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Argonne National Laboratory

REACTOR DEVELOPMENT PROGRAM

PROGRESS REPORT

December 1964

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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

December 1964

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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 over-all 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:

June 1964	ANL-6912
July 1964	ANL-6923
August 1964	ANL-6936
September 1964	ANL-6944
October 1964	ANL-6965
November 1964	ANL-6977

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I. LIQUID-METAL-COOLED REACTORS

A. General Fast Reactor Physics

1. ZPR-III

Experiments with Assembly 45A were completed.

The experiments performed, aside from Doppler investigations, were determinations of central fission ratios, central worths of plutonium and depleted uranium, axial reaction rate traverses of Pu^{239} and U^{238} , and central and full-length axial sodium worths relative to void. The central worth of sodium was negative, and the full-length axial value was positive. Values for the reactivity await the results of the analysis of rod calibration experiments. The remainder of the data is being analyzed.

The final Doppler measurements included: (1) the Doppler effect of 100% plutonium oxide elements when surrounded by a 1/2-in.-thick layer of natural boron carbide; (2) the Doppler effect of the same elements when surrounded by normal core material instead of a buffer region; (3) a repeat of the experiment with the plutonium sample surrounded by the buffer region; (4) a measurement of a fuel-type composition, namely, a 7:1 natural uranium:plutonium sample; and (5) the Doppler effect of an enriched U^{235} sample. The results of the experiments are being analyzed.

Assembly 45 was disassembled, and the inert blanket materials for the FARET mockup are being loaded into the ZPR-III matrices.

2. ZPPR

a. Description of Facility. The Zero Power Plutonium Reactor (ZPPR) Facility is to be constructed in the east area of the NRTS, approximately 1000 ft southeast of the EBR-II reactor building. ZPPR is a split-half critical machine for studying the physics of large plutonium-fueled fast reactors. The assembly machine is sized to permit eventual mockup of engineering cores as large as 6000 liters which contain as much as 3000 kg of plutonium.

Reactor containment is provided by a 50-ft, right circular concrete cylinder which has a gravel roof supported by steel cables and wire mesh. Blast doors and seal doors in the access corridors, as well as seal valves and seal connections, complete the containment. Should an emergency occur for which particulate containment would be necessary, the cell would be vented through the gravel roof, which would act as a filter.

Other areas of the facility are the workroom, fuel-storage vault, and the support wing. Included in the support wing is a room which will house the AFSR, presently located in the west area.

The details of the facility are now being developed. The architect-engineering firm selected for the construction project is the Mason & Hanger-Silas Mason Co., Inc. Title I design started in October and is presently estimated to be finished in February.

b. Reactor Cooling System. The use of large amounts of plutonium presents a cooling problem in the mockup cores, owing primarily to alpha emission from plutonium. As the Pu^{240} content increases, the rate of heat generation also increases. Based on a plutonium fuel containing 8 to 10 w/o Pu^{240} , the heat-generation rate of 3000 kg of plutonium is approximately 6000 W or 2 W/kg.

Experience with plutonium loadings in ZPR-III showed that a cooling system that maintains constant reactor temperatures would be most advantageous. Long equilibration times were required due to fluctuating ZPR-III reactor temperatures when the halves are separated and then closed again.

High coolant flows and low temperature gradients will reduce temperature fluctuation in the reactor. However, for physics reasons, it is also desirable to keep voids at a minimum in the reactor.

Air from the reactor cell is the coolant and is pulled through the reactor core to remove heat. As a starting point, 6000 cfm at the 5000-ft elevation of the NRTS was selected as the coolant flow rate through a 6000-liter core containing 3000 kg of plutonium. The increase in air temperature as it passes through this zero-power subcritical core is estimated to be slightly less than 2°C .

The clearances above the loaded drawers presently in ZPR-III are minimal. To allow for a coolant passage through each matrix tube and above each drawer in ZPPR, it was estimated that the matrix tube vertical dimension should be increased 0.1 in. The pressure drop across the core and blanket obtained by the above flow rate (6000 cfm through 1470 matrix tubes) was estimated to be 7 in. H_2O .

To verify these calculations, so that the cooling blowers as well as the matrix tubes can be specified for the reactor, a mockup test apparatus was assembled. To obtain clearances, 0.10 in. was machined from some ZPR-III aluminum plates and 0.10 in. from the top edges of some ZPR-III core drawers. From a series of flow measurements pressure drops through coolant channels would be estimated. It appears that the entrance and exit losses as well as the internal expansion and contraction losses can be reasonably calculated through use of established friction coefficient tables. However, calculated friction losses due to air flow over the mockup materials appear to be approximately 60% too low. This can probably be attributed to the increased turbulence caused by irregular

surfaces of the mockup core materials. By increasing the friction factors approximately 60%, the calculated pressure drops would then agree with the measured values.

The coolant channels have introduced approximately 5 v/o void in the core. By reducing the size of the coolant channels, the voids in the core may also be reduced. However, the pressure drop then increases. Additional flow tests are planned to optimize the coolant channel size.

The heat transfer characteristics of the reactor fuel to the cooling system are not well defined since there are many contact heat transfer surfaces. Some preliminary tests are planned to characterize the temperature distribution and heat flow patterns.

c. Reactor Bed and Tables. The technical specifications for the reactor bed and tables are essentially complete. The complete package is expected to be ready to go out for bids in January 1965.

The bed will be approximately 25 ft long, and each table will be 14 ft wide and 9 ft long. The separation of the halves will be 7 ft. The nominal load capacity of the bed and tables assembly will be 180 tons.

3. ZPR-IX

Assembly No. 6, which will be used to study the effect of uniform distribution of small quantities of hydrogen and the effect of boron in the reflector, has been loaded to criticality. Each core drawer consists of four 1/16-in. columns of enriched uranium, seven 1/8-in. columns of tungsten, and seven 1/8-in. columns of 45% density aluminum. Each drawer contains 1.65 kg U^{235} . The core length is 58 cm. Criticality was achieved with a loading of 258 full drawers and 38 half drawers with a mass of 458 kg U^{235} . The core radius at critical was about 34 cm. The available excess reactivity with this amount of fuel is approximately 73 lh, and the edge fuel worth is approximately 21 lh/kg.

Control rod calibration, α or β/l measurements, fission ratios, and material replacement experiments are currently in progress.

B. Fast Reactor Systems and Concepts

1. 1000-MWe Metal-fueled Fast Breeder Reactor Study

The basic data for the reference design of a 1000 MWe metal-fueled fast reactor was presented in ANL-6977 "Reactor Development Program Progress Report, November 1964." The reactor study involves Argonne's total developmental efforts in this particular area including reactor physics, engineering, fuels processing and refabrication.

a. Physics. Current analyses are concerned with boron-containing control and shutdown rods. An estimate of the boron poison effect was obtained for the system with the beryllium region at the core-blanket interface. Two volume percent natural B_4C was homogeneously substituted for an equal volume of core sodium. The resulting reactivity change was -5.5% k. The corresponding incremental changes in internal and total breeding ratios were -0.07 and -0.10 , respectively. As the reactivity swing in a fuel cycle (see Progress Report for November 1964, ANL-6977, p. 14) is estimated to be 2 to 3% k, the time-average reactivity variation will be about one-half of these values, i.e., 1 to 1.5% k. The effect of the poison on the breeding will therefore be only about one-fourth of the previous increments (~ 0.02 and -0.025 , respectively).

The effects of the 2% poison on the sodium void and Doppler values have also been calculated. For the case of a uniform reduction of 40% in the core the presence of the poison reduces the negative reactivity effect by about 80 to 90 percent. Similarly, the negative reactivity U^{238} Doppler effect in the core is reduced by about 35 to 40 percent. By linear interpolation the estimate of 1% maximum poison required by the fuel-cycle swing of 2 to 3% k would halve the listed percent decreases for the sodium and Doppler effects.

b. Dynamic Behavior of Modules. The dynamic behavior of a single core module of the 1000-MWe system has been studied for two kinds of transient conditions important to considerations of safety: ramp insertions of reactivity and loss of coolant flow. Calculations were performed using RP 351 X, a code that permits computation of the temperature distribution as a function of time across an average fuel pin and across the hottest fuel pins under conditions of varying reactivity and, if desired, varying coolant velocity. The code includes feedback from Doppler effect and from expansion of fuel, clad, coolant, or structure.

Doppler coefficients appropriate for systems both with and without a beryllium region were used. The results shown in Table I indicate that the time before fuel melting is increased 20% by the addition of the beryllium region. Fuel temperature coefficients of reactivity corresponding to axial expansion results in a 33% increase under the same conditions. For problems involving the loss of coolant flow, the input reactivity was held constant. These indicate that a radial expansion coefficient of reactivity characteristic of stainless steel expansion contributes more feedback than the Doppler effect even with the beryllium region present.

The calculations show that the amount of negative feedback available in the temperature coefficients is not large. Moreover, if the expansion is as large as that of an unrestricted fuel pin or stainless steel grid structure at midplane, more feedback can be expected from the expansion than from Doppler effect.

Table I. Core-module Behavior

No.	Type of Problem	$\left(\frac{dk}{dT}\right)_{\text{expansion}}$ $\times 10^6/^{\circ}\text{F}$	$\left(\frac{dk}{dT}\right)_{\text{Doppler}}$	Fuel Melts (sec) ^d		Coolant Boils (sec) ^d		Feedback Reactivity ^a (\$)
				Peak Channel	Average Channel	Peak Channel	Average Channel	
1	Ramp insertion \$2/sec; large Doppler coef.		-0.00468	0.44	0.52	P.T. ^c	P.T.	-0.18
2	Ramp insertion \$2/sec; small Doppler coef.		-0.00117	0.36	0.45	P.T.	P.T.	-0.04
3	Ramp insertion \$2/sec; small Doppler coef. and axial fuel expansion	-8.1	-0.00117	0.48	0.74	1.46	1.56	-0.40
4	Ramp insertion \$0.20/sec; small Doppler coef.		-0.00117	2.18	2.39	2.49	2.57	-0.10
5	Loss of coolant flow; ^b large Doppler coef.		-0.00468	6.25	7.40	7.60	8.16	-0.17
6	Loss of coolant flow; large Doppler coef. and radial expansion	-3.7	-0.00468	9.20	9.70	P.T.	P.T.	-0.58
7	Loss of coolant flow; small Doppler coef. and radial expansion	-3.7	-0.001178	8.30	8.80	9.47	9.47	-0.42

^aTotal feedback reactivity at time corresponding to peak channel fuel melting.

^bFlow decays exponentially from 22 ft/sec for 13.2 sec to a final velocity of 0.8 ft/sec.

^cP.T. means problem terminated before condition was reached.

^dFuel melting and coolant boiling represent the start of melting or boiling at the hot spot.

c. Fuel Cycle. A preliminary estimate of capital costs has been prepared for an integrated pyroprocessing and refabrication facility specifically for the 1000-MWe reference design reactor. The estimate is based on a plant design reflected by the process flow sheets (see Figs. 3 and 4 in Progress Report for December 1964, ANL-6977, pp. 15 and 16). A summary of the capital costs is given in Table II. The grand total for the three portions of the integrated facility is \$11,997,000.

Table II. Preliminary Capital Costs for Fuel Cycle Facility
(thousands of dollars)

	Pyroprocessing	Refabrication	Waste Storage
I. Building			
Nonprocess Facilities	\$ 746	\$ 248	
Process Facilities	2,470	3,194	
II. Equipment Installed			
Process	814	2,183	
Auxiliary	314	198	
Total	\$4,344	\$5,823	\$1,830

The integrated fuel cycle facility includes an on-site waste facility providing interim storage of active wastes and on-site dispersion or burial of low-level wastes. Residual high-level wastes are in a form suitable for shipment and ultimate storage at another site. In this report only the capital cost of the on-site waste facility is presented.

The waste effluents from the combined pyroprocessing and re-fabrication plant are the following:

1. waste metal (from two process steps);
2. waste fused salt (from two process steps);
3. noble gas and volatile fission products (released when the metal fuel is melted);
4. cladding metal scrap;
5. miscellaneous solid wastes including spent fume traps, crucibles, and molds;
6. intermediate-activity liquids from analytical laboratory operations, gas scrubbers, and decontamination activities;
7. miscellaneous liquid wastes requiring storage, and solid wastes requiring burial.

The high-level and long-lived activity is contained in the metal and salt wastes. The bulk of the fission product heat is contained in the metal waste. Gaseous activity is primarily short-lived and is dispersed to the atmosphere after allowing storage for decay. The balance of the wastes (liquid and solid) are of intermediate and low levels. Only the metal and salt wastes contain residual activities considered high enough to be sent to a special ultimate storage area.

The molten metal waste effluents from pyroprocessing are poured into a stainless steel cylinder as they are produced. When full, the waste containers are capped and the cap welded to the cylinder by remotely operated welding equipment. The containers are moved through the cell air lock into a coffin outside the cell. The coffin is moved to the interim storage canal. The cylinder is lowered into a holder and stored under water until the heat content drops to 2 W/liter. The waste containers are then shipped to another site for ultimate storage.

The salt wastes are handled in a similar manner as the metal, with due allowances for heat release of the contained activity and for the physical properties of the solids.

Active gaseous process effluents are stored for 120 days, and after filtration, are dispersed to the atmosphere through a 200-ft stack. Krypton, xenon, and tritium are discharged through this route. Iodine is retained on solid traps.

Intermediate-level aqueous wastes are concentrated by evaporation and the very low-activity condensate, along with other low-level wastes, is dispersed through a storage lagoon. The concentrated waste is stored on-site until activity is low enough for disposal to the environment through the same route.

Miscellaneous solid wastes consisting of traps, molds, etc., are buried on-site.

The capital investment listed for waste storage represents the initial investment required at startup. Additions to the initial storage facilities are estimated at \$820,000 with the first 25% of this total expended about 10 yr after startup and the remainder spent at uniform intervals through the assumed plant lifetime of about 30 yr.

C. General Fast Reactor Fuel Development

1. Development of Jacket Materials

a. Vanadium Alloys. Ternary vanadium-titanium alloys are being studied to find if improved resistance to corrosion by sodium is obtainable without undue loss of fabricability. Chromium principally, but also molybdenum, tantalum, and niobium, are being evaluated as the ternary addition. The alloys were prepared by arc melting small buttons that had been "canned" in stainless steel and hot rolled to an approximate 50% reduction in area. After removal of the steel can, the alloys were cold rolled to 0.08 cm (0.031 in.) sheet for supplying corrosion and mechanical-property test samples.

The ternary alloys produced, and resultant fabricability comparisons with vanadium-20 w/o titanium (V-20), are as follows:

Alloy (w/o)	Sheet fabricability (as compared to V-20 w/o Ti)
V-5Ti-15Cr	Difficult
V-15Ti-5Cr	Similar
V-15Ti-7.5Cr	Similar
V-20Ti-5Cr	Slightly more difficult
V-30Ti-5Cr	Considerably more difficult
V-30Ti-10Cr	Considerably more difficult
V-20Ti-2Mo	Similar
V-20Ti-2Ta	Similar
V-20Ti-2Nb	Similar

All the alloys with the exception of the V-5 w/o Ti-15 w/o Cr alloy were amenable to cold (room-temperature) rolling after hot breakdown of the arc-cast button. It was necessary to "warm" (to 450°C) roll the V-5 w/o Ti-15 w/o Cr alloy to an approximate 60 percent reduction before any reasonable amount of cold rolling could be accomplished without excessive cracking.

Test samples for sodium corrosion have been delivered.

