ANL-6328
Reactor Technology
(TID-4500, 16th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

February 1961

N. Hilberry, Laboratory Director

Division	Director
Chemical Engineering	S. Lawroski
Idaho	M. Novick
Metallurgy	F. G. Foote
Reactor Engineering	B. I. Spinrad
Remote Control	R. C. Goertz

Report coordinated by R. M. Adams

Issued March 15, 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

DISCLAIMER

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.

Previous reports issued
in this series:

August, 1960	ANL-6215
September, 1960	ANL-6234
October, 1960	ANL-6253
November, 1960	ANL-6269
December, 1960	ANL-6295
January, 1961	ANL-6307

TABLE OF CONTENTS

I. WATER COOLED REACTORS (040101)

A. EBWR

1. 100 Mw Modifications

The modifications to the EBWR plant to permit operation at high power levels, and thus establish performance characteristics of the system up to 100 Mw, are complete except for the primary reboilers. These units are being repaired in the vendor's shop and will be reinstalled only after the Laboratory is satisfied with the integrity of the units.

While awaiting approval of the 100 Mw Hazards Summary Report (ANL-5781, Addendum-Revised) the reactor will be put back in operation at a power level up to about 20 Mw using the core loading now in the reactor (20 spike fuel elements). The purpose is to check out the system and instrumentation, and to obtain preliminary operating experience in preparation for the high power experiments. This phase of the operation will be carried out with substantially the old EBWR system and hence does not depend upon the completion of repairs to the reboilers.

2. Development and Testing

a. 17-4 PH Investigation - The recent failure of a 17-4 PH tube used in the Dresden control rod drives has prompted an AEC investigation of 17-4 PH components on all reactors. Preliminary results of the investigation indicate that 17-4 PH material heat treated to 1100°F for one hour and air cooled will provide the necessary ductility for reactor components (previous heat treatment - 900°F for one hour and air cooled).

The increased ductility resulting from the higher heat treatment (1100°F) causes a decrease in hardness which may affect the life of wear surfaces of 17-4 PH components on the EBWR rack and pinion control rod drives.

The EBWR control rod drive test facility is being made ready to test several 17-4 PH racks, pinions, and seal shafts heat treated to 1100°F and air cooled. The tests will be started about the middle of March.

b. Soluble Poison - The various systems that will be used for soluble poison control of EBWR have now been completed. One primary component of the continuous soluble poison monitor alarm, a 31-point meter-relay, was defective and has been returned to the factory for adjustment. The continuous monitor can be operated for preliminary tests without this component.

c. Flux Monitoring - A method for monitoring axial neutron flux distributions in a given fuel assembly has received preliminary study with promising results. Very small spherical "activation foils" will be irradiated in a miniature "rabbit tube" made of $\frac{1}{8}$-in. OD tubing looped through the fuel assembly. Air has been used to transport the "foils" in the tubing in preliminary tests in the laboratory. Further development is being continued.

d. Void Simulators - Testing of these units indicated several welds which leaked. A modification of these units is required to permit full entry into the fuel element.

e. Instrumented Fuel Assemblies - An improperly manufactured thermocouple probe has delayed the final fabrication of the second instrumented fuel assembly. The hot junction end of this thermocouple probe had to be cut off to correct this condition. As a result, these thermocouples had to be recalibrated. Recalibration has now been accomplished and final fabrication of the fuel element is in progress.

3. EBWR Control Rod Follower Burnout Test

A loop experiment was set up to study the flow and heat transfer characteristics of the region of the fueled control rod followers in EBWR at 100 Mw. Heat is generated in the followers and the cross-sectional area for flow is less than that of a fuel element channel.

A small heated section and riser hydrodynamically representative of the control rod channels in EBWR [0.635 cm ($\frac{1}{4}$in.) ID heated section 91.4 cm (3 ft) long and a 0.635 cm riser 244 cm (8 ft) long] were installed in parallel with the existing test section and riser in the large scale loop. With the large section operating at the average heat flux in the hottest channel of EBWR at 100 Mw and the subcooling controlled to correspond to EBWR conditions at 100 Mw the parallel channel was heated with a power supply separate from the main loop.

