less than the minimum amount of energy (340 MW-sec) required to rupture the vessel.

Fig. 25. Strain Energy in AARR Reactor Vessel

A computer code will be programmed so that the full range of the potential release of chemical energy, for example, energy released from the 4-in. inactive fuel plate, side plates, etc.,¹⁷ can be studied.

4. Development of Experimental Facilities

A partial mockup of the AARR reactor internal structure has been completed. It approximately simulates the proper dimensions and relative elevations of such items as the upper vessel head, beryllium reflector, and control-rod guide shroud. This mockup is part of the development program required for the experimental irradiation facilities.

Internal components and handling-tool design problems are now being studied for the hydraulic-rabbit facilities to be located in the permanent beryllium reflector. The rabbit sample tubes for these facilities are required to penetrate unaligned openings in the upper vessel head and openings in the permanent reflector.

Steel tubing, 15 ft long and with proper bends and diameters to simulate both the 1/2-in.- and 1-in.-dia rabbit facilities, were successfully inserted into the intended positions.

B. Direct Conversion

1. Liquid-metal Linear Generator

a. Effects of Interfacial Transport Processes on Film Condensation.

A study of the entrance effects resulting from momentum exchange at a gas-liquid interface is underway. Visual observations in the Lucite test section

¹⁹JANAF Thermochemical Tables, Dow Chemical Co., Midland, Mich. (1965).

²⁰Fano, L., et al., Compressibility Factors, Density, Specific Heat, Enthalpy, Free-energy Function, Viscosity, and Thermal Conductivity of Steam, NACA TN 3273 (Aug 1956).

have confirmed the presence of two- and three-dimensional disturbances at the gas-liquid interface for the values of the ratio of the gas to liquid flow rates of interest. Due to the presence of these interfacial waves, various direct methods of obtaining information about the liquid film have been tried.

Two methods of measuring the mean velocity distribution in the liquid film have been considered: (i) the hydrogen-bubble technique, and (ii) the hot-wire constant-temperature anemometer. The hydrogen-bubble technique is not practical because the condensation makes necessary the use of excessively high voltages to produce the bubbles in high-purity water. The constant-temperature hot-wire method proved to be a promising tool for making local measurements in thin, high-speed liquid films. However, it was found that the time dependence of the velocity field resulting from the presence of interfacial waves may negate its use in condensing flows. This follows from the fact that velocity is presumed to be a function of heat flux from the sensor, i.e., the temperature difference between the sensor and fluid is assumed constant. In the case of condensation on the film, it is expected that the temperature in the film will exhibit a spatial dependence. Superimposed on the spatial dependence will be time-dependent fluctuations, estimated on the basis of wave frequency from 10 to 200 cps. This necessitates correlating velocity with both instantaneous heat flux and temperature difference. These measurements may not be possible experimentally if the magnitude of the temperature fluctuations are large. By using the sensor as a resistance thermometer in the condensing film, these fluctuations will be measured.

Average thickness of the liquid film, as a function of the distance from the injector, has been measured using the conductance-probe technique. Data have been taken for values of the ratio of gas to liquid flow rates from 0.0 to 0.426, including both laminar and turbulent flows. The results thus far indicate the validity of some of the assumptions made in developing the theoretical model describing the flow.

Since the measurements taken with the conductance probes have indicated a strong dependence of the electrical conductivity of the liquid, the needle contact probe will not be used to measure film thickness in the condensing flow. However, the conductance probes will be used to study the correlation between the average film thickness and wave amplitude to provide a method of calibrating the needle contact probe.

b. Condensing Injector Studies. A program describing the one-dimensional equations of motion and energy for the condensing flow in the injector mixing section has been written. As soon as debugging has been completed, the results will be reported. Also, a vapor desuperheating system was installed in the Freon Loop during the past month.

V. NUCLEAR SAFETY

A. Accident Analysis

1. Heat Transfer

a. Steady State. The Steady State Heat Transfer Module (HTSS) has been checked out. It is capable of obtaining the desired temperature profiles (radial and axial) in a fuel rod. As reported earlier (see Progress Report for November 1966, ANL-7229, pp. 46-47) HTSS takes into account the temperature dependence of the thermal properties, thermal expansion (unrestrained), and melting of the fuel.