2. Zero-power Reactor Fuels

a. Excursion Test Plates. Six 5 x 15 x 0.63 cm (2 x 6 x 0.25 in.) stainless steel-jacketed zero-power reactor plates were made for excursion tests in TREAT. Leak detection revealed that five of these plates had defective jackets. Cracks appeared just below the weld in the region contacted by the coppered chills. The cracks were of a grain-boundary nature, but no evidence of grain-boundary constituent could be detected microscopically. However, copper was detected by means of a microprobe analyzer in quantities several times higher than background. It was found that particles of copper were gouged from the chill bars during the assembly of the welding fixture and deposited on the surface of the jackets. The attack of this copper on the grain boundaries in the stainless steel under the influence of welding heat produced hot shortness. The assembly technique was changed to prevent copper contamination and the jacket defects were eliminated.

b. Reactivity Coefficient Specimens. A total of 1.25 kg U^{233} was received for manufacture of danger-coefficient specimens for ZPR-VI. This material was separated at ORNL on November 19 and upon receipt December 10 showed a gamma radioactivity of approximately 1 R/hr at 5 cm (2 in.). A six-cavity graphite plate mold was machined and tested by casting depleted uranium into it. The depleted uranium plates were used to test procedures for jacketing the plates in aluminum inner jackets and stainless steel outer jackets. Because of the high radioactivity of this material, a time study is being made on each operation to estimate the total contact time required for the operation. The time required for fabrication of the specimens will be apportioned such that radiation exposure is kept well below the maximum permissible limit.

D. General Fast Reactor Fuel Reprocessing Development

1. Skull Reclamation Process

The skull reclamation process for the recovery of uranium from skull material remaining in a crucible after melt refining of fuel is being studied in a pilot plant, using charges containing about 1.5 kg of uranium. In a pilot-plant run, uranium oxide is suspended in a halide flux, and noble metal fission products are leached from the flux with zinc; after removal of the waste zinc solution, the uranium oxide is reduced in a zinc-magnesium solvent and is then separated from other fission products by being selectively precipitated from alloy solutions which are subsequently removed from the process crucible; the purified uranium is finally dissolved in a zinc-magnesium solvent for transfer from the process crucible to retorting equipment where the volatile solvents are vaporized from the uranium.

A major problem in recent runs has been the entrainment of uranium (in the form of uranium-zinc intermetallic) in the waste zinc-magnesium phase which is transferred out of the crucible after the intermetallic is precipitated. It is believed that the precipitate was kept in suspension during recent runs by excessive agitation of the mixture by blowback of argon through the transfer tube (in order to keep the transfer tube open for subsequent transfers). Two major modifications were made in the most recent run in order to decrease uranium entrainment:

1. The temperature of the reduction step was changed from 700°C, at which most of the reduced uranium is insoluble, to 800°C, at which all of the reduced uranium is soluble. Nearly complete precipitation of the intermetallic was effected by cooling the solution to about 525°C. An initial temperature of 800°C rather than 700°C should promote the formation of large, readily settling crystals.

2. An extended quiescent period, with no blowback, was provided at the end of the intermetallic-precipitation step to allow settling of the uranium-zinc intermetallic compound.

Another reason for reducing at 800°C was to obtain improved zirconia reduction and greater solubility of zirconium in the zinc-magnesium solvent. Since the zirconium remains with the metal solvent during the following intermetallic precipitation, improved zirconium removal is expected following reduction at the higher temperature.

Analyses of waste ingots and product from this run are in progress.

2. Materials and Equipment Evaluation

Eight metals (Inconel-600, Inconel-800, Hastelloy C, Hastelloy X, Type 304 stainless steel, Type 316 stainless steel, tungsten, and molybdenum-30% tungsten alloy) which may see service in the Argon Cell have been exposed to argon-5% nitrogen (the expected equilibrium composition of the atmosphere in the Argon Cell of the EBR-II Fuel Cycle Facility) at 900°C for 1000 hr. The two Inconel alloys and the two Hastelloy alloys were embrittled. This embrittlement was not considered serious since these alloys possessed adequate physical strength and reasonable shock resistance after 1000 hr, but it is possible that the nitridation effect will limit the useful lifetime of these alloys in the Argon Cell atmosphere. The usefulness of tungsten, molybdenum-30% tungsten, Type 304 stainless steel, and Type 316 stainless steel was not impaired by 1000-hr exposure.

3. Advanced Processes

- a. Codistribution of Praseodymium, Plutonium, and Uranium between 30 m/o NaCl-20 m/o KCl-50 m/o MgCl₂ and Copper-Magnesium Alloy. A series of exploratory experiments are in progress to determine

the effect of various solvent alloys containing magnesium on the distribution behavior of plutonium, uranium, and rare earth fission products between selected alloys and salts. In current work, copper-magnesium alloys and 30 m/o NaCl-20 m/o KCl-50 m/o MgCl₂ have been used.

Determinations of the codistribution behavior of uranium, plutonium, and praseodymium between copper-magnesium alloys of various concentrations and the salt system have been made at 600-800°C. Praseodymium distributes strongly to the salt phase at low magnesium concentrations. Both plutonium and uranium favor the metal phase, the uranium more strongly than the plutonium. Of considerable interest are the separation factors in this system. With 20 w/o magnesium in the metal phase, the separation factors at 800°C were as follows:

	<u>Separation factor</u>
Uranium-praseodymium	5600
Uranium-plutonium	33
Plutonium-praseodymium	170

The copper-magnesium-30 m/o NaCl-20 m/o KCl-50 m/o MgCl₂ system is of particular interest because, unlike other systems studied to date, it offers the possibility of uranium-plutonium separations by liquid-liquid extraction. The data also suggest that an adequate separation of rare earth fission products from uranium and plutonium is possible by a single-stage extraction.

b. Ion Exchange in Fused Salts. An exploratory study of ion exchange between molten salt solutions and certain inorganic solid materials has been completed. In the experimental work, potassium hexatitanate (Tipersul) was used as the ion exchange material. A run to determine the distribution of Ba⁺² (added as anhydrous BaBr₂ or BaCl₂ spiked with ¹³³Ba and analyzed by gamma counting) between Tipersul and molten NaCl-KCl-MgCl₂ has been completed. 0.015M Ba⁺² in 30 m/o NaCl-20 m/o KCl-50 m/o MgCl₂ at 450°C was taken up on Tipersul with a distribution coefficient (conc. in Tipersul/conc. in salt) of about 6. This technique appears to be potentially useful for the cleanup of molten chloride salt solutions used as extractants for fission products in pyrometallurgical processes.

4. Consolidation of Fission Product Wastes in Alloys.

Data are being obtained on the extraction of rare earth fission products from molten chlorides by aluminum-magnesium alloy (see Progress Report for November 1964, ANL-6977, p. 25). An extraction technique is a possible method of purifying salt for recycle to plutonium-rare earth separation steps and of consolidating most of the fission products in

a small volume of metal for storage. Distribution coefficients (conc. in salt/conc. in metal) have been determined for yttrium, cerium, and praseodymium between 30 m/o NaCl-20 m/o KCl-50 m/o MgCl_2 and aluminum-magnesium alloys at nominal magnesium concentrations of 13, 20, and 30 w/o and at temperatures from 550 to 900°C. Solubilities of these rare earth elements in these three compositions of aluminum-magnesium at temperatures in the 550-900°C range also were determined. In aluminum-20 w/o magnesium the solubilities were as follows:

Element	Solubility (w/o)	
	500°C	600°C
Y	0.91	3.45
Ce	2.24	3.27
Pr	2.53	3.91

As in the case of lanthanum, the solubilities increased with increasing temperature up to about 700°C and then became retrograde; the solubility of all four elements decreased with increasing magnesium concentration.

Distribution coefficients between aluminum-20 w/o magnesium and the salt were as follows:

Element	Distribution Coefficient	
	550°C	600°C
Y	2.42×10^{-3}	4.39×10^{-3}
Ce	4.12×10^{-4}	7.08×10^{-4}
Pr	-	4.4×10^{-4}

These data indicate that aluminum-magnesium alloy should be a very satisfactory medium for the reduction and consolidation of rare earth fission product wastes.

As another phase of this investigation, NaCl-KCl- MgCl_2 salt which had been used in a distribution experiment with magnesium-aluminum alloys was analyzed to determine aluminum content at the end of the experiment. Results showed an aluminum concentration in the salt of 1.5×10^{-3} to 4.9×10^{-3} w/o. A salt with this aluminum content would contribute only negligible aluminum contamination if used to process uranium and plutonium.

5. Analytical Methods

A method for the determination of hydrogen present as bicarbonate or caustic is required for the evaluation of sodium peroxide for use in large fast critical experiments with plutonium oxide cores. Scouting tests indicated that treatment of peroxide with carbon dioxide would release hydrogen as water, which could be removed by heating and determined by gas chromatography.

E. EBR-II

1. Sticking of Control Rod and Oscillator Drive Mechanisms

The sticking oscillator assembly was removed this month. The oscillator holddown and drive shaft assembly was freed from the Stellite-lined No. 8 control rod hole in the reactor vessel cover on December 11. An upward pull of about 2500 lb, supplemented by impact, was required to free the drive. The reactor vessel cover was raised manually 14 in. to permit insertion of the two "ball catcher" tools, inserted through the No. 7 and No. 9 control rod drive holes. The tools were oriented, opened, and then raised to the thermal baffle adjacent to these control rods on the under side of the cover. The oscillator drive shaft was then removed from the reactor cover and the primary tank on December 14. The "ball catcher"

tools were removed, and the reactor vessel cover closed manually and locked.

Figure 1. Oscillator Holddown and Drive Shaft Assembly

Inspection of the oscillator holddown and drive shaft assembly (see Figure 1) showed a series of vertical grooves caused by metal particles jamming in the clearance between the Stellite sleeve in the reactor vessel cover and the oscillator shaft during removal operation. No balls or debris were found in either "ball catcher." Balls were not imbedded on the oscillator drive shaft or in the labyrinth section except in the vertical grooves mentioned above. During cleaning of the assembly one small steel particle, about $1/8 \times 3/64 \times 5/64$ in., was recovered.

After cleaning, the lower end of the holddown assembly was disassembled. The ball bushing located about 58 in. above the bottom of the holddown tube was found to be intact; however, the lowest ball bushing was disintegrated.

All ball bearings (about 100) were missing. All but one of the ball retainers were severely damaged, and were found outside and just above the bearing. The two end rings of the bushing were badly worn and loose on the shaft. The snap ring bushing retainer was in place, but bent.

The bayonet end of the oscillator drive shaft gripper was bent open slightly and the 1.469-in.-diameter shaft was badly grooved axially between a distance of about 3 to $4\frac{1}{2}$ in. up from the bottom end. The grooves were apparently due to the jamming of ball bearings in the clearance between shaft and upper chamfered area of the holddown tube.

A special gripper tool which locks into the oscillator thimble was inserted through the reactor vessel cover to remove both the oscillator rod and thimble from the reactor core. During attempted removal, the oscillator thimble jammed in the control rod thimble, probably because of debris between the two. Continued efforts have not succeeded in removing the oscillator rod and thimble. By pulling with a total force of 500 lb as read on a dynamometer and impacting with a 7-lb weight both in the "up" and "down" direction, only about $1/8$ in. of movement was obtained. Additional impacting produced no measurable movement.

A preliminary measurement of the location of the oscillator thimble with respect to the control rod thimble indicates that the top edge of the oscillator thimble is about $7/16$ in. below the top edge of the control rod thimble. It was expected that previous movement of the oscillator thimble should have put it about 1 in. above the control rod thimble. The oscillator rod position is being verified. If substantiated, the low position could mean that the key bar in control rod position No. 8 is damaged.

The top end of the oscillator rod was found to be $17/32$ in. below the top end of the control rod thimble. Control rod thimble No. 8 was found to be at the same elevation as control rod thimble No. 9.

The special EBR-II magnetic "ball retriever" tool was installed in control rod positions 7, 8, and 9 and the Stellite sleeve flange face was probed for debris. No balls or metal particles were found in the catcher screen of the tool or adhering to the small horseshoe magnet.

A plug gage tool 0.020 in. larger than the labyrinth of the control rod drive shaft was successfully passed through each Stellite sleeve in the reactor vessel cover. Sleeve No. 7 appeared to have a burr in it. With a little effort, however, the plug gage was made to pass freely through the sleeve. Small scratches were found on the plug gage after removing it from control rod positions No. 7 and 8.

2. Maintenance

Both systems for purification of sodium remained shut down during the month. The composition of the secondary system argon remained fairly constant at 10 ppm hydrogen and 500 ppm nitrogen. The composition of the primary system argon showed less than 10 ppm hydrogen and a small increase in nitrogen content to approximately 1.5 percent.

Two duplex strainers were installed in cooling water lines in the Sodium Boiler Plant to provide a supply of filtered cooling water to all heat exchange equipment in the building. A flowmeter with alarm contacts was installed in the cooling water line to the main secondary sodium pump.

During the last power run, there were indications of a deficiency of cooling water supply to the Sodium Boiler Plant. Preliminary tests indicate that this incoming flow is limited by the building supply. Further testing is in progress.

The plugging indicator control panel was relocated to provide safer access to the plugging valves. The plugging indicator was cut free from the system to allow installation of block valves. This will allow isolation of the loop in case of future bellows rupture and subsequent sodium dripping.

Completion of the installation of the check valve in the feedwater line just below the steam drum in the Sodium Boiler Plant was delayed because of questions concerning the certification of welding electrode and qualification of welding procedures. Details of the procedure have been agreed on, the welding shop has qualified a procedure and certified two welders, and resumption of work is awaiting word that the welding electrodes to be used pass the chemical and physical tests satisfactorily.

The fabrication and installation of a bypass in the cooling tower riser No. 1, similar to the one previously installed on the No. 4 riser, is now almost complete.

3. Fuel Cycle Facility

a. Cells. Some difficulties have been encountered in opening and closing the steel shutters used to reduce radiation exposure of the cell shielding windows. The difficulties were traced to a reduction in the torque transmitted by slip clutches in the shutter drive units. A method has been devised for remote adjustment and testing of these clutches, using Model A manipulators. Testing is carried out with a spring scale and lever arm.

Several portable quartz-xenon lights have been made for in-cell use as supplemental lights. Two or three of these will be placed on the emergency power circuit.

b. Fuel Recovery and Fabrication. Following successful out-of-cell testing of a decanner modification, consisting of a remotely replaceable scrap separator tube, the in-cell unit was similarly equipped. The fuel-element-removal machine has been remotely installed on the dismantler in the Air Cell.

A few operational tests indicate that the 91 fuel elements of a core subassembly can be removed, inspected, and loaded into magazines in about 30 min. These operations require approximately 375 manually initiated operations at the control console and 50 operations of the master-slave manipulator.

Bonding and bond testing to reclaim the 2417 fuel elements, obtained chiefly by disassembly of special assemblies prepared for dry and wet critical and related operations, have been completed. The total number of acceptable elements obtained was 2233. The principal cause for rejection was an indication of the presence of bubbles or voids in the sodium bond. Other causes included damaged tips, sodium levels not as specified, leaks in weld or jackets, inadequate data, and damages occurring in testing or assembly.

Four additional core assemblies and two control assemblies have been completed. The assembly straightness testing and correcting machine is being fabricated, and preliminary testing has been carried out with a leak tester modification which provides for pressure determination and sampling of fission gas in discharged elements.

c. Skull Reclamation Process Development. Fabrication of a prototype (M-2) skull oxide reclamation furnace is continuing (see Progress Report for November 1964, ANL-6977, p. 31). The furnace will be resistance heated and will accommodate a 14-in.-ID pressed-and-sintered tungsten crucible. A preliminary heat test of the split-cylinder furnace heater was performed which involved heating a solid steel cylinder (8 in. in dia by 12 in. long) inside a thin-walled steel pipe. The steel cylinder acted as a heat sink to prevent rapid heat-up of the furnace heater and as a radiant heat source at the center of the furnace when the heater was opened. As a radiant heat source, the steel cylinder would subject the furnace support columns to heating conditions similar to those which will exist during plant operation.