A continuous trace of flow in the small section (as measured by a pressure transducer across a venturi in the inlet section) and inlet temperature was taken. Up to a flux of 41 w/cm^2 [130,000 Btu/(hr)(ft^2)], small oscillations in the flow ($\pm$6 cm/sec, $\pm$0.2 ft/sec) were noted. At 41 w/cm^2 [130,000 Btu/(hr)(ft^2)] the system became violently unstable with venturi pressure oscillations corresponding to flow variations from 0 to 112 cm/sec (3.7 ft/sec). The system remained unstable up to a flux of 85 w/cm^2 [270,000 Btu/(hr)(ft^2)] where a temperature excursion caused power to be tripped off.

Owing to differences between hydraulic and heat transfer equivalent diameter, the corresponding heat fluxes in EBWR will be a factor of two greater. That is, oscillations in the EBWR fuel follower channel would

probably occur at about 79 w/cm^2 [250,000 Btu/(hr)(ft^2)]. However the departure from nucleate boiling could still occur at a flux of 85 w/cm^2 [270,000 Btu/(hr)(ft^2)] as demonstrated in the test because of the inception of flow oscillation.

4. Analysis of Hazards Situations

As part of a general safety review of the reactor certain items of kinetics information were calculated for hypothetical incidents. Two cases of accidental rod withdrawal have been studied:

(a) The reactor vessel is closed and a rod is removed at the normal rate of 1.225 cm/sec.

(b) The reactor vessel is open and a rod is inadvertantly removed (e.g., by a crane) at a rate of 7.64 cm/sec.

Case (a) - 1.225 cm/sec - At this rate of withdrawal, the average rate of insertion of reactivity is about 0.04%/sec. At this rate it would take 19 sec to reach prompt criticality if there were no negative feedback. Analysis of the incident shows that there will be ample time for the coolant to warm up and that the resulting negative feedback due to the increased temperature will prevent an excursion.

Case (b) - This case yields a reactivity insertion rate of about 0.6%/sec while the control rod tip is moving through the central two-thirds of the core. This implies an insertion of 2.4% of reactivity in 4 sec; 1.2% to 1.6% may be compensated for by the metal negative temperature coefficient, the remainder will lead to prompt criticality and eventual burnout due to film boiling.

5. Core II

Revised drawings for the reference Core II fuel have been prepared for issuance with the request for bid.

The sample end fitting and shroud were completed and have been tested. Several changes on the end fitting and shroud are required to make the latching and unlatching easier. The unit was latched and unlatched a total of 20 times. This is greater than any requirement which can be visualized for a single fuel element.

The end fitting shroud unit was loaded to 81.8 kg (180 lb) in tension without loosening the end fitting pins from the shroud. The unit was checked in compression to 909 kg (2000 lb) without the shroud slipping over the seat on the end fitting. It appears that this type of unit will be adequate for the desired service.

B. BORAX V

1. Construction

All of the major piping has been installed and radiography is complete except for one portion of feedwater piping. Hydrostatic testing of most of the piping is also completed. The ASME Power Boiler 1050 psi hydrostatic test of the pressure vessel and associated piping to the first block valve was satisfactorily completed. Work has now been started on installing insulation on the piping in the subreactor room. Stress-relieving of the main steam lines was completed.

The painting color code was established and painting is in progress.

The electrical contractor installed conduit and cable trays in the access shaft and subreactor room. Work was begun on termination of cables coming from the control building. Final inspection and acceptance of the 12.5 kv power line from the main underbuilt section to the Borax substation was accomplished.

The estimated total completion for the project is 98%.

2. Procurement, Fabrication and Installation

Sixteen hundred boiling fuel rods were returned to Westinghouse Electric Corporation at their request for X-ray examination. Examination of selected rods revealed a defective pellet in one of the first group of rods assembled. No extension beyond the final delivery date of March 15 is anticipated for examination and replacement of any defective rods.

Work on the core components is proceeding in the Laboratory's shops. Owing to delays in receipt of some materials, the completion date for the boiling fuel assemblies has been extended from March 10 to March 30.

Orders were placed for the purchase of the differential thermopile, the reactor water ion-exchange tank, and polishing demineralizer tank.

Requests for bids were sent out for thermocouples for the boiling assembly thermocouple "rake."

The control and instrument cables from the reactor and turbine building were terminated in the control building.