The temperature drop in the gas bond is a strong function of the bond thickness, and there is a 10% volume change during melting of the UO_2 fuel. Consequently, it was found feasible to have partial melting in some radial nodes of the fuel in order to achieve compatibility between the bond thickness, the temperature drop in the bond, and the temperature profile in the fuel.

Compatibility can be achieved by considering only full or no melting of any node provided one uses a large number of radial nodes. However, the added computing time becomes enormous compared to the extra computing time necessary to allow partial melting.²¹

b. Transient State. The Transient State Heat Transfer Module (HTTS) has been checked out by solving several sample problems.

In order to test compatibility between the HTSS and the HTTS modules, a dynamic equilibrium test is performed. This consists of computing a steady state using the HTSS module, taking the steady state as the initial condition for the HTTS module, and observing the drift in temperatures. It was found that the drift was less than 1° in 100 time steps at any radial node.

The stability of the modified Crank-Nicolson scheme used in the HTTS module is also tested. It is well known that when the quantity

$$\epsilon = \frac{K}{c\rho} \frac{\Delta t}{(\Delta r)^2},$$

²¹ Greebler, P., et al., GEAP-4090 (Oct 1962).

where

K is the thermal conductivity;

c is the specific heat;

ρ is the density;

Δt is the time increment;

Δr is the space increment,

is greater than $1/2$, an explicit differencing scheme becomes unstable. Since the method used here is implicit in nature, no instability was observed even for $\epsilon = 2$.

Further, the method is also found to have excellent accuracy for a value of ϵ as high as 2. The transient temperature profiles (with centerline temperatures around 3000°K) for two cases, namely, for $\epsilon = 2$ and $\epsilon = 0.1$, differed only by about 5° .

2. Linking of the Heat Transfer and Two-phase Coolant-flow Modules

Linking of the modules is essentially complete. The linking is accomplished in an overlay structure.

There are four main computing modules, namely,

- a) Steady State Heat Transfer in the fuel rod (HTSS),
- b) Steady State Coolant Flow (COOLSS),
- c) Transient State Heat Transfer in the fuel rod (HTTS),
- d) Transient State Coolant Flow (COOLTS)

All transients are considered to start from a steady state. The sequence of calculations is given below:

1. COOLSS calculates steady-state temperature, pressure, and the thermal properties of the coolant.
2. HTSS computes the steady-state temperature profiles, fraction melted, and expansions of the fuel rod.
3. HTTS calculates changes in fuel-rod-temperature profile for the first half time step (assuming no change in the coolant for the first half time step).
4. COOLTS computes changes in coolant-flow characteristics and properties for the first time step.
5. HTTS computes fuel-rod-temperature profile for first time step (taking into account changes in the coolant).
6. Steps 4 and 5 are repeated using suitable values for time steps to cover the necessary period of the transient.

The heat-transfer modules communicate with the coolant modules via the heat flux into the coolant, and the coolant modules in turn communicate with the heat-transfer modules via the coolant temperatures, the heat transfer coefficient (clad to coolant) and the time step in use.²²

The overlay tree structure is presented in Fig. 26.

Fig. 26. Overlay Tree Structure

The total load module length is 119,620 bytes. This includes the root segment called MAIN; the longest module in the region ALPHA, namely, HTTS; the length of region BETA, namely, the length of $\bar{q}_j$ and the length of $K(T)$; and the entry tables used to call different modules.

Specific transient problems are being solved to check out the above linkage.

B. Coolant Dynamics

1. Coolant (Water) Expulsion Studies

This experiment has been completed and a topical report is being prepared.

2. Sodium Expulsion Experiment

In this experiment, it is planned to investigate the mechanism of coolant expulsion in a simulated reactor environment. Thus, the experiment includes measurement of the void distribution, expulsion velocities, pressure transients, and liquid superheats during coolant expulsion.

²² The module COOLTS controls the time step.