The furnace heater temperature was maintained at 900°C for 2 hr, after which the power to the heater was turned off, and the heater halves opened for cooling purposes. The results of this test showed the following: (1) there were no large sources of heat loss during heating, and (2) the maximum temperature of 160°C which the support columns for the furnace heater attained was not considered excessive and did not interfere

with the proper functioning of the support column. The heater halves were opened and closed satisfactorily, indicating little or no warpage of the furnace and its supports.

Additional heating tests are planned with the M-2 furnace in place within the furnace heaters.

In the retorting step, magnesium-zinc is distilled at 800-900°C from the product material resulting from the uranium dissolution step, after which the residual uranium is melted (~1200°C) to form an ingot. All the steps prior to the retorting step are conducted in a tungsten crucible. Although present plans call for the use of beryllia crucibles for the retorting step, the use of tungsten crucibles is being investigated because of their greater structural reliability and their availability in plant-scale sizes. Operations at the maximum retorting temperatures do not result in attack of the tungsten crucible, whereas consolidation of the uranium product by melting in a tungsten crucible results in attack of the tungsten by the molten uranium. However, the residual uranium product remaining after vaporization of the zinc and magnesium sometimes adheres to the surface of the tungsten retorting crucible. One possible means of freeing uranium from a tungsten crucible surface is by converting the uranium to the hydride. It is also possible that the hydrided uranium product rather than uranium metal can be the final product of skull processing and can be used as make-up material in melt refining.

Two retorting runs (see Progress Report for November 1964, ANL-6977, p. 33) were made to prepare hydrided uranium product for testing its use as make-up material in a melt refining run. The runs were conducted in a modified melt refining furnace. The equipment was operated in an inert atmosphere (argon-5 v/o nitrogen) glovebox.

In these retorting runs, charges were prepared which simulated an incomplete transfer of the flux heel from the uranium reduction step. The charges to a pressed-and-sintered tungsten crucible consisted of Zn-12 w/o Mg-10 w/o U (nominal composition) and fresh flux (MgCl₂-50 w/o CaCl₂) containing added MgO and Ce₂O₃ to simulate the process flux heel. The flux was first pretreated with magnesium to remove residual moisture. In the first run, the feed was composed of about 4.6 kg of Zn-Mg-U ingot, 200 g of flux, 20 g of MgO, and about 5 g of Ce₂O₃. (This quantity of flux heel would have represented a very poor flux transfer and amounted to a 1-in.-thick layer on top of the ingot.) The distillation was carried out at 950°C and at reduced pressures of between 32 and 50 Torr. At the end of the run, the uranium retort product contained flux, oxide, and undistilled zinc-magnesium as impurities. An appreciable amount of zinc-magnesium and flux had escaped from the graphite retorting enclosure and condensed on the wall of the water-cooled bell jar enclosure. This leakage was attributed to the relatively large amount of flux present in the charge.

The layer of molten flux on top of the zinc-magnesium metal phase appears to have inhibited the normal vaporization of the zinc and magnesium. Distillation of the zinc-magnesium from beneath the molten flux resulted in a great deal of spattering of the melt from the crucible.

In the second run, two changes were made in the operating conditions in order to try to achieve more complete retorting of the zinc-magnesium and flux and to reduce the escape of these materials from the graphite retorting enclosure: (1) a reduction in the amount of flux heel in the feed and (2) distillation at a pressure lower than that in the first run in order to improve the distillation of the molten flux layer. The feed was composed of about 5 kg of Zn-Mg-U ingot, 88 g of flux, 9 g of MgO, and 2 g of Ce_2O_3 . This feed was charged to the crucible containing the retort product from the first run. Except for a new graphite condenser, the distillation apparatus was the same as that used in the first run. The distillation was carried out at 950°C and a pressure of less than 1 Torr. At the end of the run, the retorted product was mainly in the form of a metallic, sponge-like mass which adhered to the crucible bottom; a small amount of the material adhered to the crucible sidewall. A considerable amount of zinc-magnesium and flux had escaped from the graphite retorting enclosure and condensed on the wall of the water-cooled bell jar cover. The graphite condenser, collector, and secondary crucible cap which were bonded to each other by condensed metal and flux could not be disassembled from each other and were removed as one piece. They were readily lifted off the secondary crucible container without sticking. The vapor opening above the retorting crucible was almost completely plugged with condensed material. [In previous retorting runs, the presence of small amounts of flux heel ($<1/8$ -in.-thick layer) which had been transferred along with the uranium product solution (Zn-Mg-U) did not result in bonding of the components of the distillation apparatus to each other.] On the basis of these test results, it is planned to examine each zinc-magnesium-uranium ingot before retorting it and to remove any flux heel that can be seen.

Following the distillation step, the crucible and its contents were subjected to a hydriding step in a hydrogen atmosphere at a temperature of 250°C . The hydrided material (~ 970 g), which readily poured from the crucible, was used as a make-up material in one of the melt refining runs discussed in the following paragraph.

Two melt refining runs with unirradiated fuel material were carried out in a zirconia crucible. In both runs, the furnace atmosphere during melt refining was argon (1 atm), and the molten metal was held at 1400°C for liquation times of 2 hr. In the first run, 10 kg of chopped, depleted uranium-5 w/o fissium pins and 40 g of sodium were melt refined; the poured yield was 92.9 w/o. The surface appearance of the ingot was bright and shiny. In the second run, the charge consisted of chopped, depleted uranium-5 w/o fissium (~ 9.1 kg), sodium (40 g), and the above-described hydrided material (~ 970 g). The crucible and contents were

first heated at 400°C for 1 hr under reduced pressures (30 to 165 Torr) in order to decompose the hydrided material. About 5 gram-moles of hydrogen were evolved and evacuated from the furnace. After decomposition, the charge was meltrefined; the poured yield was 93.0 w/o. The surface appearance of the ingot was somewhat dull. However, it was considerably brighter in appearance than any ingot produced for the initial EBR-II fuel loading. (The melt refining furnace for these ingots was operated in an air atmosphere; an argon atmosphere was used within the melt refining furnace itself.) Analyses of samples from the two ingots are not yet available.

Fabrication of the plant-scale retort is complete. It has been assembled and is undergoing leak tests. Following the leak tests, shake-down retort runs with 30- to 40-kg charges of Zn-Mg (no uranium) will be made in this equipment, which will be initially operated in air. Subsequent to the tests, the retort will be installed in the large inert atmosphere enclosure (see ANL-6977, p. 31) for plant-scale demonstrations of the skull reclamation process.

4. Irradiation Surveillance

Ten groups of six pieces per group of round 0.204-in.-diameter multi-notch impact bars were prepared from quarter-thickness regions of 2-in.-thick stainless steel (SS) 304 plate utilized in the fabrication of the EBR-II reactor vessel cover. A SS 304 group was included in each irradiation-surveillance capsule containing the materials utilized in the construction of the reactor. The irradiation surveillance program contemplates yearly withdrawal of capsules to assess irradiation damage to the various structures being irradiated in the core region. In the EBR-II, irradiation damage to the reactor vessel itself is not a problem because of the remoteness of the walls from the neutron source.

Additional irradiation-damage specimens in the form of magnetic sonic bars were also prepared for inclusion in the surveillance program for the study of the damage mechanism at elevated temperatures. Measurements of resonant and harmonic frequencies, damping, and the magnetic properties of the EBR-II SS 304 vessel cover steel were delayed by the limitations and operational failures of the test equipment and by the nonmagnetic properties of the steel. Sonic properties of the bars, resonant and harmonic frequencies, and associated log decrement, were measured with modified forms of the original SS 304 bars. The bar modification, necessary for the cave use of the electromagnetic driver and detection transducer on highly radioactive SS 304, introduced a 1/16-in.-diameter by 1/4-in.-long, slightly oversized, soft steel slug into each end of the bar. Even with this modification, the detector signal required amplification before reliable measurements were possible. The instrumentation troubles for the mapping of the magnetic properties of the steel-tipped stainless steel magnet bars appear to be concentrated in the voltage-stabilizer section of the DC plate power supply common to several amplifiers.

F. FARET

1. General

The status of the design packages of Title II is as follows:

Package I - Site Preparation - complete.

Package II - Reactor Vessel - complete. Last month it was reported (see Progress Report for November 1964, ANL-6977, p. 34) that proposals from bidders be in by November 30, 1964. Because of factors such as workload and adequacy of facilities, some of the pre-selected firms notified the Laboratory that competitive proposals would not be submitted and withdrew from the selection process. Subsequently, the emphasis in selection was revised from competitive bidding to determining that of the best qualified firm available to do the work. It is expected that the selection process will be completed in the near future.

Package III - Liquid Metal Heat Exchangers - This package is being prepared in final form by Bechtel for submittal to the Laboratory.

Package IV - Liquid Metal Pumps - The Architect-Engineer has essentially all Laboratory comments on this package.

Package V - Control and Instrumentation - Bechtel's draft is being reviewed at the Laboratory. A meeting with Bechtel engineers was necessary to clarify certain outstanding questions.

Package VI - Special Piping Materials and Valves - ANL comments have been returned to the Architect-Engineer with instruction to prepare the final specifications.

Package VII - Special Early Procurement Items - Laboratory review of the draft of this package has been completed, and comments are ready for transmittal to Bechtel.

Package VIII - Specialty Steel - This package has been deleted.

Meetings have recently been held both at Bechtel and at the Laboratory to handle late design considerations preparatory to issuing Package IX to the Laboratory. Because of the advanced status of the design schedule, these meetings were limited to discussions of special design considerations of high priority. These meetings provided Bechtel with a description of Laboratory expectations of the method of handling these problems. Major subjects covered included the following:

1. Fuel and Equipment Transfer System
2. Wet Fuel Storage Tank Temperature Distribution and Thermocouple Wire Routing
3. All Trap Specifications
4. Plugging Meter Specification
5. Final ANL Cell Penetration Requirements
6. Minor Cell Missile Criteria
7. Argon Cooling Cavity - Vault Blast Considerations
8. Final Cell Floor Pits Configuration
9. Alignment Tolerances for Cell Components
10. Construction Test Requirements for Containment
11. Revisions to Set Back Control Mode
12. Plant Protection System for Fuel Handling
13. Reactor Vessel Thermocouple Wire Routing
14. Cell Floor Configuration at Reactor Vessel
15. Vacuum Provisions for Liquid Metal Systems
16. Out-of-cell Wiring Provisions for Cell Activities

2. Shielding Windows

A rough draft of the technical specifications for the FARET Shielding Windows has been completed.

Assembly of the components of the window test fixture is proceeding as planned. After the assembly has been checked for leak tightness, the hydraulic pressure test will be conducted.

3. Cell Penetrations

The basic design (see Progress Report for April 1964, ANL-6885, p. 34) and general requirements for penetration sleeves and internals have been completed for connecting electrical and gas services to in-cell equipment. A mock-up to check the feasibility of the design has been completed.

Five samples of submarine-type cable, for use in the electrical penetrations, have been procured and are being tested. The samples were 38 #22 ga. solid conductor, with smoked sheet base gum insulation (no antirads added), and a Hypalon base jacket compound. The samples varied in length and in the nature of blocking compounds.

One 6-ft cable (Sample 2) with vendor's blocking compound, #30-02-DP, was tested for leak tightness in the penetration mock-up. The cable was secured to the bulk headplate with a cable gland drawn up to hand tightness. The pressure chamber was filled with a mixture of nitrogen and Freon to 32.5 psig. A Freon leak detector, with a sensitivity of 0.09 cu in./day, was used to monitor leakage. Although there was no detectable leak through the cable, the gland and pressure gauge did leak with a resulting 4.5-psig pressure drop in the pressure chamber over a 24-hr period, which corresponds to a leak rate of approximately 0.147 cu ft/day. When the test chamber was pressurized to 2 in. H₂O above atmospheric, there was no detectable leakage through either the cable or gland. This test has demonstrated acceptability of the preliminary design. Another test of a cable with a natural rubber (with antirad additive - Antiox 40 ID) jacket will be conducted and compared with the Hypalon-jacketed cable.

Since the cables are expected to perform within the cell up to radiation doses of 10^8 Rads, short lengths of cable and blocking compound samples are being irradiated in CP-5 to enable selection of the most desirable insulating material for the submarine-type cable.

4. Fuel and Equipment Transfer System

Figure 2 shows the basic design which has been developed for the fuel and equipment transfer facility. The criteria used for the design of the transfer port and the handling equipment includes the supposition that the cell-containment criteria and the argon atmosphere of the cell will be maintained during transfer operations. The system will permit the transfer of radioactive and nonradioactive fuel subassemblies, radioactive waste, and other core components out and into the cell, as well as nonradioactive and radioactive equipment, such as the reactor vessel head, including the integral control rod drive mechanisms. Since radioactivity and sodium contamination is anticipated, provisions are included to decontaminate the facility by a washing-down type of operation. Radiation levels of < 2.5 mR/hr are anticipated when the facility is not in operation to allow for normal access of personnel. For greater activity levels during transfer operations, limited access is anticipated although most of the operations may be performed from the control area. There is a small transfer plug within the large equipment transfer plug, called the fuel transfer port. The frequency of operation for the small port is expected to be of the order of once per day, whereas the large port will be used on the order of once per year.

The equipment transfer port and plug are sealed on the inside of the cell by means of an oval cover, 9 ft 0 in. x 8 ft 3 in. The shape of this cover and the port allows for replacement of the cover and its integral seal should a leak develop.

Figure 2. Fuel and Equipment Transfer Tunnel

A large equipment container can be bolted to the equipment transfer port on the tunnel side to enable the transfer of equipment from the container to the cell while excluding air from entering the cell.

The small fuel transfer port is 14 in. in diameter. This size allows for the transfer of waste containers and fuel. The fuel transfer casks or containers may be sealed to the port surface on the tunnel side by means of a hydraulic hoist.

Additional support by means of screw jacks is furnished to prevent lowering of the container. This support is adequate to withstand additional loads on this cask or container up to a pressure of 30 psi in the cell. The port cover is removed by the cell crane and stored in the cell during the transfer operation. After this operation, the cover is returned and the seal is tested before the container can be removed. Provisions are included for the purging of the cask and containers with air and argon.

Container transfers between the cell transfer ports and equipment hatch (located in the rear-cell area) are made by means of a

carriage traveling on suspended rails. The rails for the carriage were located to provide support above the center of gravity.

5. Wet Fuel Storage Tank

The basic design of the internal structure for the wet fuel storage tank has been completed. These internals consist of a top grid plate, a lower support plate with attached funnel-type guide tubes, and connecting pipe columns, permitting storage of 151 spent elements arranged on a 5-in. triangular pattern. The top grid plate positions and maintains each element in its proper radial orientation. The lower plate, carrying the spent fuel elements, is supported by six equally spaced pipe columns connected to the top grid plate. Funnel-type tubes are located and welded to the lower support plate for aligning and guiding the lower ends of the fuel assembly when being lowered. The internal structure can be replaced as a unit if required.

6. Fuel-handling Machine

As much of the cooling equipment as practical, especially the compressor, the filters and vapor traps, the gas purifier, and the heat exchanger, are to be located outside the cell. The vapor traps as well as the argon gas returned to the compressor will be cooled by refrigerant obtained from the cell atmosphere cooling system. This is conveniently possible because the system was sized originally so as to permit heat loads to be discharged into the cell atmosphere. Therefore, no change in capacity is involved if the intermediate heat path through the cell atmosphere is eliminated and heat is discharged directly into the refrigeration system. The cooling equipment will be located underneath the cell in an extension of the fuel and equipment transfer tunnel.