Installation of the majority of the air-ejector-exhaust equipment was completed and connection of the air-ejector-exhaust instrumentation and control equipment was started. Overhaul of turbine auxiliary equipment continued.

3. Design

Revisions were made to core structure, hold-down boxes, chimneys and dummy fuel assembly drawings. Design of the pressure-vessel-head, nozzle-shield plugs was completed. Revisions to reactor control and instrumentation drawings were completed, and detailed design drawings for the superheater fuel elements were checked.

Work by the reactor physics group included re-evaluation of shutdown worth of the outside control rods in the all-boiling core under cold conditions. The worth of the peripheral assemblies in all three full size cores was also rechecked. Studies were made on the flux measurements to be made in BORAX V cores and a tentative design for a counting scanner was evolved. A choice of flux wires was made and consideration was given to methods of introducing and removing the wires from the reactor. Zr-U^{235} wires are favored as giving the most reliable picture of fission density in this reactor.

The Design and Hazards Summary Report has been completed and is being reviewed. Writing of the Operating Manual is in progress.

The preliminary design of terminal boxes and seals for bringing in-core instrumentation leads out of the reactor vessel was completed. Final design of in-vessel thermocouples, differential thermopile and their installation fixtures was completed. The design of the boiling assembly thermocouple "rake" was also completed. Drawings for the superheater assembly flow venturi are nearing completion.

4. Development and Testing

a. Boiling Fuel Elements - A total of 1600 fuel rods for the boiling core have been received. The first 100 fuel elements were evaluated using ultrasonic techniques. The standard used in the tube integrity inspection was a V-notch 3 mils deep and 0.5 in. long. This notch was located on the ID of the tube. No internal discontinuities were detected in the fuel. However, a number of fuel elements had surface defects such as scratches, flat marks, chips, pits, and pinch marks, and because of an assembly defect all 1600 rods were returned to the vendor for reinspection.

b. Superheat Fuel - All nondestructive tests and all destructive tests, except chemical analyses of core samples, as noted last month, have been completed. Over 1700 measurements were made of the face cladding and the core thickness; 36% of the face cladding measurements were below specifications, and 83% of the core thickness measurements were above specifications.

Measurements of end cladding, side cladding, stringer length, and maximum oxide particle size have been taken and are now being tabulated. Two end cladding samples from each plate have been subjected to the Strauss test. One sample which had been subjected to the test exhibited grain boundary cracking after being bent 180° around a $\frac{1}{8}$ in. diameter mandrel. Metallographic examination of a piece taken from the specimen which had exhibited extensive cracking showed a continuous network of carbides to be present at the grain boundaries. Specimens taken from other cladding samples exhibited grain boundary carbides to varying extents, but the carbide networks were not as heavy or continuous as in the former case. Samples of cladding from each of the four different plate types are being analyzed for carbon content. These analyses will aid in the interpretation of Strauss test results.

Destructive examination of plate samples substantiated the results of nondestructive testing for bond integrity. No nonbonded areas were found at clad-to-clad or core-to-clad interfaces.

On the basis of extensive testing and experience obtained at Oak Ridge National Laboratory with various brazing alloys for stainless steel, Coast Metals 60 alloy has been selected as the reference alloy for brazing BORAX-V plate assemblies. This material has suitable ductility characteristics and good corrosion resistance. An assembly of stainless steel plates is now being made using a carrier cement of polystyrene in benzene to apply the alloy along the sides of the wire spacers and at the plate to side plate junctions.

c. Control Rod Drive - The old drives from EBWR are being reworked for use on BORAX-V. The extension shafts are made of 17-4 PH stainless steel, a high hardness steel which has aroused concern because evidence of stress cracking has been noted in several reactor applications.

A second extension shaft, which was an original EBWR spare and had not been used in operation, is undergoing metallographic analysis. Both the first and second shafts showed evidence of cracks which penetrated the base metal. This initial investigation is still in progress.

Based on the early reports, twelve new extension shafts have been ordered for the drives. A record of their total history will be maintained from the time they leave the mill until they are finish machined. In order to get a wear test on the new shafts (which will be heat treated differently, and possibly not to as high a hardness), two of the original shafts have been stripped of chrome plate and are being rehardened to the new specifications. They will be rechromed and then wear tested.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. ZPR-III

Studies on Assembly 31 were completed with a set of measurements on the reactivity worths of aluminum, aluminum oxide and sodium at various radial positions along the axial midplane of the core. Measurements were made at core center, along radial arcs at an intermediate position, and at the edge of the core.