Final design details have been frozen and fabrication is continuing. The rectangular-cross-section heated piece with an adapter has been made. Tests are underway to determine whether or not the proposed method of introducing thermocouples into the sodium is satisfactory in regard to the anticipated temperature and pressure transients.

3. Superheat Experiment

Approximately 300 test runs of the liquid-superheat experiment have been completed. The superheat test runs to date include pressures ranging up to 18 psia, heat fluxes ranging up to 0.5×10^5 Btu/hr-ft², liquid superheats required to initiate boiling ranging up to approximately 200 psia (ranging up to 400°F for pressures below 1 psia), steady-state superheats ranging from 5 to 30°F, and liquid-sodium temperatures ranging up to 1800°F. Pressure pulses ranging up to 200 psia have occurred both during the initiation of boiling and during steady-state boiling.

4. Studies of Critical Flow

During the investigation of steam-water critical flow, a new method has been developed which allows a more accurate determination of the critical pressure and appears to eliminate experimental discrepancies encountered in previous investigations. This method reveals that two-phase critical flow is independent of the back pressure throughout the quality region, as is single-phase critical flow. In view of this new technique, additional exit voids as well as axial void-fraction profiles are being measured. Hopefully, these new measurements, along with developed analytic tools (see Progress Report for December 1966, ANL-7286, pp. 66-68), should result in quantitative estimates of the existing metastability and slip.

Construction of the test facilities for identical sodium and water critical-flow experiments has been completed. The sodium loop has been operated successfully up to 1450°F, and several "blow-downs" of sodium have been carried out to test the flow-measuring device and pressure transducers. Excellent performance of these components has been demonstrated. The first two-phase sodium critical-flow data point (critical flow rate, critical pressure, equilibrium quality, and stagnation properties) has been obtained and is being analyzed.

5. Electron-bombardment Heater Test Experiments

These experiments consist of testing various electron-bombardment heaters which will simulate in-pile experiments pertinent in reactor safety considerations. A large evacuated vessel will be used for examining the operation of the various electron-bombardment heaters. The tests involve the fabrication and testing of various heater configurations. Both single- and multi-pin heaters will be built and tested.

The test apparatus is in the design stage and working drawings should soon be completed.

6. Convective Instability

This experiment is being carried out to obtain fundamental data on transient-free convection heat transfer in narrow-channel liquid-metal systems. The data obtained will enable analysis of the initial stages of heat transfer in fast-reactor safety analysis. Concurrent with the experiment, a theoretical analysis is being made to determine the effect of system parameters on the heat transfer. A wide range of parameters will be varied in both the experimental and the theoretical studies.

The two-dimensional boundary-layer equations for transient-free convection have been reduced from three independent variables (x, y, t) to two independent variables (x, t) . Two techniques are being applied to reduce the problem to a solution of ordinary, nonlinear, singular equations. These techniques are the method of lines and the method of integral relations. The method of integral relations has yielded the best results for the amount of effort expended for a solution, if the profiles are judiciously approximated by assuming that the velocity and boundary-layer thickness are products of a function of time and the steady-state solution in longitudinal space x . When a polynomial approximation is made, such as

$$u \sim \sum_{n=1}^N A_n(t) x^n \text{ and } \delta \sim \sum_{j=1}^M b_j(t) x^j,$$

a much larger number of approximations is required to obtain reasonable accuracy. The accuracy herein mentioned is deduced by a comparison with the time for steady state to be reached and the values reached at steady state with the exact values.

At the present time, no solutions have been obtained by the method of lines for the same problem, but in the near future the computer programs should be debugged and a comparison of the two solutions by the different methods will be made.

7. Component Dynamics

The study of the mathematical model of the parallel-plate coolant-structure system (see Progress Report for December 1966, ANL-7286, pp. 69-70) revealed a reasonably expected structural response to a rapid energy insertion. The computer program was run again to demonstrate the influence of the channel walls (as implied by their computed natural frequencies) on the peak pressures. The computer simulations gave the

dynamic response of coolant and structure to rapid transient heating in the form of a rectangular power burst. The parallel-plate structure was chosen to simulate a reactor coolant channel, viz., AARR's nominal channel (0.040 x 2.5 in.). The channel was assumed to be heated over a 10-in. length and closed at the bottom, thereby requiring the coolant to be expelled vertically upward.