A gripper, a holddown device, and seals, comprising the main mechanical component of the lower pick-up portion of the fuel-handling machine, are being fabricated for testing.

7. Fuel Slip-fit Experiment

A sample of a slip-fit fuel element is under test in the CP-5 reactor (see Progress Report for November 1964, ANL-6977, pp. 38-41). The measured central temperature is 1030°C. The measured cladding temperature is 210°C. These temperatures correspond to a measured power output from the fuel of 3500 W or ~230 W/cm.

The fuel pellets have been thermally cycled about 12 times. The cooling rate of the sample (in seconds) is much greater than the heating rate (in minutes) by virtue of the reactor scram and startup times. To date, no effects of cycling are apparent, i.e., temperatures are the same as initially. This sample will remain in the reactor for the remainder of the month.

The temperature difference expected between the fuel center and the clad outer surface has been calculated by the same methods used earlier to predict that an average fuel temperature change of over 500°C is possible in the FARET test zone. On the assumptions that the oxide thermal conductivity is $1.5 \text{ B}/(\text{hr})(\text{ft})(\text{F})$ and the fuel-clad gap conductance is $600 \text{ B}/(\text{hr})(\text{ft}^2)(\text{F})$, the calculated value for the CP-5 ΔT is 890°C as compared with the measured value of 820°C . At least part of this 8 percent discrepancy is due to the self-shielding which occurs in a thermal flux. In any event, the discrepancy is not large enough to cause concern that a 500°C fuel temperature change cannot be obtained in FARET. The FARET test zone heat fluxes are $265 \text{ W}/\text{cm}$ average and $330 \text{ W}/\text{cm}$ maximum. Future fuel for the CP-5 experiments will be more highly enriched in order to obtain these higher fluxes.

8. Core Instrumentation

a. Electrical Connectors. Seal tests for the electrical connectors (see Progress Report for September 1964, ANL-6944, pp. 47-49) were performed in a test fixture as shown in Figure 3. The seals were tested at various clamping pressures. A vacuum pump was used to evacuate the seal chamber. Seal leak rates as a function of clamping pressure were thus obtained.

Figure 3. Fixture for Seal Tests

Four relatively promising seals (see Figure 3) were selected for intensive testing. All were silver plated except seal #3, which was bare Inconel. Surfaces of the fixtures mating with the seals were of stainless steel, carefully lapped to an approximately 4- μ in. finish.

Each specimen was subjected to at least ten runs in which at least four measurements at stepped clamping pressure levels were made. Between runs the fixture was unloaded and the seals relaxed. Results are tabulated in Table III.

Table III. Seal Tests

Seal No.	Clamping Pressure (lb)	Vacuum Obtained (absolute pressure in μ Hg)									
	Run Number	1	2	3	4	5	6	7	8	9	10
1	25	85	80	90	90	110	100	100	90	85	80
	75	78	85	78	75	92	82	85	65	68	61
	200	68	48	40	33	55	42	65	35	27	37
	400	28	11	17	7	38	14	25	5	8	17
	500	12	3	7	5	17	8	12	1	2	12
2	30	50	60	58	68	55	75	80	82	85	70
	60	28	30	30	32	25	35	28	32	30	30
	90	22	20	22	20	19	27	22	20	22	15
	120	18	17	17	18	17	20	18	14	18	20
3	100	150	140	Sealing surface finish damaged by seal.							
	300	150	135								
	600	125	120								
	700	120	120								
4	30	150	175	155	150	175	150	100	150	150	100
	60	40	60	32	40	65	35	70	28	40	24
	90	25	38	22	22	30	27	60	20	26	18
	120	18	25	18	17	19	19	20	17	17	12

Additional work has been initiated to test the above seals at temperatures up to 1200°F, first in dry argon and subsequently in sodium vapor.

9. Hydraulic and Thermal Investigations

a. Water-flow Tests of Full-scale Subassemblies. A leg has been added to an existing water loop to permit hydraulic testing of full-scale FARET subassemblies. Until dummy test zone and driver subassemblies are delivered, tests have been conducted with an empty subassembly.

Leak rates in the annuli between subassembly lower adapter and core support grid plates were measured. Results indicated that such leak rates are difficult to calculate accurately, probably because of the unknown eccentricity of the annuli. However, both the calculations and measurements predict leak rates which can be accommodated.

The ability of the subassembly to remain seated in the lower grid plate, even with the equivalent of a 100-psi pressure drop through the core, was confirmed. Under this condition, it was calculated that there is a net force of over 80 lb seating the subassembly. The test results verified the calculations.

b. Reactor Vessel Model. In order to make a rough check of the flow distribution, mixing, transient-temperature-change attenuation, and response-to-leak characteristics of the reactor vessel, a 1/8-scale transparent, water-flow model has been constructed (see Figure 4). Slow-motion movies have been made of the model while ink was injected at different locations. Analysis of the movies is not yet complete. No major problem areas have been uncovered through operation of the model.

Figure 4. Water-flow Model

c. Wet Fuel Storage Tank Cover. The thermal design of the wet fuel storage tank cover has been completed. This design is intended to prevent the cover flexible seals from being heated above a maximum allowable temperature of 400°F when the storage tank sodium is at 600°F. By using loosely hung radiation shields for insulation, the desired temperature gradient in the cover should be achieved without producing any excessive thermal stresses or distortions.

d. Cooling of Irradiated Subassemblies. Calculations have given the rate at which irradiated subassemblies will heat if cooling cannot occur (see Figure 5). The parameters varied at irradiation time, cooling time after irradiation, and core size. The results are of interest to the fuel handling equipment designers, who may be able to provide back-up cooling devices if the time available for taking action is adequate.

Figure 5. Maximum Subassembly Fuel Temperature at Midplane vs. Time after Loss of Argon Cooling

10. Fuel Assembly Sodium Flow Test Loop

The pressure vessel has been received from the vendor. All vessels are on hand and meet engineering specifications. Shop personnel are removing the shipping plugs on the nozzles and machining the nozzle ends for welding. A 2-in. elbow and a 10-in. to 4-in. reducer must be welded to the vessel before it can be lowered into position. The vessel should be in place and ready for assembly of heaters, thermocouples, and insulation in about two weeks.

The dump tank and the expansion tank are ready for final installation. All heaters, thermocouples, and insulation are in place. Installation should be completed within two weeks. A welded box enclosure will be placed around the dump tank. The space between the dump tank and the inner walls of the enclosure will be filled with a loose insulating material.

The interim-pump case is nearing completion. Some of the welds required removal and rework which caused a delay. The case will be finished and returned to Building 308 within the next two weeks, depending on completion of radiography.

The heat exchanger is now welded into its shroud and awaits the welding of the two 5-in. elbows to the stub ends.

The 3-in., 4-in., and 5-in. pipe lines are about 80% welded in sub-assembly form. The welds appear excellent and have passed radiographic inspection.

Much of the operating instrument panel board assembly is in place. The electricians are connecting the transformers and doing other necessary electrical work.

Two-inch dump lines were rejected because of corrugations on the inside of the bends. New pipe bends are being fabricated.

A preliminary layout of a fuel-element-handling enclosure has been completed for cost-estimating purposes.

A tentative design of a creep apparatus for the testing of specimens of Type 304 stainless steel in an environment of sodium flowing at a temperature of 1200°F has been prepared to be inserted into the 5-in. pump inlet line.

11. Measurement of High-temperature Strain

An experimental program was initiated to investigate high-temperature strains in the piping system of the FARET Fuel Subassembly Test Facility. Of several commercial gauges for which good results up to 1000°F were claimed, a weldable full-bridge strain gauge was selected. The items have been ordered and are expected soon. Combined with the welder and the strain gauge read-out units, sufficient equipment will be available for high temperature strain measurement. Concurrently, plans are being made for the mechanical measurement of pipe displacement and temperature determination.

At the present time, preliminary effort is expended on several high temperature trial gauges. Heavy reliance is put on the information to be gained from these tests. It is hoped that the difficulties usually encountered with high temperature gauges can be discovered before the actual piping system is completed.

12. Fuel Fabrication Development

A design for the modification of the injection-casting furnace for the fabrication of FARET fuel has been completed. The modification involves an extension collar for the furnace bell, a longer mold heater, and pallet stand to accommodate the long FARET molds. Bids have been received for the tooling and production of three different sizes of Vycor

molds, all 1.1 m long. The bore diameters of these molds are 3.75 mm (EBR-II size), 3.96 mm (FARET driver rod), and 4.18 mm. The earliest date for mold delivery is late February.

The EBR-II fuel rod was made with a strut-type restrainer to hold the fuel pin below the sodium level. In the FARET driver rod the strut occupies so much volume that the rod becomes excessively long. A design providing for a partition-type restrainer has been proposed. This consists of a fluted plug held in place by beads that are embossed in the jacket and engage grooves in the plug. Specimens made with stainless steel restrainers and EBR-II fuel-jacket tubing were tensile tested at room temperature. The pressure required to displace the plug held by a single bead was 570 to 710 kg/cm² (8,000 to 10,000 psi). Additional specimens have been made with multiple grooves and beads that will be tested at elevated temperature.

II. GENERAL REACTOR TECHNOLOGY

A. Experimental Reactor and Nuclear Physics

1. Fast-neutron Hodoscope

One of the 320 fast-neutron detection channels used in the hodoscope is shown in Figure 6.

Figure 6. Diagram of Detection, Processing, and Display System for Fast-neutron Scanner

(The high-voltage supply system is not shown)

Each channel consists of a fast-neutron detector, signal-processing equipment, and a display system. Also each channel is focused at a different point on the vertical plane through a fuel element to provide spatial resolution. A hodoscope has been designed to monitor fuel meltdown experiments performed in TREAT. The hodoscope, which incorporates these detection units, has been designed for linear response and possesses a time resolution of 1 msec. A fuel-meltdown experiment is normally completed in a few seconds.

B. Theoretical Reactor Physics

1. Studies of the Doppler Effect

Tungsten has been proposed as a major structural material for some fast reactors. A preliminary study has been made to evaluate the temperature-dependent cross sections of tungsten in the energy regions of interest.

The most recent measurements of the resonance parameters of tungsten extend to about 500 eV.¹ Since the measured values seem to show that the level densities of individual tungsten isotopes are rather low, with perhaps the exception of W¹⁸³, it is extremely difficult to obtain a reliable set of average parameters to be used in the statistical region. However, s-wave average parameters for individual isotopes, obtained by using known parameters (the resonant widths for neutron emission, Γ_n^0 , and for gamma emission, Γ_γ) and some guesses, are shown in Table IV.

Table IV. Resonance Parameters of the Tungsten Isotopes

Mass Number of Tungsten Isotope	Average Spacing $\langle S \rangle$, eV	$\frac{\langle \Gamma_n^0 \rangle}{\langle S \rangle}$, $\times 10^4$	Γ_γ , MeV
182	60	1.00	60
183	52(J=0); 17.38(J=1)	2.63	100
184	105	1.67	60
186	105	1.67	60

In order to check the accuracy of these s-wave average parameters, calculations for the infinite dilute cross sections above 500 eV were made. The Porter-Thomas distribution was used in the statistical averaging. The results seem to show excellent agreement with the measured values between 500 and 7000 eV reported by Kapchigasheve and Popov.² The calculated values begin to drop off beyond 7 keV. It is believed that p-wave contribution becomes significant beyond this energy. Calculations for the case $\sigma_p = 30$ b/tungsten atom, taking into account the resonance self-shielding, were also made for various temperatures.

¹ Reports to the AEC Nuclear Cross Sections Advisory Group, WASH-1042 (February 1963); WASH-1046 (January 1964); WASH-1048 (June 1964).

² Kapchigasheve, S. V., and Popov, Yu. P., At. Energ. (USSR), 15(2), 120-126 (1963).

2. ZPR-VII Data Analysis

In order to study the effect of cross-section homogenization for cladding and water moderator on the Dancoff factor C , the values of C for cylindrical lattices were calculated for both the homogenized case and the case where the clad and water regions were treated separately. The BORAX-V And Hi-C lattices were constructed in the ZPR-VII critical assembly and studied. Calculations were made in terms of the unit-cell approximation and with isotropic reflection at the outer boundary. Values of C for the three-region problems were found to be a little larger than for two-region values. This difference was greater for aluminum-clad fuel (Hi-C lattice) than for stainless steel-clad (BORAX-V lattice), since the cross-section difference between aluminum and water is greater than that between stainless steel and water.

A Fortran program was written for the CDC-3600 to compute L' tables for the resonance-integral calculations where the function $L'(Z', \xi)$ is defined by Adler et al.³ Values were calculated for slab, cylindrical, and spherical geometries to be used in the CDC-3600 version of GAM-I. These values extend and replace the original tables, which were inaccurate for certain values of the parameters.

3. Numerical Estimation of Laplace Transforms

Formulas for estimating Laplace transforms from equally spaced sets of function values were constructed by transforming the interpolation polynomial based on the given values (see Progress Report for September 1964, ANL-6944, p. 56), but numerical studies with the formulas revealed serious inaccuracies. Expressions for the truncation error (the error incurred in approximating the function by a polynomial) have now been derived in the form of the integral of the product of a derivative of f , the function being transformed, and an appropriate kernel function, which is independent of f . These integrals can be used to obtain bounds on the error and to predict the effects of spacing and number of points upon the error. They can also be used in the derivation of optimum formulas for estimating the Laplace transform of functions in a particular class.

If $f(t)$ is continuous and $f'(t)$ is continuous except for a finite number of finite jumps, the best formula for estimating the Laplace transform is

$$Lf(t) \equiv \int_0^{\infty} e^{-st} f(t) dt = \frac{1}{s} \left[(1 - e^{-sh})f(0) + (e^{sh/2} - e^{-sh/2}) \sum_{k=1}^{n-1} e^{-ksh} f(kh) + e^{sh/2} e^{-nsh} f(nh) \right] + R_n(s, h; f)$$

³Adler, F. T., Hinman, G. W., and Nordheim, L. W., The Quantitative Evaluation of Resonance Integrals, Proc. 2nd UN Intl. Conf. on Peaceful Uses of Atomic Energy, Geneva, Switzerland (1958), 16, 155.

The error is bounded by

$$|R_n(s, h; f)| \leq \left(\frac{e^{-sh/2} - 1}{s} \right)^2 \sum_{k=1}^n \left(\max_{(k-1)h < t < kh} |f'(t)| \right) e^{-ksh} \\ + \max_{nh \leq t} |e^{-\sigma t} f'(t)| \frac{e^{-(s-\sigma)nh}}{s(s-\sigma)},$$

where σ is any real number chosen so that $e^{-\sigma t} f'(t)$ is bounded.

It will be observed that the factor $[(e^{sh/2} - 1)/s]^2$ is at least as large as $(h/2)^2$, which limits the allowable interval, while the factor $e^{-(s-\sigma)nh}$ sets a minimum value on the product nhs , and thus on the total number of points. It is also worth noting that the coefficients are all positive, so that serious cancellation errors of the type which caused difficulties with previous formulas are not to be expected.

C. Other Reactor Fuels and Materials Development

1. Nondestructive Testing

a. Ultrasonic Imaging. Comparison tests with the television imaging system and the mechanical-scan, through-transmission ultrasonic system were made on 21 prototype AARR fuel plates. A Lamb wave orientation was employed with the television system, and a normal orientation was used with the mechanical-scan system. No defects were detected by either method.

b. Development of a Neutron-image Intensification System. Additional evaluation of the thermal-neutron image-intensifier tube has been directed at the measurement of the output phosphor brightness and of the response of the intensifier tube to high-energy gamma radiation.