In these experiments all of the 45% density aluminum in the front 2 in. of each drawer was replaced first with 100% density aluminum and later with aluminum oxide.

In the sodium substitution experiments, stainless steel cans filled with sodium were loaded as in the aluminum and aluminum oxide experiments. However, the sodium-filled cans were substituted for all the aluminum pieces in a 2 in.-thick region surrounding the sodium substitution region.

The results are shown in Table I.

Table I. Radial Substitution Worths in Assembly 31

	Region I	Region II	Region III
Average radius (cm)	0	28.35	43.71
Aluminum Substitution			
Mass (kg)	1.909	1.697	1.272
Worth (Ih)	+1.67	+7.40	+9.37
Reactivity coefficient (Ih/kg)	+0.875	+4.36	+7.37
Al_2O_3 Substitution			
Mass (kg)	3.263	0.967	2.175
Worth (Ih)	+28.9	+7.95	+6.78
Reactivity coefficient (Ih/kg)	+8.86	+8.22	+3.12
Sodium Substitution			
Mass (kg)	1.183	1.051	0.788
Worth (Ih)	+11.6	+13.6	+9.5
Reactivity coefficient (Ih/kg)	+9.76	+12.9	+12.1

Recent studies (Assemblies 29-31) have shown that the criticality, and therefore critical mass, predictions are inadequate for large dilute power reactors. The preliminary conclusion is that the cross-section input data are inadequate. The cross-section for U^{235} could be shown to

account for about half of the discrepancy, but little quantitative information could be obtained from the basic aluminum and steel cross-section data, which appeared to account for much of the remaining discrepancy.

The problem can be most easily and simply studied by building simple assemblies containing only uranium-235 and a diluent. At various times some such assemblies have been constructed in ZPR-III and it was felt that Assembly 32 should be a uranium-235 and stainless steel assembly designed to shed some light on the adequacy of the steel cross-section values used in the multigroup calculations. Additionally, it was thought that it would be worthwhile to follow this assembly up with an assembly containing steel and sodium.

Assembly 32 was built with drawers containing 12 in. of fuel material. The fuel was two columns of uranium-235 and fourteen columns of stainless steel. The assembly has a critical mass of 227 kg U^{235} and has the following dimensions: 61.1 cm x 52.4 cm diameter. The calculated critical mass was 176 kg U^{235} for a sphere of radius 28.97 cm.

By building this assembly, a series of simple assemblies containing fourteen columns per drawer of the common fast reactor diluents has been obtained. Some of their properties are shown in Tables II and III. Each had two columns of enriched uranium. The inadequacies of the aluminum and steel data can be observed from Table II.

Table II: Comparison of Small Simple Assemblies (2 columns U^{235} and 14 columns diluent)

Assembly	Diluent	Dimensions length (cm) x diameter (cm)	Critical Mass (kg)	
			Experimental*	Calculated**
14	C	45.9 x 46.3	136	126
23	Al	51.0 x 101.0	258	~200
32	Steel	61.1 x 52.4	225	176
22	U^{238}	51.0 x 90.7	244	237

*No corrections have been made for either homogeneity or shape

**Using Okrent-Yiftah 16-group cross section set.

It is difficult to compare the reactivity worths given in Table III without some normalization and it is recommended that the reactivity cross-section for Pu^{239} at the center should be used for this purpose. It is known to vary least of all cross sections with change in spectrum and thus such normalization allows some insight into the problem. The fissile results appear to show a consistent trend, but this cannot be said for the non-fissile elements.

Table III: Comparison of Reactivity in Small Simple ZPR-III Assemblies

	Central Reactivity Worths (Ih/g mole)		
Assembly	23	32	22
Diluent	Al	Steel	U^{238}
Material			
U^{233}	65.9	95.91	112.4
U^{235}	36.2	50.04	60.7
U^{238}	0.87	-0.55	-2.78
Pu^{239}	69.2	96.87	115.1
C	0.79	2.08	-0.37
Al	0.069	0.85	-4.54
Stn Stl	0.016	0.088	-0.82
$B_4^{10}C$	-66.56	-113.15	-97.25
Zr	0.29	0.67	-1.20
$(\nu-1)\sigma_{f_9}-\sigma_{c_9}$	3.39	3.29	3.26 barns

The assemblies reported in Tables II and III all possess hard spectra and with the present trend towards large reactor systems, it may well be necessary to build assemblies which are both simple and dilute.