One of the objectives of the study is to determine the influence of the bounding structure on pressure generation and coolant expulsion. Toward this end, various fuel-plate thicknesses were assumed (see Table XXXIII) and the corresponding system response was simulated on the digital computer. The influence of the flexibility of the bounding structure (characterized by the fundamental natural frequency of the channel wall) on the peak pressure and maximum channel-wall displacement is shown in Fig. 27 for a rectangular power burst of 100 kW and initially saturated water. Note that the structural response ($W_{\max} = 0.5$ in.) corresponding to the case of a 10-mil-thick channel wall ($f = 348$ cps) must be reconsidered in relation to the assumption of small deformations used in writing the equation of motion for the structure. It may be necessary to use large deformation theory, in which case the membrane effect of the plate would be included. Nevertheless, the pressure transient is correct for the given structural response.

TABLE XXXIII. Channel-wall Characteristics

Plate Thickness h (in.)	Fundamental Frequency f (kc)
0.01	0.348
0.02	0.697
0.03	1.04
0.04	1.39
∞ (rigid)	∞

A motivation for the study was the anticipation that for certain input and structural vibration periods, the coupled coolant-structure system might be such that the structure would experience oscillations which would be reflected in the pressure transient. Such a condition did exist for the case of 10-mil plates. However, the oscillations were of low amplitude.

The pressure-pulse model being employed in the system analyses is a very simplified model. To indicate the degree of accuracy with which it predicts resulting pressure pulses, a problem was run assuming a rigid container and using the same channel cross section ($A = 0.344$ in.²) as used

in experimental water-expulsion tests of Singer.²³ In comparing the predicted response with the corresponding experimental response for nearly saturated water, the general shape of the pulse correlated reasonably well: peak pressure predicted was 2.49 atm, compared to 2.99 atm experimental.

Fig. 27. Structural Influence on the System Response to a Rectangular Power Burst of 100 kW ($T = 80$ msec)

In view of this correlation and the fact that the mathematical model contains coefficients (relating to heat transfer) which can be adjusted based on experimental observations, the present pressure-pulse model can be used to obtain meaningful results from coolant-structure system analyses for certain heat-transfer thermodynamic conditions, particularly when almost all the input energy goes into latent heat of vaporization and the resulting vapor does not recondense, but is continually formed. This, for example, is the case with saturated or nearly saturated water.

Future study will investigate other channel geometries and the effects on coupled-system response of parameters, such as liquid-column height and ambient pressure.

²³ Singer, R. M., Experiments on Water Expulsion Following Rapid Transient Heating, Trans. Am. Nucl. Soc. (1966 Winter Annual Meeting, Oct. 30-Nov. 3, 1966, Pittsburgh).

C. Fuel Meltdown Studies with TREAT

1. Behavior of Irradiated Oxide Fast Reactor Fuel Pin in TREAT Excursions.

A single, gas-bonded, half-length EBR-II-like oxide fast reactor fuel pin, previously irradiated to an estimated burnup of 6.23 a/o, has been tested in a photographic TREAT experiment. This test was an extension of those reported previously (see Progress Report for July 1966, ANL-7245, p. 60) for identical oxide pins irradiated to estimated burnups ranging from 0.70 to 3.45 a/o. The nominal 6% burnup pin was given a transient of 207 MW-sec and 0.5-sec duration, a TREAT excursion calculated to approximate the most severe transient of the four previous pins. That transient gave 1141 J/g energy input to a 2.95-a/o burnup pin, raised the maximum fuel temperature to the melting point, and produced a maximum temperature of the fuel surface at the time the center reached the melting point calculated to be 2220°C. The series of tests on all 5 pins was designated RP TREAT Series XLVIII.