The measurements of output phosphor brightness were made at Juggernaut reactor, with the reactor operating at varying power as a means of changing the neutron intensity striking the tube. The intensifier tube displayed a brightness of approximately 20 ft-Lamberts (2.2 meter-Lamberts) at the peak thermal-neutron intensity of 2.2×10^7 n/cm²-sec. The brightness decreased linearly (on a log-log plot) with neutron intensity; the slope of the line was very close to 1.0. The output brightness of the intensifier tube is, therefore, adequate for visual observation of thermal-neutron intensities of the order of 10^4 n/cm²-sec and, with dark adaption, for use with thermal-neutron intensities at least as low as 10^3 n/cm²-sec. Confirmation of the visibility of the image for an incoming neutron intensity of 10^4 n/cm²-sec was obtained at both Juggernaut and CP-5 reactors. At CP-5, neutron-diffraction patterns having thermal-neutron intensities of the

order of 10^4 n/cm²-sec were visually observed. Studies under dark adaptation are planned.

Gamma sensitivity of the intensifier was also checked. The output phosphor of the tube was observed while a radioactive fuel capsule was positioned adjacent to the input face of the tube. A gamma-radiation intensity of 4000-R/hr at the tube face could easily be overshadowed by an X-ray excitation of the tube. The output brightness due to the 4000-R/hr gamma-radiation intensity was approximately equal to that produced by an incoming thermal-neutron intensity of 5×10^5 n/cm²-sec. The tube would, therefore, be useful for the neutron inspection of reasonably radioactive material.

2. Determination of Elastic Moduli of High-temperature Materials by Ultrasonics

Two factors affect the time of transit of sound in an indexed region: the sound velocity (as a function of the elastic moduli) and the length of the index region. As the temperature changes, most of the change in transit time is due to changes in elastic moduli rather than to change in length. Thus an approximate value of changes in moduli with temperature can be obtained even without considering the length (and density) changes. Values of Young's modulus, Poisson's ratio, and of the shear modulus for a vanadium-20 w/o titanium (TV-20) alloy have been calculated as a function of temperature on this basis; values of Young's modulus compare well with other data for this alloy.⁴ The Young's modulus shows a 10% drop between room temperature and 1100°C, compared with the Young's modulus of stainless steel, which dropped by 35% between room temperature and 925°C. The shear modulus and Poisson's ratio were even more nearly constant for the TV-20 alloy, whereas for stainless steel they both showed significant changes. Information about the thermal expansion of TV-20 up to 950°C has recently become available, and final calculations of moduli will be made by using the new data.

A tungsten sample is being prepared for measurement. A modified flange is being made for an existing very high-temperature furnace so that the tungsten bar and other samples may be run at much higher temperatures than can now be attained in the present furnace.

⁴ Hill, W. H., and B. A. Wilcox, Elevated Temperature Dynamic Moduli of Vanadium, Titanium, and V-Ti Alloys, WADD-TR-60-245 (AD-243479) (1960).

D. Remote Control Engineering Development

1. Electric Master-Slave Manipulator Mark E4

Components for the shoulder joint are being designed, and layout work is proceeding on the elbow joint.

Most of the parts are fabricated for the servo drive units for a pair of manipulators. One-third of the units are nearly complete in their assembly.

2. Servo Studies

Further investigations and development work are being carried out on methods of reducing the number of conductors needed for electric master-slave manipulators. Various methods of mixing velocity and position are being studied.

3. Studies of Higher-performance Manipulator Systems

Studies of parameters and other conditions needed for higher-performance manipulator systems are continuing. The improvements most urgently needed for manipulators are: much better slave "hands" or grasping devices, much better handles, and better static and dynamic force feedback. Various methods of developing these improvements are being investigated and a few mockups of various new types of handles have been made.

4. Viewing Systems

The effect of radiation on the ultraviolet (200-350 $m\mu$) absorption of glasses containing small concentrations of elements commonly occurring as impurities has been investigated. The intrinsic absorption in the ultraviolet (UV) was found to be mainly due to Fe^{3+} and Pb^{2+} in aluminoborate glass and to Fe^{3+} in silicate glass.

Upon irradiation, both Fe^{3+} and Pb^{2+} trap electrons and, as a result, are reduced respectively to Fe^{2+} and Pb^0 (atoms). The Fe^{2+} have absorption bands in a different region of the spectrum than Fe^{3+} ; Pb has some absorption bands in the UV, but these are less intense than those of Pb^{2+} . The result is an increase in UV transmission on irradiation. In glass, generally there are other overlapping bands in the ultraviolet which grow upon irradiation. The resultant radiation-induced absorption is due to the combination of two opposing processes, one tending to increase the absorption, the other tending to decrease it. One observes an increase or decrease in transmission depending on which of the two processes predominates.

In alkali silicate glasses from about 200 to 275 $m\mu$ irradiation produces a decrease in absorption for exposures up to about 4×10^5 R. The

change then reverses and the absorption increases with further exposure. In aluminoborate glasses, there is usually an increase in absorption from the beginning except when there is a high concentration of Fe^{3+} initially present.

E. Heat Engineering

1. Two-phase Flow Studies

a. Void-Fraction - Pressure-drop Facility. The loop has logged approximately 2300 hr at temperatures above 1000°F, the last 800 hr being above 1150°F. Boiling of sodium has been obtained for several hours at a time, highly unstable in the temperature range from 1325 to 1400°F, but increasingly stable up to 1435°F, the maximum operating temperature to date. In this latter range increased boiler power density, flow rate, and loop pressure have significantly decreased the variation of void fraction to a point where average void-fraction and pressure-loss measurements are meaningful. Breakdowns in the instrument voltage regulator and the level-gauge oscillator have postponed higher temperature operation; repairs are nearly complete. A high-speed recorder for the flowmeter outputs is being installed.

2. Boiling Liquid Metal Technology

a. Niobium-1% Zirconium Loop. The construction schedule for the Nb-1% Zr loop is now complete, and a revised, increased cost estimate has been approved and sent to the Pratt & Whitney-CANEL facility. It is anticipated that the entire loop, including the electromagnetic-pump tube, will be received about May 15, 1965.

The loop support and shutter assembly construction drawings have been reviewed and procurement of material necessary to fabricate a major portion of the loop-support components is proceeding. The final design of the "shutter" will not be completed until a discussion on thin gauge tantalum shaping (crimping-dimpling, etc.) is held with a material supplier.

In the assembly of the boiling sodium loop, the installation of the lower portion of the vacuum chamber and the vacuum pumping system is continuing. Rebuilding of the bell jar (upper portion of chamber) is being completed at the vendor's plant.

Instrumentation design is still underway. Price and delivery information on the W-5% Re/W-26% and the chromel-alumel thermocouples have been obtained. An order will be placed as soon as the thermocouple lengths have been checked. Circuitry has been designed to measure accurately the temperature profile at the heater wall. A calibration system is being designed for use prior to installation and during loop operation to determine drift effects.

The instrument panels for measurement of temperature, pressure, and flow rate are under construction.

Cost estimates of pressure transducers as obtained from outside vendors are excessive. These transducers will be fabricated at ANL, according to our own design, and incorporating the use of linear differential transformers.

The 150-kW saturable reactor power supply has been completely checked and found entirely satisfactory for use with the thermal radiation heater.

b. Heater Experiments

(i) Thermal Radiation Heater Experiment: The small-scale radiation heated loop has been operated an additional 100 hr at higher flow rates to check the single-phase heat transfer regions and to observe the approach to boiling. Small flow oscillations accompany the initiation of boiling at low pressures (less than 10 psia). Qualities up to 0.03 have been obtained at the boiler outlet. More extensive data analysis is underway.

(ii) Electron Bombardment Heater: The small-scale electron bombardment heater assembly is nearly ready for operation. A leak was found in the loop system at a thermocouple feedthrough and repairs are underway prior to the sodium fill procedure.

3. General Heat Transfer

a. Heat Transfer in Double-pipe Heat Exchangers

(i) Liquid Metal Cocurrent Turbulent Flow. Calibration procedures continued following modification of the loop (see Progress Report for November 1964, ANL-6977, p. 65). New electrodes for the flowmeters were made by copper plating polished nickel pins, and the output from these new pins was roughly the same as that obtained by the use of copper pins.

Following the completion of the flowmeter-calibration, the loop will be insulated and the first heat balance will be taken.

The two nickel test sections are in the final stages of fabrication.

4. ANL-AMU Program

Other heat engineering experiments, performed as part of a joint program between the Laboratory and the Associated Midwest Universities (AMU), are described below.

a. Two-phase Flow Characteristics during the Complete Vaporization of a Fluid. The electrically heated test section of the forced-circulation loop with visual observation sections has been disassembled in order to install a new heated section. Repair of the original section was attempted but the leakage occurred in sections that were not accessible. While the new section is being assembled, an electrically heated nickel tube (of $1\frac{1}{4}$ -in. diameter) has been installed in the Freon loop in order to proceed with the observation of the phase distributions and pressure drops in the unheated test section.

The flow orifices and the pressure transducers are being calibrated. The Freon loop has been operated, and pressure was sustained which indicated a minimum of Freon leakage. The preliminary check-out of the loop should be completed shortly.

In order to provide information concerning the flow patterns in the unheated section, an X-ray unit with fluoroscope to provide TV monitoring of the flow patterns is to be installed. It is expected that the fluoroscope information will provide the basis for flow-regime transitions in the unheated section. Gamma-ray traverses of the unheated, rectangular section will be made in two planes perpendicular to the flow, as well as along the direction of flow. Thus, the combination of the above results should lead to a quantitative representation of the phase distribution during flow-regime transitions as a function of flow rate system pressure, and type of fluid.

b. Power-to-Void Transfer Functions. The experimental measurement of the void response in a reactor coolant channel as the heat production in the fuel is varied (power-to-void transfer function) has been continued at pressures of 200 and 600 psia. Axial and transverse steady-state void-fraction distributions were also obtained at 200, 300, 400, and 600 psia under operating conditions corresponding to those at which dynamic measurements have been made. For previous results refer to Progress Reports ANL-6923, July 1964; ANL-6936, August 1964; and ANL-6965, October 1964.

The "Kjeller Model"⁵ for the dynamics of coolant channels in boiling water reactors has been programmed for the control data 3600 digital computer employing a step-by-step integration procedure. This theoretical model can be used to generate power-to-void transfer functions even for those cases in which there is an appreciable subcooled boiling region since it incorporates Bowring's⁶ physical model for steam voidage

⁵Solberg, K. O., The Kjeller Model for the Dynamics of Coolant Channels in Boiling Water Reactors, Part I: Theory, EAES Symposium, Studsvik, Sweden (Oct 1963).

⁶Bowring, R. W., Physical Model, Based on Bubble Detachment and Calculation of Steam Voidage in the Subcooled Region of a Heated Channel, HPR-29 (Dec 1962).

in the subcooled boiling region. In this digital program various models⁷⁻⁹ relating void fraction and quality are used to obtain theoretical curves of steady-state void fraction and power-to-void transfer functions which are then compared with the experimental results. To date the validity of the "Kjeller Model" employing various physical models for quality-void relationship appears promising, but a more refined approach is required to handle the effects of the temperature-dependent boundaries separating the three regions: nonboiling, subcooled-boiling, and bulk-boiling. In Figures 7 and 8 are contained a comparison of the "Kjeller Model" with that of Bankoff *et al.*,¹⁰ in which the subcooled boiling region is not accounted for and thus the transitional boundary effects neglected.

Figure 7
Transient Void Amplitude vs. Frequency by
"One Shot" γ -attenuation Technique

Figure 8
Transient Void Phase-lag
Response for Experimental
Conditions Identical with
Figure 7

⁷Marchaterre, J. F., and Hoglund, B. M., Correlation for Two-phase Flow, *Nucleonics*, 20(8) (1962).

⁸Griffith, P., The Prediction of Low-quality Boiling Voids, Paper No. 63-HT-20, Trans. of the ASME, Jour. of Heat Transfer (1963).

⁹Bankoff, S. G., A Variable Density Single-fluid Model for Two-phase Flow with Particular Reference to Steam-Water Flow, ASME, Jour. of Heat Transfer (Nov 1960).

¹⁰Hudson, J. L., Atit, K. M., and Bankoff, S. G., Response of a Boiling Channel to Power or Inlet Flow Modulation, *Chem. Eng. Sci.*, 19, p. 387-402 (1964).

F. Chemical Separations

1. Fluidization and Volatility Separation Processes

a. Recovery of Uranium from Low-enrichment Ceramic Fuels

(i) Laboratory Support Work. Laboratory work in support of the pilot-plant investigation of the fluid-bed fluoride volatility process has continued. Studies were made in 1½- and 2-in.-dia fluidized-bed reactors of the chemistry of the decladding and fluorination reactions employed in reprocessing ceramic UO₂ fuels. Stainless steel-clad UO₂ fuel elements are chemically declad by the high-temperature oxidation of the cladding with gaseous mixtures of HF-O₂ (see Progress Report for September 1964, ANL-6944, pp. 69-70). The oxide fuel reacts with the decladding agent to form a powdered mixture of UO₂F₂ and UF₄. This material is then fluorinated with fluorine gas to recover uranium (and plutonium) as the volatile hexafluoride.

Experimental work in the 1½-in. reactor is primarily directed toward the determination of the effect of decladding products on plutonium retention by the alumina bed after fluorination. The results of two tests in which a mixture of UO₂F₂, PuF₄, fission product fluorides, and the oxide products from the decladding of stainless steel was fluorinated for 20 hr were previously reported (see Progress Report for November 1964, ANL-6977, p. 67). Additional data from these tests have shown that at least 95% of the iron, but only 39% of the chromium, are retained within the reactor after the 20-hr fluorination period. This fact indicates that a volatile chromium compound is formed during the fluorination step. Further evidence for the formation of a volatile species was shown by the presence of chromium in the UF₆ and PuF₆ cold-trap condensers.

Three tests have been performed in the 2-in. fluid-bed reactor to check the equipment and operating procedure prior to runs with plutonium-bearing fuel materials. Other purposes for these experiments were to study the behavior of sintered UO₂ bodies during reaction with HF-O₂ mixtures at 400 to 550°C, and to evaluate the recovery of UF₆ by fluorination of the products of the decladding reaction. In these tests the reaction between UO₂ and HF-O₂ mixtures resulted in the formation of fine particulate material which was readily elutriated from the fluid bed into the upper sections of the reactor. No significant difference in the quantity of fines elutriated from the fluid bed was observed in tests in which the alumina particle size distribution was varied from -40 +100 mesh to -40 +170 mesh. The elutriation of fines from the fluid bed appears to have an adverse effect upon the fluorination reaction, since the temperatures in the upper portions of the reactor were generally lower than those in the fluid bed. Unless the elutriated fines are returned to the fluid bed at frequent intervals, this condition would result in low reaction rates, low

fluorine utilization efficiencies, and, perhaps, incompletely reacted solids remaining at the end of the fluorination step. Further tests may therefore be required to establish an optimum frequency of return of fines to the bed by blowback of the sintered metal filters located at the top of the reactor.

A fluid-bed reactor, designed to reduce the number of inclined surfaces in the disengaging and filter sections on which fine particles might accumulate, is under construction. This reactor, designed for plutonium service, will be installed in the glovebox enclosure. Mockup studies in a plastic column are underway to define more clearly the behavior of fines during fluidization.

(ii) Cleanup of Cell Exhaust Air Contaminated with Plutonium Hexafluoride. Four small-scale experiments were performed under conditions which simulated the release of dilute PuF_6 into ambient air. Two air streams, one containing PuF_6 vapor and the other moisture, were rapidly mixed and passed through a segmented bed of glass spheres. The quantity of plutonium retained on the glass spheres within the various bed segments was used to estimate the rate of hydrolysis of PuF_6 . Data from these experiments indicate that PuF_6 hydrolysis rates are similar to UF_6 hydrolysis rates reported previously (Progress Report for August 1964, ANL-6936, pp. 67-68). This suggests that, in a release of PuF_6 into a glovebox atmosphere, the quantity of plutonium passing through a series of filters will be nearly identical to the quantity of uranium from a UF_6 release under similar conditions.