The measurements which have been completed on Assembly 32 so far are reactivity measurements for the common reactor materials at the center of the core and for a few selected materials towards the edge of the core. The latter have been taken in order to review and compare the transport properties of the materials with calculation and reactor experiments. The results are given in Table IV.

Table IV: Central Reactivity Worths (inhours/g mole) in Assembly 32

Material	Center	Edge	Material	Center	Edge
U^{233}	95.9	-	C	2.08	1.69
U^{235}	50.0	-	Al	0.85	1.03
U^{238}	-0.55	-	Fe	0.005	0.36
Pu^{239}	96.9	-	Ni	0.204	0.63
Cu	0.22	-	Cr	0.46	0.79
Zr	0.67	-	Stn Stl	0.09	0.46
Mo	-0.64	-	Na	0.33	1.69
V	1.48	-	Th^{232}	-3.51	-1.09
$B_4^{10}C$	-113.15	-	U^{238}	-0.55	0.56
Al_2O_3	6.38	-			

2. ZPR-VI and ZPR-IX

a. Construction - All of the concrete block walls in the building have been erected with approximately 75% of the surfaces requiring paint already primed. The following items have been completed and/or installed during this month: thermal insulation on service floor ceiling, the concrete topping and waterproofing over south loading dock, and the exhaust fans and filter racks in the penthouse

The construction of, and installation of equipment in Cell #6 is complete. The test cell door operators and butterfly valves have been installed in Cell #5 and checked. The pneumatic control system for heating and ventilating the cell is approximately 75% complete. The cell air compressor and electrical unit heater have been installed, and the electrical power distribution and security alarm system is approximately 80% complete. In Cell #4 the test cell doors are being realigned and their operators installed. Lights, butterfly valve operators and the cell air compressor have been installed.

The necessary preparations for making measurements required to monitor the pressure tests of the thick-walled concrete cells have been completed. All strain gauge equipment has been obtained and the instruments calibrated and errors minimized. The gauges selected are A-8 standard SR-4 type and are coupled with a three-lead self-compensating wiring circuit. Temperature effects on this system have been measured. Techniques have been devised for removing mill scale and rust from the reinforcing steel to permit attachment of gauges with Duco Cement which has been selected as the adhesive which gives minimum creep under extended load. The sample No. 8 reinforcing bars used in technical development have been made into simple cantilever calibration bars. Using these, a study was made of errors introduced by angular misalignment. Five-degree misalignment results in an error of less than 2% so that special locators are not needed.

b. Procurement - The design, procurement and testing of the many components required for the ZPR-VI and IX systems continues, with the usual succession of difficulties encountered and compromises or solutions of these problems achieved. The procurement of enriched uranium plates is a representative illustration of the type of difficulties encountered.

An initial shipment of 1,000 plates was received recently from the supplier. The inspection showed that the surfaces of these plates were abraded, probably by belt sanding. By such a procedure, the tolerances on thickness could be met, but by this operation the vendor had removed the hard black oxide coat that had resulted from the rolling operation. This left exposed the soft undersurface which will oxidize at a high rate. Unfortunately,

the contract for procurement of this material did not cover a condition of this sort. Consequently, it is planned to proceed promptly with the addition of a protective coating for those plates which meet the specifications.

The coating machine is ready for use. Because of the enriched nature of the material, it is necessary first to develop a procedure for the handling, transportation, coating and repackaging of the enriched plates. Coating will begin as soon as the procedure, now being formulated, has been reviewed and approved by the Laboratory's Criticality Hazards Control Committee.

Production of the stainless steel matrix tubes for the ZPR-VI system at a rate of approximately 80 units per day, with shipments scheduled to begin in February, was mentioned in last month's Progress Report. As yet none of these has arrived. The supplier for the aluminum matrix tubes needed for ZPR-IX has indicated that shipment of those tubes should begin in mid-March.