Post-transient inspection of the four lower-burnup pins may be briefly summarized as follows:

1. Moderate longitudinal cracking occurred along the fuel outer surface.
2. A slight degree of fission gas bubble formation was noted.
3. There was apparently some localized heating and reaction between fuel and cladding.
4. Microscopic examination supported by an electron-microprobe analysis showed the fuel to be nonhomogeneous.

Each fuel sample consisted of a stack of oxide cylinders contained in a Type 304 stainless steel tube of 0.396-cm inner diameter and a 0.023-cm wall and in an argon gas atmosphere. The 6% burnup fuel sample was 13.970 cm long and had a diameter of 0.381 cm before steady-state irradiation in MTR. The fuel was initially homogeneous, 93% dense UO_2 with an O/U ratio of 2.001 and 13% enriched. However, studies of the irradiated fuel have indicated that the fuel was nonhomogeneous after irradiation. The cladding temperature was monitored at 2.54 cm above and 2.54 cm below the fuel centerline by two Chromel-Alumelfast-response thermocouples spotwelded to the cladding. The pin, No. 16, contained in capsule 5 of RP Series XLVIII, underwent the following test conditions (a preliminary transient was run as a check):

Shot No.	Total Integral Power (MW-sec)	Max Temp of Clad Surface, Lower Thermocouple (°C)	Max Temp of Clad Surface, Upper Thermocouple (°C)
1	74.3	889	850
2	207.0	1300 ^a	1300 ^a

^aThe cladding melted.

The fuel was originally composed of five cylinders stacked atop one another inside the stainless steel tube. One second after heating began, the cladding had melted and was freely flowing down the fuel, which had solidly fused into one rod. The ceramic UO₂ fuel rod became plastic and bowed 0.4 cm without separating. During the post-transient inspection it was determined that the fused joint was stronger than the parent material. Inspection of the sample remains in the ANL Hot Laboratory yielded the following information:

1. Moderate to very large cracking extended from the fuel central void to its outer surface.
2. Extensive fission gas bubble formation occurred.
3. There was definitely localized heating between the pellets, causing them to fuse together, and between fuel and cladding.
4. Extensive bubbling occurred over all the fuel surface, varying in intensity and degree.

2. Unbonded EBR-II Pin in Flowing Sodium

A transient experiment has been proposed in which an unbonded (actually, an argon-bonded) EBR-II fuel pin is run in a Mark-I integral TREAT loop to obtain experimental information for use in guiding analyses of a possible cladding failure of a fuel pin in the EBR-II core. Conditions specified for the experiment include:

a. The TREAT transient is to be a "flat top" excursion with power sufficient to produce a nominal constant power density in the 6% enriched sample of 40 cal/sec per gram of fuel (i.e., 168 W/g), a starting sodium temperature of 370-400°C, and sodium flow of 4 m/sec through the test section. Also specified were tantalum wrappers to be placed around the test section to shape the axial power distribution.

b. One Mark-I integral sodium loop was available for the experiment. The loop has been loaded and checked out, and is ready for shipment to TREAT when review of the proposed experiment is finished. In order to simulate flow conditions in the core, the sample pin was surrounded by a ring of six dummy pins consisting of argon-filled, EBR-II-pin cladding tubes.

On the basis of the kinetics analysis made when the TREAT facility was used as a test reactor, and corrected for the specific core loading to be used with the loop, it has been calculated that the specified sample power level will be equal to the peak sample power generated by a temperature-limited TREAT transient initiated by 0.85% Δk and that this power density can be sustained by a slow ramp addition of 0.18% $\Delta k/\text{sec}$.

One quantity needed for analysis of the experiment, which is believed to be not capable of being calculated on the basis of current knowledge, is the effective thermal conductance of the gas gap between fuel and cladding. A value of 0.28 W/cm²-°C has been obtained for this conductance by analyzing experimental data from the fast-response cladding thermocouples used in a series of earlier experiments conducted to study the cladding penetration of both the unbonded and bonded EBR-II pins.²⁴ Those tests were run with two pins, one bonded and one unbonded, in each capsule, contained in an inert gas atmosphere to permit cladding temperature and penetration time instrumentation.