Previous tests¹¹ have shown that AEC filters are highly efficient in removing the plutonium particulate material obtained from hydrolysis of PuF_6 . In tests involving the release of 1.5 to 25 mg PuF_6 into moist air, the penetration of two AEC filters in series varied from 1 part per million to 2 parts per billion. Therefore, on the basis of the similarity in hydrolysis behavior of UF_6 and PuF_6 , large-scale experiments, in which scale-up effects are evaluated with releases of up to 2 g of hexafluoride, will be made with UF_6 .

b. Recovery of Uranium from Highly Enriched Uranium Alloy Fuels

(i) Pilot-plant Demonstration Runs. Four additional tests have been performed in the pilot plant installed to demonstrate the recovery of uranium from multiplate assemblies of uranium alloy fuels by means of the fluid-bed fluoride volatility process. The process employs three principal steps in which chemical separation of the various fuel constituents is

¹¹Chemical Engineering Division Research Highlights, May 1962 - April 1963, ANL-6766, pp. 87-88.

achieved by selective volatilization of chlorides and fluorides. The initial process step involves the separation of the alloying and cladding material (aluminum or zirconium) from uranium by reaction of the fuel at approximately 360°C with hydrogen chloride while the fuel is submerged in a fluidized bed of inert alumina particles. Separation occurs through the formation of volatile chlorides of zirconium and aluminum, ZrCl_4 and AlCl_3 . Uranium is converted to solid uranium trichloride during hydrochlorination and remains for the most part associated with the fluid-bed material. However, a portion of the uranium chloride is entrained in the gas stream leaving the fluid-bed reactor; therefore, a packed-bed, high-temperature gas filter is employed in series with the reactor to insure retention of the uranium within the system. The volatile chlorides are converted directly to solid oxides by the gas-phase reaction with steam at 350°C in a fluid-bed reactor.

In the second step of the process, the uranium trichloride in the fluid-bed reactor and packed-bed filter is converted to uranium tetrafluoride by reaction with hydrogen fluoride gas. Recovery of uranium as UF_6 is achieved in the third step by the reaction between uranium tetrafluoride and fluorine at temperatures of 250 to 500°C.

Two runs involved processing uranium-aluminum fuel charges (ETR-type fuel) and two runs were performed with uranium-Zircaloy-2 fuel (simulated PWR seed). These runs were made with unirradiated alloy fuel. Operating conditions for processing of the fuel assemblies, shown in Table V, were similar to those reported previously (see Progress Report for July 1964, ANL-6923, pp. 70-72), except that the depth of the alumina bed in the packed-bed filter was increased from 7 to 12 in., and that during hydrochlorination of the aluminum alloy in Run 16 the temperature of the packed-bed filter was increased from 295 to 335°C. The temperature of the filter was maintained at greater than 320°C to prevent condensation of iron chlorides which may have contributed to difficulties associated with bed caking in previous runs (see ANL-6923, p. 72). Aluminum fuel elements contain about 50 g iron.

In Run 15, successive charging of three Zircaloy fuel elements was evaluated as a means for reducing processing time and for increasing HCl reaction rate and utilization efficiency. The processing method involved charging of the fuel elements in a manner such that, at the time of charging, about 60% of the previously charged material had reacted with hydrogen chloride. With use of the successive charging technique, a significant increase in HCl utilization and average reaction rate was observed compared to values obtained in processing single-element charges (Run 13).

Several generalizations can be made from results obtained thus far concerning the reaction of Zircaloy and aluminum alloy fuel assemblies with hydrogen chloride: (1) the reaction rate is virtually independent of temperature in the range studied during intervals in which the

Table V

OPERATING CONDITIONS FOR FLUID-BED FLUORIDE VOLATILITY PROCESSING OF
UNIRRADIATED, URANIUM-ZIRCALOY AND URANIUM-ALUMINUM FUEL SUBASSEMBLIES

Inert Bed Material				
Reactor (6-in. dia):	40 kg Alcoa T-61 alumina (28-100 mesh), 48 in. static bed height			
Packed-bed Filter (9-in. dia):	16 kg Alcoa T-61 alumina (14-48 mesh), 11 to 12 in. bed height			
Pyrohydrolyser (6-in. dia):	10 kg of final bed from previous run with similar fuel, 20 in. static height			
	Run Number			
	12	16	13	15
Run Charge				
Alloy Type	Aluminum	Aluminum	Zircaloy	Zircaloy
Weight (kg)	5.962	6.025	12.720 (9 plates)	20.380 (15 plates) 20.972 (16 plates) 15.038 (11 plates) 56.390 (11 plates)
Uranium (g)	280.5	273.6	117.0 ± 4	546.0 ± 10
Hydrochlorination Data				
Reaction Rate (kg/hr): Avg	0.92	0.87	1.39	2.06
Max	2.8	1.4	3.0	3.3
Reaction Time (hr)	6.5	6.9	9.2	27.4
Utilization Efficiency (%): Avg	37.4	43.2	22.1	33.5
Max	65.0	67.0	53.0	59.0
Temperatures (°C)				
A. Hydrochlorination Step				
Reactor Bed: Avg	360	364	360	372
Max	366	408	388	403
Element Channel: Avg	-	380	370	377
Max	-	520	424	428
Reactor Wall	350	351	347	355
Packed-bed Filter	197	335	346	365
Pyrohydrolyser Bed	326	355	333	355
B: Hydrofluorination Step (2 hr)				
Reactor Bed	360	349	357	360
Filter Bed	350	362	354	353
C. Fluorination Step				
Reactor Bed and Packed-bed Filter				
(1) Reactor Fluidized	250-500°C (7.5 hr)	250-500°C (2.8 hr)	250-500°C (6.2 hr)	250-500°C (2.4 hr)
(2) Reactor Static	500°C (2.0 hr)	500°C (3.0 hr)	500°C (2.1 hr)	500°C (2.1 hr)
Reactant Concentrations (v/o)				
A. HCl in N ₂	91	83.3	85	79
B. HF in N ₂	21	20	35	17.5
C. F ₂ in N ₂				
(1) Reactor Fluidized	5	4.6	4.6	4.6
(2) Reactor Static	68	68	68	68
Gas Velocity in Fluid-bed Reactor (ft/sec)				
A. Hydrochlorination	0.39	0.40	0.44	0.42
B. Hydrofluorination	0.41	0.40	0.38	0.49
C. Fluorination				
(1) Reactor Fluidized	0.39	0.40	0.40	0.40
(2) Reactor Static	0.04	0.04	0.04	0.04

HCl utilization is high; (2) higher reaction rates, with no concomitant decrease in HCl utilization efficiencies, were observed at high fluidizing gas velocities (~ 0.6 ft/sec) than were observed in tests with gas velocities of 0.4 ft/sec; (3) a satisfactory combination of high reaction rate and high HCl utilization can be achieved by use of 80 v/o HCl in N_2 as the feed gas to the reactor; (4) successive charging of fuel elements to the reactor results in a significant increase in HCl utilization and overall reaction rate.

The hydrofluorination step was carried out at 360°C to convert the residual chlorides in the bed to fluorides, thereby minimizing the formation of chlorine fluorides in the fluorination step. Residual chloride concentrations in the bed material after hydrofluorination ranged from 0.005 to 0.05 w/o, which corresponds to 2.0 to 20 g chlorine in the entire system.

The fluorination procedure involved first introducing dilute fluorine (5 v/o in nitrogen) with the reactor bed fluidized (gas velocity in reactor 0.4 ft/sec). The temperature of the fluid bed was then slowly raised from 250 to 500°C over a 2- to 7-hr period. The fluorination was completed by operating with 30 to 70 v/o fluorine under static bed conditions (gas velocity in reactor, 0.04 ft/sec) for an additional 2 to 3 hr at 500°C .

Caking of the filter bed material occurred during the fluorination step of Run 12 as a result of a temperature excursion of 50°C as the fluorine concentration was increased to 50 v/o. This behavior has occurred previously in the processing of aluminum-uranium alloy fuels (see ANL-6923, p. 72) and has been attributed to the accumulation of particulate materials and/or condensed iron chloride on the upper surface of the filter bed.

However, no temperature excursions or caking of the filter bed occurred during the fluorination step of Run 16. Two process modifications were employed in this run to eliminate such occurrences: (1) the filter bed was maintained at 335°C during the hydrochlorination to avoid condensation and accumulation of iron chloride on the filter, and (2) the aluminum fuel element was kept submerged during the hydrochlorination. Aluminum fuel elements normally float in the fluid bed with the upper portion of the element protruding above the surface of the fluidized bed. Under this condition, a considerable quantity of fluid-bed material is ejected from the channels of the element by the high velocity gas stream within the channels. Spouting of particulate material in this manner might result in the entrainment of more material in the gas stream than if the fuel element is submerged. Zircaloy fuel elements do not float under normal operating conditions.

Complete evaluation of the extent of uranium recovery during fluorination will be made when all analytical results are received. Preliminary indications are that fluorination at 500°C with fluorine concentrations greater than 80 v/o would be required to achieve high uranium recovery from the alumina bed material in processing of Zircaloy fuel elements.

2. Chemical-Metallurgical Process Studies

a. Chemistry of Liquid Metals

(i) Thermodynamics of Lanthanum-Cadmium Systems. Measurements of the decomposition pressures of the 1 to 1 compounds of lanthanons and cadmium are being continued to allow computations of the free energy functions for the several intermetallic compounds found in these systems. Preliminary data have been obtained for the decomposition pressures in the two-phase regions HoCd-Cd, TmCd-Cd, YbCd-Cd, and LuCd-Cd. Decomposition pressures for the various systems ranged from 3×10^{-2} to 7×10^{-4} Torr at temperatures between 400 and 600°C.

(ii) Praseodymium-Zinc Galvanic Cell Studies. The thermodynamic functions of the praseodymium-zinc system have been studied by means of a high-temperature galvanic cell. An objective of this study was to obtain data on the thermodynamics of dilute solutions of praseodymium in liquid zinc in order to correlate the extensive data which have been obtained on the partition of praseodymium between fused salts and liquid metal solutions.

The emf of a galvanic cell of the form

was measured as a function of temperature. It has been found that the free energy of formation of PrZn_{11} may be represented by the equation $\Delta G_f^\circ (\text{kcal/mole}) = -85.25 + 42.65 \times 10^{-3} T$ over the temperature range from 696 to 875°K. Provisional values for the free energy of formation, enthalpy of formation, and entropy of formation of PrZn_{11} at 500°C are $\Delta G_f^\circ = -52.3 \text{ kcal/mole}$, $\Delta H_f^\circ = -85.3 \text{ kcal/mole}$, and $\Delta S_f^\circ = -42.7 \text{ cal/deg-mole}$, respectively.

3. General Chemistry and Chemical Engineering

a. Preparation of Uranium Monocarbide. Three fluid-bed hydriding runs on a 100-g uranium scale have been completed in a stainless steel column, using a fluidizing gas velocity of 0.25 ft/sec and a fluidizing time of 5 hr. In a run at 600°C with a 72 v/o H_2 -28 v/o C_3H_8 mixture, UC with a 4.6 w/o carbon content was produced (stoichiometric UC contains 4.8 w/o carbon).

In another run, a 85 v/o H_2 -15 v/o C_3H_8 mixture and a temperature of $700^\circ C$ were used. Uranium monocarbide with a carbon content of 4.8 w/o was produced. The particle size distributions of the UH_3 feed to this run and the UC product were each determined by sieve analysis. There was only a small amount of particle breakdown during the run - for example, the less than $5\text{-}\mu$ fraction increased only from 0.3 w/o to 1.2 w/o - and it was concluded that the fluidizing characteristics of the particles did not change significantly during the run. Examination of the product by X-ray diffraction indicated the major phase to be UC, with possible minor phases of UC_2 and βUH_3 .

In a third run, the objective was to remove excess carbon from hyperstoichiometric UC produced in an earlier run by reacting the UC with 100 v/o H_2 at $700^\circ C$. After 5 hr of this treatment, the original carbon content of 6.54 w/o was reduced to 6.05 w/o. The reduction of the carbon content of hyperstoichiometric UC will also be attempted at 750 and $800^\circ C$.

b. Preparation of Uranium Mononitride. In earlier work (see Progress Report for October 1964, ANL-6965, p. 78), uranium nitride powder was successfully prepared on a laboratory scale by reacting uranium turnings with nitrogen at 200-400 mm Hg pressure and $1050\text{-}1300^\circ C$. In another run, UN has been prepared by reacting 300 g of 1-mm uranium shot with nitrogen. Reaction times and temperatures were: $1/2$ hr at $1050^\circ C$, $1\frac{1}{2}$ hr at $1050\text{-}1330^\circ C$, and 8 hr at $1330^\circ C$. The product particles in the lower portion of the reacted material were found to be stitched together, presumably as a result of partial leakage of uranium from the nitride skin surrounding the pieces of shot. However, one-third of the product was in the form of free-flowing 1-mm UN balls. This method, modified to allow nitriding below the melting point of uranium, appears to be attractive for preparing kilogram quantities of UN for making UN-sodium paste for engineering studies.

Uranium nitride balls from the run just described were used in several wetting experiments. In one experiment, 25 g of UN was wetted when it was contacted with sodium at $550^\circ C$ in a stainless steel vessel. After the experiment, the mixture was removed from the reaction vessel and was observed while it was heated in the range from 100 to $200^\circ C$. The UN particles were wetted by sodium over the entire temperature range. In several other experiments, wetting of UN by sodium could not be initiated at $250^\circ C$ or lower.

In an experiment to determine the effects of the oxygen content of the UN on wetting, 24 g of UN balls were oxidized for 10 min at about $300^\circ C$. The weight gain was equivalent to the formation of about 4% UO_2 . Subsequently, when the partially oxidized shot was contacted with hydrogen at $900^\circ C$, no pressure drop due to reaction was noticed. This material could be wetted by sodium at $300^\circ C$ and did not dewet on cooling. Apparently, UN particles can be wetted easily by sodium, and any oxidation of

the UN particles in the paste will not cause them to be dewetted. There are no apparent difficulties in obtaining a stable UN-sodium paste; however, additional data are required to reliably evaluate the UN-sodium wetting problem.

G. Plutonium Recycle Program

1. EBWR Facility

a. Irradiation Surveillance. Approximately 135 standard and front-chamfered Charpy impact bars were prepared from the quarter plate-thickness regions of the original 4-in.-thick EBWR SA 212B steel. Seventy-five front chamfered bars were loaded into new irradiation capsules for irradiation within the EBWR during the forthcoming plutonium Recycle operations. Encapsulation operations, which includes final capsule closure welding operations under a helium filling pressure of 500 psig, are approximately 75% completed. After helium leak testing, utilizing the encapsulated gas itself, the five cannisters will be ready for insertion into the reactor during the loading of plutonium fuel.

Two capsules of steel specimens, one for each irradiation thimble, will be irradiated within the existing "bare" and 2 w/o boron shielded thimbles. The fifth capsule will be placed in the super steam dome portion of the reactor vessel to monitor the thermal effects during the irradiation cycle.