The machining and assembly work on the bed and table assemblies for both ZPR-VI and IX is essentially complete. Inspection at the vendor's plant is scheduled for March.

c. Control Rod Drives - Two types of control rod drives are to be used for these reactors: an expulsion type, whereby a fuel drawer will be ejected from the core; and an insertion type, which will insert a boron-filled shim rod into the core.

The drive formerly known as the Fuel Drawer Expulsion Drive will henceforth be called the Dual Purpose Control Rod Drive.

The drive has completed its 1000-cycle life test with ease. The fuel drawer used was loaded to a weight of 30 kg and its bottom and sides, as well as the tube in which the drawer must slide, was coated with a baked-on film of molybdenum-disulphide and graphite. This coating has virtually eliminated all galling and scratching previously experienced between the two Type 304 stainless steel surfaces. After more than 1000 scrams, the scram time for the first 15 cm travel remains at 105 ms and for the full 61-cm stroke at 275 ms. The full stroke time varies from test to test depending primarily on the condition of the fuel drawer. The 300-ms time reported last month was for a moderately scored chrome-plated drawer, which later proved to be unsatisfactory. A time of 275 ms is perhaps the best that can be expected, and a time of 300 ms the absolute worst.

The blade insertion drive is undergoing a 1000-cycle life test. Scram time for 75 cm travel is holding steady at 110 ms after about two dozen scrams. The blade is filled with a boron powder, and at this early date no effect of tamping can be detected. General operation of the drive is considered satisfactory.

d. Experimental Program - Preoccupation with the problems of procurement of the critical assembly building and the ZPR-VI and ZPR-IX systems, and the preparation of the Hazards Summary Report for the ZPR-VI program, have made it impossible to devote a major effort to preparations for the experimental program. Consequently, the only item on which a significant amount of work is being done is the investigation of alternate techniques for the determination of fast neutron spectra in the ZPR-VI system. Previous Progress Reports have recorded the efforts using a coincidence recoil technique as well as several alternate schemes under consideration. One of these is the use of a methane-filled proportional counter in a manner analogous to the nuclear emulsion technique, which has been used successfully for many years on Argonne's fast reactor systems. Because of its simplicity, this approach is being explored experimentally.

As a first step, a proportional counter has been built and an investigation of its characteristics for the detection of fast neutrons is in progress. The counter is filled with two atmospheres of methane and small additional amounts of nitrogen in order that the 600 kev neutrons due to the $N^{14}(n,p)$ reaction may be used to establish an energy scale and determine the resolution. The response of the counter to a high gamma ray field and to thermal neutrons has been measured. Resolution with two atmospheres of methane and ten centimeters of nitrogen is approximately 35%. A reduction of the total pressure to one atmosphere improved the resolution significantly, to about 12%. Thus, it appears that the resolution is determined principally by impurities in the filling gas, rather than wall effects. Additional testing will follow.

Meanwhile, a calibration run with the coincidence recoil spectrometer was performed at the Van de Graaff generator with a filling pressure of 5 cm Hg. This corresponds to a useful neutron energy range of approximately 200 to 650 kev. A resolution of approximately 13% was measured. This is about 50% higher than the resolution expected for this spectrometer, and compares with that obtained in the preliminary tests with the methane-filled proportional counter. The recoil spectrometer and its associated circuitry is performing in a satisfactory manner, and it is now considered essentially ready for spectral measurements on the Argonne Fast Source Reactor (ZPR-IV') or the ZPR-III system. However, it appears advantageous to delay any such tests until the promise shown by the methane-filled proportional counter is confirmed by additional testing. The proportional counter procedure for fast neutron spectrum measurements is preferable because it does not introduce the additional problem of extracting a beam of neutrons having a spectral distribution representative of the assembly itself.

3. JUGGERNAUT

The JUGGERNAUT is a water-cooled and graphite-moderated reactor of the ARGONAUT type. It is to operate at power levels up to 250 kw and provide a versatile facility for nuclear research and component development.

Phase 3 of the construction is now complete (see Progress Report, September, 1960, ANL-6234, p. 12) except for wiring of the nuclear chambers and placing of the concrete blocks at the Universal Test Cave. Completion of Phase 3 terminates the construction of the reactor, and testing of the reactor components has begun.