D. Materials Behavior and Energy Transfer

1. Sodium-Air Reactions

Safety analyses for sodium-cooled reactors require knowledge of the temperatures and pressures generated during sodium combustion in air. Combustion of both pools and sprays have to be considered. It was previously pointed out²⁵ that, regardless of the burning mechanism, the maximum pressures and temperatures are limited by the thermodynamic properties of the system. Theoretical calculations have shown (see Progress Report for June 1966, ANL-7230, p. 70) that the maximum pressure of 9.1 atm is generated at a sodium-to-oxygen ratio of 5 in dry air. The peak values in air with 100% humidity at 25°C are predicted to be lower.

An experimental effort is now being made to study the reaction of sodium sprays in air. Pressure-generation rates, peak pressures and temperatures, and the nature and postreaction behavior of the reaction products will be studied as functions of the initial conditions.

The apparatus consists of a pneumatically operated piston spray injector that will spray 15 g of molten sodium into a reaction chamber of known volume in a period of about 20 msec. The initial temperatures of the sodium, the water content of the air in the reaction chamber, and the sodium-to-oxygen ratio will be varied to determine their relative importance.

²⁴Walter, C. M., and Dickerman, C. E., Nucl. Sci. Eng. 18, 518 (1964).

²⁵Chemical Engineering Division Semiannual Report, January-June 1965, ANL-7055, p. 208.

A glass chamber is being used to permit the evaluation of the sodium spray in the absence of reaction. The chamber will be filled with cold nitrogen to quench the sodium spray rapidly. This chamber will be replaced with a steel vessel fitted with view ports for experiments made in air.

The spray-injector head forms the bottom of the reaction chamber. Under the injector is located the pneumatic drive apparatus. The top flange contains the two pressure transducers (one of high- and one of low-speed response) and the entrance port for several thermocouples. The thermocouples will be located at several points inside the chamber in order to ascertain both the average temperature and the peak temperatures obtained. The transducers are protected from direct sodium impingement by a baffle attached to the top flange.

The sodium injector is shown in Fig. 28. The lower piston is driven by nitrogen introduced through a large orifice solenoid valve from a ballast

Fig. 28. Cross Section of Spray Injector for Sodium-Air Experiments

tank. This piston pushes the upper piston to drive the molten sodium upward into the reaction chamber through a discharge disc containing suitable orifices. Different orifice sizes and geometries are obtainable by changing the discharge disc.

Loading of the sodium injector is done in a helium-filled glovebox. The injector is assembled with both pistons in the lowest positions and the assembly heated to a temperature above the melting point of the sodium. Molten sodium is poured in, and the assembly sealed by means of a copper gasket and a copper rupture disc. The assembly is then cooled and attached to the reaction chamber. At this point, the injector assembly is reheated to the desired initial conditions. The chamber is filled with the desired atmosphere and the sodium injected. After the reaction is over, the residue may be analyzed.

Fig. 29. Spray Pattern from Spray Injector Using Water as a Test Fluid (0.030-in. orifices)

Preliminary tests of the injector system using water as the driven fluid have been made. Figure 29 shows a typical spray pattern obtained from a circular pattern of 0.030-in. orifices.

E. Operations TREAT

Two samples of prototype Fast Ceramic Reactor fuel material (C4C and C4D) were irradiated for the General Electric Company. Sample C4D will be stored at TREAT until data from this irradiation are analyzed and specifications for the next transient are received from the General Electric Company.

A Zircaloy-clad UO_2 pin (CEN-225-T) was irradiated in room-temperature water in a transparent capsule to obtain additional photographic data on the behavior of this type of fuel during and following a transient which uses extensive metal-water reaction.

Metal-water reaction sample CEN-227-T was subjected to two transients in transparent capsules to determine the energy input required to cause cladding failure and/or measurable metal-water reaction. This sample of vibratory-compacted UO_2 fuel clad in Zircaloy did not melt during these transients, but small cracks were found near the top of the pin after it was removed from the test capsule. Samples of the blanket gas from the capsule have been submitted for mass-spectrometric analysis to determine the extent of metal-water reaction.

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