The remaining square-shouldered and front-chamfered Charpy Vee-specimens will be utilized in determining (1) the initial impact-transition temperature, (2) the dispersion of data (from the breaks of three bars at each temperature), and (3) the effects of the front chamfering of the Charpy bars.

b. Field Inspection. The Materials Testing Laboratory was engaged to inspect nondestructively portions of the primary equipment, piping, and auxiliary equipment in advance of plutonium Recycle operations. Although no failures in any of this equipment have been experienced, the following examinations by applicable ultrasonic, dye-penetrant, magnetic-particle, and radiographic techniques are being carried out.

a) Vessel Closure Forging: inter-bolt ligaments and bolt studs.

b) Vessel Closure Cover: inter-bolt ligaments.

c) Six-in. Carbon Steel Piping: 24-in. straight lengths of Kanigen-plated upstream and downstream piping welded to elbows; elbows in the turbine supply and safety valve lines; elbows and piping at high-pressure steam safety valves; shop-fabricated transition welds between SS 304 and carbon steel above the SS 304 cooler-dryer.

d) Four Feedwater Filters: two roll-bonded stainless steel-to-carbon steel shells; two all-stainless-steel-Type 304 shells; transition stainless-to-carbon steel welds.

e) Startup Heater Primary Piping: substantially all of the SS 304 piping and welds between reactor face and heater.

f) Boric Acid Injection Piping: stressed piping loop at reactor face containing many welded elbows, check valve, control valve, and structural attachments.

g) Ion Exchanger Piping: 24-in. length of 2-in. SS 304 piping between reactor and IX system containing water frozen by dry ice in acetone to permit maintenance on IX and prevent drainage of reactor containing fuel.

h) Carbon Steel Condensate Piping: feedwater pump suction line from turbine condenser.

In addition to the examination of the selected reactor facility regions, fillet strength welds on the air lock doors of the containment shell will also be magnafluxed.

2. Physics Calculations

In the EBWR studies, homogenized cross sections were determined from the spectral information furnished by the SOFOCATE calculations. One-group DSN-8 calculations were used to supply flux-weighting information to the space-independent SOFOCATE code. Since the flux-weighting and spectrum calculations are performed in two separate problems, this procedure at best involves an iterative procedure. In addition, the use of the one-dimensional DSN-8 code requires solving a series of problems. For example, the flux-weighting factors in EBWR are first evaluated for the fuel pins in the interior of the fuel element in a cylindrical cell. The results of this pin-cell calculation are used in the study of a slab-geometry cell to provide interelement flux-peaking information. Finally, a large, 9-element cell is studied in slab geometry to include the effects of water gaps and control rod guide structure.

The one-dimensional THERMOS code which is now available combines both flux weighting and spectral averaging in a single multigroup calculation. Since THERMOS is also a one-dimensional code, it would still be necessary to run a series of problems to cope with the heterogeneous details of the reactor under investigation.

To facilitate the EBWR plutonium Recycle studies, the THERMOS code is being modified to handle automatically the series of problems required to establish the flux-weighting factors; the user indicates which regions for the problem are to be homogenized for input to the next more geometrically complicated problem. The THERMOS code will be programmed to evaluate automatically the correct energy dependent flux

weighting factors and generate a spatially homogeneous multigroup cross section for the system.

3. Plutonium Recycle Fuel

The vibratory compaction ("Vipac") process used at Hanford to load the PuO_2 - UO_2 fuel rods for the central zone is based upon vertical excitation of the tube. Each tube is clamped at the lower end plug to the vibrator head. The vibration forces are transmitted to the tube through the end plug.

Equipment being used at Hanford can not be programmed to obtain the most desirable frequency response for optimum loading. It was noted that vibrations normal to the driving force sometimes occurred in a tube. Under certain adverse conditions harmonics of these transverse vibrations occurred at the lower end plug-to-tube weld. It was believed that these forces might initiate an internal crack at this location.

Along with the overall development of the Vipac process, inspection equipment was being developed to verify the presence of any possible defects. Existing ultrasonic and radiographic inspections proved inconclusive for examining the end plug-to-tube weld section. Consequently an ultrasonic shear-wave technique using a focused transducer with a 9-mil spot was developed at Hanford specifically to inspect this section of each loaded rod. This inspection method is capable of detecting any transverse discontinuities greater than 1 mil in depth in either the tube wall or the body of the end plug.

Loaded fuel rods at Hanford were inspected by this improved technique and a number of defects were indicated. These defects were confirmed by destructive examination. When this inspection method was applied to all 1350 rods that had been delivered to ANL, 468 rods were found with defects greater than $51\ \mu$ in depth at the end plug-to-tube weld junction. These rods were returned to Hanford for replacement of the lower end plug closure. Replacement of the end plug and rewelding of the lower end closure is being done without removing any of the fuel material and without contaminating the fuel jacket or weld area.

The 882 acceptable fuel rods at ANL are being assembled into fuel elements. This amount is sufficient to load 24 fuel elements with 36 fuel rods per element. Acceptable components have been delivered to ANL to complete the assembly of 24 elements. These elements will be available for loading into the reactor in January.

In addition to the regular PuO_2 - UO_2 fuel for the central zone, 25 additional test fuel rods will be used for physics measurements on the core. The following test fuel rods are completed and are being evaluated by the Physics Group at Hanford:

- (a) 5 rods containing Al-3.35 w/o Pu alloy fuel jacketed in Zircaloy-2 tubes; the plutonium has an isotopic content of 8% Pu^{240} ;
- (b) 5 similar rods in which the plutonium has an isotopic content of 26% Pu^{240} .

The following test fuel rods are being processed at Hanford:

- (a) 5 rods containing natural UO_2 fuel;
- (b) 5 rods containing regular Nupac and Vipac UO_2 -1.5 w/o PuO_2 , in which the plutonium has an isotopic content of 20% Pu^{240} ;
- (c) 5 similar rods in which the plutonium has an isotopic content of 26% Pu^{240} .

All of the 6% enriched fuel rods for the shim-zone loading have been fabricated, corrosion tested, and inspected. This fuel is being assembled into fuel elements at United Nuclear Corporation as the required Zircaloy-2 fuel cans become available. Approximately 12 fuel cans are now available for these elements.

The production of Zircaloy-2 fuel cans is proceeding at a rate of 10 cans per week. The first shipment of 11 fuel cans for the central-type plutonium fuel elements was received.

III. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

1. Fuel and Core Design

Fuel specimens produced by Battelle Memorial Institute (BMI) and the Martin Company under their respective development contracts have been delivered to the Phillips Company for irradiation in the G-12 loop of ETR.

Photomicrographs of the BMI roll-compacted plates made with plasma-sprayed UO_2 show that there is appreciable stringering and crushing of this low-density oxide as compared with the negligible crushing and stringering experienced with the high-density Sol-Gel UO_2 . However, the amount of crushing and stringering observed in the irradiation samples is significantly reduced from that observed in the Phase I work with other spherical UO_2 . It is hoped that this first set of irradiation tests will help to establish lower values of parameters for oxide density and crushing strength as these bear on the relationship between the cermet microstructure and irradiation stamina (i.e., plate swelling).

The Martin irradiation samples, made by the coated-particle, modified PM-1 core-3 process with the plasma-sprayed UO_2 , showed negligible crushing and stringering of the oxide during fabrication. It is expected that these plates will be used as a reference for determining the relationship between change in physical properties and accumulated irradiation at high dosages.

The twenty-one prototype AARR fuel plates from Martin and Battelle were inspected by conventional ultrasonic through-transmission technique. A test frequency of 5 Mc was used. Only the clad-core areas of the plates were tested. Inspection of clad-clad areas was performed on the ultrasonic imaging system. No nonbonds, laminations, excess particle concentration, or large particles were detected.

2. Heat Transfer

An experimental heat transfer program has been initiated in order to provide information regarding the maximum power level at which flow reversal, fuel removal, and certain accidents can be tolerated in AARR. This study is required for the final safety analysis and will help minimize the reactor downtime for refueling by eliminating unnecessary conservatism.

The first tests, which will be concerned primarily with flow reversal, will utilize a test section consisting of four parallel round tube heated flow channels, 0.080 in. ID. One of the tubes will produce about 3 times the heat of each of the other three. In the final flow reversal tests, rectangular test sections with a similar power variation will be used.

The test procedure for flow reversal will involve forced-circulation downflow at a given power level and flow conditions typifying the situation involving removal of decay heat. Pump power will be cut off, the flow will decrease to zero, and natural-circulation upflow will have to establish itself. The effect of system depressurization to atmospheric pressure will be studied together with forced flow reversal by the use of valving and the pump used for downflow. The purpose of these tests will be to determine the conditions under which flow reversal can be safely achieved.

An emergency cooling heat transfer program will follow the flow-reversal tests to study experimentally the operating power limits of the AARR core during the refueling process and during emergency conditions or accidents.

A transient heat transfer test program to establish a method of predicting fuel melting in the AARR during transients has been started. The present method of predicting this failure is based on an analog model which uses steady-state data for predicting transient behavior. This approach is conservative, and the tests are intended to investigate and possibly reduce the degree of conservatism involved.

The approach at this time is to develop a calculation technique with a model which will predict when test-section failure occurs during a simulated reactor power excursion. This model will be used for predicting melting during AARR transients with rectangular flow channels. The time-dependent nature of the calculated power for the AARR will be simulated. Electrically heated, parallel, 0.080-in.-ID, round-tube test sections will be used instead of rectangular test sections, because tubes may be actually melted in the tests. Later, rectangular test sections typical of AARR may be used, but in a limited number of cases.

An analog-computer investigation has been initiated to determine the effect of transients on the nonuniform heat generation in an AARR fuel plate. This study (by determining the time-dependent variation of the heat fluxes on each side of the plate and the temperature distribution in the fuel plate) is intended to verify if the hot-spot allowance made for the effect of the front-to-back ratio is adequate.

A simplified one-dimensional model has been proposed that considers a single particular horizontal region of one fuel plate, with coolant on each side. The coolant is pressurized water and is capable of boiling. In the analog solution the fuel plate is considered to have seven homogeneous regions, the first and seventh being cladding and the five inner regions being fuel.

Initially, several simplifying assumptions will be made in the simulation. The densities and specific heats of the fuel and clad will be assumed constant and equal. Prior to boiling, the film coefficient of heat transfer between the clad and the water will be constant. Boiling will be assumed to occur when the clad-coolant interface temperature reaches a certain value, and boiling heat transfer will be simulated by holding the clad temperature constant at this value.

The parameters to be varied are rate of heat generation in the fuel; front-to-back ratio; densities, specific heats and thermal conductivities of the fuel and cladding; coolant channel sizes; coolant flow velocities in the two channels, and thickness of cladding.

The problem has been set up for analog solution and is currently under study.

3. Transient Analysis

A series of problems have been solved by analog computer which describe the transient response of the AARR core to a variety of reactivity disturbances introduced at equilibrium core power levels of 100 MW and 1 MW. Conditions were i) a fuel plate and coolant channel wherein average core power is generated and through which inherent reactivity compensation is determined, and ii) a skewed power channel wherein the maximum of the radial power distribution and the effect of axial power skewing due to partial control rod insertion is applied; this channel is therefore considered the channel most likely to experience thermal failure. The effects of engineering-design safety factors are included in the skewed channel, and prediction of fuel-plate burnout is made by comparing transient hot-spot conditions therein with the Zenkevich-Subbotin steady-state burnout correlation.

Reactivity was taken as inserted in steps and ramps with magnitudes ranging to well above prompt critical. Reactivity compensation is effected only through inherent feedback mechanisms of the reactor, in order to establish excursive self-limiting capabilities and characteristics during a time following a disturbance which is less than the time normally required for conventional safety systems to be effective. Input reactivity ramps were chosen such that the total insertion occurred in 33 msec, which is the coolant transit time through the core.

Preliminary examination of the results has indicated the following:

- i. The power response is stable, i.e., there is no evidence of power oscillations following the initial power burst.
- ii. The effect of decreased initial power is to decrease the magnitude of peak power obtained during the short-term response.

iii. For the same total input reactivity, the effect of ramp rather than step insertion is to reduce peak power by factors ranging from 0.8 for \$0.75 reactivity to 0.5 for \$1.50 reactivity when initial power is 100 MW. No appreciable difference between the power response to steps and ramps is observed when initial power is 1 MW.

iv. For the step reactivity insertions, the short-term power peak is reached within 13 msec following application of the step when initial power is 100 MW; when initial power is 1 MW and the reactivity exceeds prompt critical, the time of occurrence of the short-term power peak is increased to a maximum of 60 msec for a \$1.00 step insertion.

v. For the ramp reactivity insertions, the short-term power peak is reached within 6 msec following termination of the ramp (33-msec duration) when initial power is 100 MW; when initial power is 1 MW and the reactivity exceeds prompt critical, this delay time is increased to a maximum of 52 msec following termination of a \$1.00 ramp.

vi. Reactivity compensation through steam voids is not significant for sub-prompt critical reactivity insertions.

vii. The threshold value of input reactivity which causes thermal failure in the skewed channel is approximately \$0.90 step insertion at initial reactor power of 100 MW. Burnout heat flux at the hot spot occurs at approximately 40 msec following application of the step. For ramp insertions the corresponding threshold value is extended somewhat, but lies in the range from \$0.90 to \$1.00. For the \$1.00 ramp burnout, heat flux at the hot spot occurs at approximately 14 msec following termination of the ramp.

For transients initiated at 1 MW, the threshold value of input reactivity for hot-spot thermal failure lies in the range from \$1.10 to \$1.25 for both step and ramp insertions. Burnout is reached at approximately 48 msec following application of a \$1.25 step and 40 msec following termination of a \$1.25 ramp.

4. Critical Experiment

Experimentation was completed with the 315-fuel-foil loading (see Progress Report for October 1964, ANL-6965, p. 85).

The core was reassembled with 615 fuel foils and 420 stainless steel strips containing 1 w/o natural boron. The loading totals approximately 31.4 kg U^{235} , with an atom ratio of 1 B^{10} to 32 U^{235} . Core geometry is the same as in the 315-fuel-foil loading, and the metal-to-water ratio remains 1:1 in the fuel region. This loading was the best available with the amount of fuel on hand; a 30% greater fuel loading would have been preferred.

Excess reactivity was measured to be $3\text{--}2\frac{2}{3}\%$; criticality occurred with two control blades fully inserted, nine control and safety blades fully withdrawn, and the remaining control blade about two-thirds withdrawn. An unshadowed blade (with compensation by a blade located 120° away from the blade under test) controlled $1\text{--}1\frac{1}{2}\%$ in reactivity, as measured by rising-period calibrations.

Measurements planned in this core include i) Rossi-alpha, ii) danger coefficients of core and control element components, iii) void worth, and iv) foil activations. Reloading with greater fuel density will be governed to a considerable extent by the delivery of fuel. The supplier's contract expires in January 1965 and an extension appears to be necessary to obtain completion of the fuel order.

B. Magnetohydrodynamics (MHD)

1. MHD Power Generation - Jet Pump Studies

It appears that a K-Hg eutectic alloy, if used in a cycle, would allow the operation of a secondary vaporizing loop to supply the jet pump. Operation of the primary system in the vicinity of 1500°F would permit the use of a SNAP type reactor using sodium as the primary coolant. In conjunction with this, the fuel could be UO_2 or UC operating at the temperatures considered.

Examination of the equations for jet pump operation and comparing them to the cycle on a T-S diagram indicate that a jet pump cycle will fall considerably short of a Rankine cycle. The reason for this is inherent in the need for the production of pumping pressure coupled with the loss mechanisms in the combiner. Final efficiencies depend upon how effectively conversion can be performed under optimized conditions in a constant-pressure combiner.