During the month the control rods were reworked because the sliding action was not smooth enough. Three of the reworked rods have been installed in the reactor; the remaining four rods will be installed in early March One of the rods will be cycled in the reactor for about 1000 to 2000 falls. The rest of the rods will be cycled a few hundred times.

The rotating top shield was filled with high density concrete and placed in the reactor. The shield weighs 3.5 tons and rotates quite well. It also may be rotated with the coffin; the combined weight is 7 tons.

Dummy fuel assemblies were inserted in the reactor. It was found that an assembly could easily be extracted from the reactor into the coffin and replaced in the reactor at any position desired. It is also possible to place an assembly between two neighboring assemblies. All of these operations were done blind; that is, the operations were done through the coffin.

The reactor seal, a 26-in. tube, inflated to about 0.33 atm, makes a seal sufficient to hold 5 cm of helium pressure.

Water dump tests were run under reactor conditions without the fuel. It takes 1.5 sec to drop the water to the top of the fuel. This is equivalent to 1% $\Delta k/k$. It takes 3 sec to drop the water completely.

4. Fast Reactor Fuels - General

a. Fast Reactor Fuel Jacket Development - The desired higher operating temperatures for future fast reactors require a fuel jacket material which will be compatible with both sodium coolant and fuel, have good fabricability and weldability, and high strength and low creep at elevated temperatures. Several niobium and vanadium base alloys possess some of these properties at temperatures in excess of 900°C and may be desirable for this application. Vanadium base alloys have been developed by Armour Research Foundation for fabrication development work at the Laboratory,

and niobium base alloys have been developed for the Laboratory by Battelle Memorial Institute. Commercial suppliers have been contacted to supply tubing of the above high strength alloys.

An additional Ta-0.1 w/o W alloy thimble was cold worked as reported in the Progress Report for December, 1960, (ANL-6295). This latest effort resulted in four lengths of tubing 0.176 in. OD x 0.156 in. ID. Finished tube lengths varied from 2.75 in. to 16 in. The tubes were leak checked with a helium mass spectrometer and no defects were found.

B. EBR-I

1. Operation

During the past month all reactor operating time was used for training personnel.

2. Mark-IV Core

a. Fabrication - Injection casting of Pu-1 w/o Al alloy fuel slugs was begun during the week of February 13. Seven melts have been produced to date using the rather severely oxidized plutonium received on the first two shipments. One of these melts was not cast because of excessive gas evolution from the melt, causing spitting of the metal and a frothy slag. One hundred seventeen slugs were cut from the first four melts and have been stored in preparation for final machining and loading.

Some difficulty has been encountered with reaction and adherence of the glass molds to the castings. An yttrium-oxide coating has been used in Building 350 in place of the less reactive ThO_2 coating to prevent alpha contamination of the area. Tests are being run to determine if a ThO_2 coating is necessary.

A total of 1740 feet of commercial Zircaloy-2 jacket tubing has been received to date. All stock required for rod tips, connectors, filler plugs, and NaK level spacers has been fabricated, nondestructively inspected, and delivered for final machining at the Rock Island Arsenal. An order was placed for commercial fabrication of the Type 304 stainless steel extension rod components for fuel and blanket rods and delivery was promised for early April. An order was placed with a commercial spring manufacturer for fabrication of the Inconel-X spring component. Delivery was promised for approximately March 20. The production machining of depleted uranium components for all fuel and blanket rods has been initiated.

Defects present in previously fabricated zirconium alloy rib wire were traced to inclusions or stringers in the original Zircaloy-3 ingot stock. Ultrasonic inspection of rolled bar stock has eliminated this

defective material prior to wire finishing operations. Commercially fabricated 0.062 in. diameter Zircaloy-2 wire has been ordered as a backup effort in the event that sufficient Zircaloy-3 wire is not available when needed.

Three Zircaloy-2 tubes, 0.400 in. diameter x 0.050 in. wall, were extruded for use in the development of 0.080 in. diameter instrument tubing. No difficulty was encountered in extruding this tubing with the extrusion direction parallel to the longitudinal axis of the original 1.625 in. diameter forged bar used for billet stock. Unsuccessful attempts were made to alter the texture in extruded tubes by machining extrusion billets from rolled slab such that the extrusion direction would be normal to the slip planes developed in the rolling plane of the slab. On every one of these extrusions the forces exerted on the mandrel caused it to break.