C. Regenerative Emf Cells

1. Lithium Hydride Cell

Studies are continuing of the determination of thermodynamic properties of the lithium hydride cell. These data are important in evaluating the cell as a direct energy-conversion device. From the calculations of activities of LiH in fused salt electrolytes, for example, it is expected that the regeneration temperature of the system can be established. Previously, data have been reported for the activities of LiH at various concentrations in LiCl and in LiCl-LiF (see Progress Report for September 1964, ANL-6944, p. 81). The present experiments have extended this study to other concentrations of LiH in LiCl. From emf data at 610°C , the following activities of LiH in LiCl have been calculated: 0.79 for 30.2 m/o LiH, 0.15 for 5.0 m/o LiH, and 0.08 for 2.5 m/o LiH. The data

indicate that at these concentrations the activities continue to exhibit strong positive deviations from ideality. Calculations have also been made for emf-temperature coefficients for cells having LiCl electrolytes unsaturated in LiH. In the range from 1 to 10 m/o LiH, $(\partial E/\partial T)_P$ is ~ 3 to 5×10^{-4} V/°C; for 15 m/o and 30 m/o LiH, $(\partial E/\partial T)_P$ is $\sim 6 \times 10^{-4}$ V/°C.

2. Bimetallic Cells

a. Sodium-Bismuth Cell Studies. In a previous experiment (see Progress Report for November 1964, ANL-6977, p. 79) the solubility of Na_3Bi in the ternary eutectic 53.2 m/o NaI-31.6 m/o NaCl-15.2 m/o NaF was determined by contacting the electrolyte with solid Na_3Bi . The solubility ranged from 1.65 m/o* at 552°C to 8.5 m/o at 839°C, and the average sodium-to-bismuth ratio was 2.97 ± 0.03 .

In the present studies, determinations were made of the solubility of sodium and bismuth in the electrolyte above a 45 a/o Bi-55 a/o Na alloy. Values for the solubility of bismuth ranged from 0.7 m/o at 550°C to ~ 1.0 at 750°C. The sodium to bismuth ratios in the electrolyte were ~ 2.8 . These data indicate that the concentrations of sodium and bismuth in the electrolyte are lower when the electrolyte is in contact with an alloy unsaturated in Na_3Bi , as in the present case, than when the electrolyte is in contact with solid Na_3Bi . The conditions of this experiment more nearly correspond to those of an operating cell than do the conditions in which the alloy is saturated with Na_3Bi .

*Recalculated value; reported as 1.2 m/o in ANL-6977.

IV. NUCLEAR SAFETY

A. Thermal Reactor Safety Studies

1. Metal-Water Reactions

a. High-temperature, High-pressure Furnace. Studies of the effects of pressure on metal-water reactions have continued. An 8-in.-long, Zircaloy-2-clad, UO_2 -core simulated fuel rod was initially held in the low-temperature zone of the high-pressure, high-temperature furnace (peak temperature, $\sim 900^\circ\text{C}$) while the air was displaced by steam (steam pressure, 15 psig). The sample was then raised into the hot zone in 8 min. The rod remained in this position (upper-end temperature, 1500°C ; lower-end temperature, 900°C) for about 3 hr. The amount of hydrogen collected corresponded to a reaction of 73.6% of the Zircaloy-2 cladding, most of the reaction occurring during the first 2 hr.

Figure 9. Zircaloy-2-clad, UO_2 -core Simulated Fuel Rod before and after Exposure to Steam at 1500°C and 15 psig

The fuel rod, after removal from the furnace, and an unreacted fuel rod are shown in Figure 9. The upper end of the rod was converted to white ZrO_2 , and the lower (cooler) end retained a metallic appearance. The oxidized cladding kept its original shape and, except for some cracks, retained considerable strength. The original diameter of the rod, 0.426 in., increased upon oxidation to between 0.458 and 0.486 in.

b. Scale-up Experiments in TREAT. Studies have continued of metal-water reactions under conditions simulating a power excursion incident in a typical bundle of reactor fuel elements in a water-cooled reactor. Additional data for the two runs described in the Progress Report for November 1964, ANL-6977, p. 83, are shown in Table VI. In experiment CEN 196-S a fission energy input of

124 cal/g U gave 10.7% uranium-water reaction. The fuel pins were melted into two large masses plus some fragments and particles. In the more energetic transient CEN 197-S, a fission energy input of 302 cal/g U gave 56% reaction and complete conversion of the fuel into relatively fine particles. The reactor-power time trace for the second experiment CEN 197-S indicated that during the transient a change in the reactivity of the TREAT reactor occurred. It is believed that this perturbation was caused by the change in geometry of the test subassembly during meltdown.

Table VI. Treat Metal-Water Scale-up Experiments with Uranium

	TREAT - CEN Transient No.	
	196-S	197-S
Reactor Power, MW-sec	91	222
Reactor Period, msec	157	140
Fission Energy Input, cal/g U	124	302
% of U Reacted with Water	10.7	56
Appearance of Cluster after Transient	Melted into two large irregularly shaped masses plus some fragments and pieces	Complete meltdown and conversion into relatively fine particles

B. Fast Reactor Safety Studies

Two of the most important characteristics influencing the safety of fast power reactors are the large quantity of enriched fuel and the short prompt-neutron lifetime. Because of these factors, it is possible to postulate a class of "meltdown and reassembly" accidents occurring in three stages as follows: (1) some operating abnormality occurs, which produces some degree of fuel failure; (2) material movements result, which increases the reactivity of the assembly; and (3) the accident is terminated by a destructive nuclear burst. The mechanisms involved in fuel failure, reassembly of fuel into supercritical mass, coolant effects, etc., are complex and not well-understood. Consequently, these mechanisms are being studied experimentally and analytically in a program centering on tests being performed in the TREAT reactor.

1. Transients with Thorium-Uranium Fuel

Thorium-uranium metallic alloy is a potential fuel for fast reactors because of superior dimensional stability to relatively high burnup and high operating temperature in comparison with other metallic fuels. Further, fast reactors using the U-Th cycle may be designed to have a more favorable sodium void reactivity coefficient than reactors using the Pu-U cycle.

Initial transient tests were previously performed on short Th-U pins which were sodium bonded to either 304 SS or Nb-1% Zr clad. Although fuel temperatures exceeded 1000°C, no sample failures occurred (see Progress Report for October 1964, ANL-6955, pp. 9-10). As an extension of these tests into a range of conditions of interest for accident analyses, a series of meltdown experiments have also been completed with longer fuel pins in TREAT. Transients were planned with the expectation that phenomena such as cladding penetration by molten alloy, cladding failure by internal gas pressure, variations in cladding attack caused by fuel phase changes, and cladding melting might be encountered. The mechanisms of failure and fuel movement by Th-U samples were expected to be more complex than those encountered with EBR-II Mark-I fuel.

Transient irradiation conditions for the six meltdown samples are given in Table VII. Test settings for samples 5 and 6 were not specified until samples 1 through 4 had been subjected to TREAT transients, and the threshold of failure was ascertained. Specifications for these final samples of the series were adjusted to insure complete fuel pin failure.

Table VII. Experimental Test Conditions for U-Th Alloy Samples

Sample No.	$k_{ex}(0)$	Integrated TREAT Power, MW-sec	Estimated Maximum Clad Temperature, °C
1	1.08	120	597
1	1.4	210	1190
2	1.5	237	1290
3	1.6	276	1400
4	0.8*	317	1300
5	1.88	280	~1400
6	1.82	312	~1400

*Constant power burst of 15-sec duration.

All samples of the series have been received from TREAT. Post-mortem examination of the experimental residue and analysis of the high-speed color motion pictures taken of the sample behavior are under way.

2. Calculation of Transient Temperatures

The ARGUS transient heat transfer code has been revised to provide a special set of equations for calculating heat transfer to a flowing coolant under boiling conditions. Transfer to the coolant is calculated by means of the equation

$$q = hA(T_1 - T_2), \quad (1)$$

where q is the power entering the coolant, h is the heat transfer coefficient, A is the heat transfer surface area, T_1 is cladding surface temperature, and T_2 is the coolant temperature.

With the new option, when $T_1 \geq T^*$ (an input temperature reference), h is calculated by

$$h = \frac{k}{De} [H_3(Re)^{a_2} (Pr)^{b_2} + H_4] + \frac{H_5(T_1 - \psi_s)^{a_3} + H_6}{T_1 - T_2}, \quad (2)$$

where k is coolant conductivity; De is equivalent diameter; Re is Reynolds number; Pr is Prandtl number; ψ_s is coolant saturation temperature; and H_3 , H_4 , H_5 , H_6 , a_2 , a_3 , and b_2 are input constants. The first term of Eq. (2) is of the standard form for forced-convection heat transfer. The second term, when substituted into Eq. (1), gives a power transfer q' due to boiling the form

$$q' = H_5(T_1 - \psi_s)^{a_3} + H_6. \quad (3)$$

Together, both terms provide a smooth transition region between the two regimes defined by the two terms of Eq. (2).

3. Samples for Fission Product Release Experiments

Three half-lengths Fermi 1 fuel pins, consisting of zirconium-clad uranium-10 w/o molybdenum alloy, have been cut into segments about 8-cm long and prepared for shipment to Oak Ridge National Laboratory. The extent of fission gas release upon heating will be determined. The specimens had been previously irradiated to about 0.3 a/o burnup and allowed to decay for 3 yr. The information from this experiment is expected to be useful in obtaining a license to ship irradiated fuel subassemblies by cask from the Fermi reactor site to a recovery plant. The tests will complement previous out-of-pile electrical meltdown experiments performed for APDA, and ANL in-pile transient meltdown experiments in TREAT.

4. Calculation of Physical Properties of Reactor Materials at High Temperatures

The behavior of substances near their thermodynamic critical point is of considerable interest to reactor safety as well as in the understanding of the fundamental properties of matter. Because of the large compressibilities in the region of the critical point, the details of the behavior of density versus energy or temperature in this region will strongly determine the details of the pressure versus energy or temperature relationships of

reactor materials. In particular, in the region where the material completely fills the available volume, the pressure increases quite rapidly with energy, and the large adiabatic compressibility must be taken into account in calculating shock wave phenomena.

Rowlinson,¹² in a discussion of the shape of the pressure-volume (pv) curve in the vicinity of the critical point, showed that at any temperature below and near the critical the difference between the critical density ρ_c and the vapor density ρ_v or the liquid density ρ_L would be a linear function of the difference between the critical temperature T_c and the temperature T on a logarithmic scale. This can be expressed by T approaching T_c as

$$(\rho_c - \rho_v) = A(T_c - T)^{\beta_1} \quad (1)$$

and

$$(\rho_L - \rho_c) = B(T_c - T)^{\beta_2}. \quad (2)$$

In terms of reduced coordinates,

$$1 - \rho_{vR} = C(1 - T_R)^{\beta_1} \quad (3)$$

and

$$\rho_{LR} - 1 = D(1 - T_R)^{\beta_2}. \quad (4)$$

These relations are valid only near the critical point and are strongly similar to the ideal-fluid equations of Guggenheim,¹³ who gave the values $1/3$ to β_1 and β_2 , and found a good fit to the liquid and vapor densities of the inert gases, N_2 , and O_2 .

Recent studies by Fisher^{14,15} on the lattice model of a gas, which shows the features of two-phase behavior, has indicated that the value of β is probably $5/16$.

Michels, Blaise, and Michels¹⁶ found that the data for CO_2 could be fitted with greater accuracy if $\beta = 0.357$. Weinberger and Schneider,¹⁷ who took special precautions for the effects of gravity in their measurements on xenon, found $\beta = 0.345 \pm 0.015$ for $(1 - T_R)$ as low as 3×10^{-5} and $(\rho_{LR} - 1) = 0.08$.

¹²Rowlinson, J. S., *Liquids and Liquid Mixtures*, Butterworth's Scientific Publications, Ltd., London (1959), Ch. 5.

¹³Guggenheim, E. A., *J. Chem. Phys.*, **13**, 253 (1947).

¹⁴Essam, J. W., and Fisher, M. E., *J. Chem. Phys.*, **38**, 802 (1963).

¹⁵Fisher, M. E., *J. Math. Phys.*, **5**, 944 (1964).

¹⁶Michels, A., Blaise, B., and Michels, C., *Proc. Roy. Soc. (London)*, **A160**, 358 (1937).

¹⁷Weinberger, M. A., and Schneider, W. G., *Can. J. Chem.*, **30**, 442 (1952).

In order to resolve the discrepancy between theoretical and experimental values of β , data for a wide variety of fluids were plotted and curves fitted to them. The fluids studied included a quantum fluid, Helium-4, a typical organic material, Freon-12, water (of possible interest to water reactor safety), a fused salt BiCl_3 , and two metals (Hg and Rb). The results are given in Table VIII.

Table VIII. Reduced Density Functions in the Critical Region

Substance	Vapor		Liquid		$(1 - T_R)_{\min}$
	β_1	C	β_2	D	
Helium-4	0.414	0.627	0.414	0.637	0.0069
Freon-12	0.445	2.80	0.300	2.30	0.0023
Water	0.346	2.04	0.346	2.44	0.000023
BiCl_3	0.333	1.76	0.356	2.24	0.0059
Hg	0.323	1.425	0.388	1.120	0.046
Rb	0.379	1.71	0.536	3.22	0.0478

These preliminary results did not use least squares and statistical techniques for rejection of experimental points, but are presented to give the progress in this study.

Except for water and helium, the exponents are different for vapor and liquid and the liquid density function has generally a greater slope. Except for Rb and He the coefficients C and D are almost equal and are in the range from 1 to 3. The effect of small changes in the critical constants on these exponents will be examined.

C. TREAT

1. Large TREAT Loop

Reconditioning of the compound bellows assemblies is under way at ANL Central Shops. New end pieces are being fabricated for the new assembly. The welds are radiographed after completion of a joint to assure acceptability.

The vendor selected to supply the single-unit, 3-ply bellows has been unable to deliver an adequate product. ANL will buy the bellows and end pieces and final welding assembly will be completed at ANL.

Electronic analog calculations are being made to investigate the transmittal pressure produced during a pulse in the loop to components in the upstream channel.

Equations describing the motion of the sodium were derived by applying Newton's second law of motion, relating the time-dependent

unbalanced forces to the rate of change of momentum. The contributions to the total unbalanced forces, including pump head, hydraulic resistance, pressure forces, and sodium weight, were written in terms of the mass and velocity of the sodium. The equations were then transformed to a stationary plane of reference by substituting the velocity perturbation, $V_i(t) = V_i^1(t) - V_i^1(0)$, for the sodium velocity, $V_i^1(t)$. This permitted the use of zero initial conditions for pressure and velocity.

The resulting equations are

$$\frac{dV_A(t)}{dt} = K_1 V_A(t) + K_2 V_A^2(t) + \frac{K_3 P_0}{K_4 + K_5 \int_0^t V_A(t) dt + K_6 \int_0^t V_B(t) dt};$$

$$\frac{dV_B(t)}{dt} = K_7 + \frac{K_8 P_0}{K_4 + K_5 \int_0^t V_A(t) dt + K_6 \int_0^t V_B(t) dt}$$

$$+ K_9 V_B(t) + K_{10} V_B^2(t) + \frac{K_{11}}{K_{12} + K_6 \int_0^t V_B(t) dt},$$

where V_A is the transient-produced velocity change of the sodium entering the test section, V_B is the velocity change of the sodium above the fuel, P_0 is the initial transient pressure, and the K 's are the constants that depend upon conditions of the problem.

The pressure pulses studied have ranged from 250 to 750 psi, with pressure decay limited to that produced by isothermal expansion of the sodium vapor as the two sodium masses separate. Preliminary results indicate that the sodium above the fuel pins will reach an appreciable upward velocity as the result of a pressure pulse and that the sodium entering the fuel subassembly may be decelerated sufficiently to attain a downward (negative) velocity. The downward displacement of the inlet sodium, during the duration of a pressure pulse, will be small because the displacement of the sodium above the fuel, and thus the decrease of the transient pressure, occur very rapidly.

These findings are consistent with early pressure pulse calculations and verify that excessive pressures of the water hammer type will not be encountered. However, momentary stoppage of the sodium flow will result in tripping of the electromagnetic pump power interlock. This may require remote re-energizing of the pump rectifier.

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