The status of all components other than plutonium fuel is as follows:

Components	Status
Rod Tip	All material in process of final machining.
Jacket Tube	79% of order received and in process of nondestructive testing.
NaK Level Spacer	All material in process of final machining.
Fuel Connector	All material in process of final machining.
Filler Plug	All material in process of final machining.
Rib Wire for Jacket Tube	In process of fabrication and nondestructive testing.
Extension Rod	On order from outside vendor.
Restrainer Spring	On order from outside vendor.
Blanket Uranium	Ready for machining pending availability of drawings approved for construction.

Rod tip girth welding, rib wire welding and final machining have been completed on 18 tip rod and jacket sample assemblies.

Welding procedures have been established to weld the tip to the Zr-2 jacket tube. The weld is being made by the TIG method with an automated glass lathe using argon as a shielding gas.

A special welding machine designed and built to weld Zircaloy wires on the EBR-I Mark III fuel rod has been modified to accommodate the EBR-I Mark IV fuel tube. A Raytheon capacitor discharge power supply of 400 mfd capacity is used to resistance weld the three Zircaloy spacer wires simultaneously to the Zircaloy fuel tube.

b. Development - The loading of 98.75% plutonium fuel slugs into a close fitting Zircaloy tube, while maintaining an external contamination of the tube of not greater than one count per square centimeter per minute, has been a serious problem. It now appears, however, that a plastic fitting has been developed which will permit loading without producing external contamination.

The device consists of heavy vinyl tubulation, several of which are sealed to a 0.030 in. vinyl bag. The fuel tube is inserted in the vinyl tube and clamped by means of a hose clamp. A 0.003 in. polyethylene funnel is placed in the funnel section of the tube and extends into the tube opening to protect the weld lip as the fuel slugs are loaded.

Bonding equipment consisting of a heating furnace with a fuel element impacting device and a pressure type leak detector has been set up. As yet, only five sets of Zircaloy hardware have been received to test the loading, welding and bonding operations.

Testing work to date on the bonding problem has been done with an eddy current bridge circuit using either an encircling coil or side scan probes. A three-channel (120° coverage per channel) side scan probe instrument is under construction for further tests. The electronics portion of the three-channel system has been completed along with a permanent type of testing fixture. A variable speed motor has been incorporated in such a way as to provide vertical handling of the elements and full scale chart recordings. Voids have been created by an inverted shaking of the available element and studied as the NaK recombines. NaK held between an off-center fuel slug and the can has been observed, but the response to very small voids is not yet known. Artificially created and maintained voids will be produced in order to check the sensitivity.

The evaluation of certain engineering and physical properties of the Pu-1 w/o Al alloy composition is continuing. The scope of this program is as follows:

1. Thermal conductivity to 530°C.

2. Hot deformation or slumping tendency under several static loads of the order anticipated in reactor operation in the temperature range 400-600°C.

3. Compatibility at 425° and 500°C with Zircaloy-2, the proposed jacket material.

4. Thermal expansion between 200-600°C.

5. The linear coefficient of expansion as a function of aluminum content in the range 0.36 to 1.23 w/o to determine the minimum aluminum content yielding a high positive and reproducible expansion coefficient.

6. Hardness, density, and metallography to characterize more thoroughly this range of alloys.

Some thermal conductivity, hardness, density, and slumping results have been reported in previous Progress Reports.

Expansion curves have been obtained for the reference alloy composition and two other compositions in the as-cast condition. Tentative linear expansion coefficients that have been obtained are as follows:

Table V. Linear Expansion Coefficients of Some Plutonium-Aluminum Alloys

Composition		Temperature Range	Linear Expansion Coefficient
w/o Al	a/o Al	(°C)	per °C
0.98	8.05	20-425	10.6×10^{-6}
1.22	9.80	20-425	11.4×10^{-6}
1.23	10.00	20-425	12.5×10^{-6}

Two diffusion couples consisting of the Pu-1 w/o Al and Zircaloy-2, the proposed jacket material, have been heated in vacuum at 425°C and 500°C for a period of one week. Neither couple showed evidence of a reaction layer between the contacting surfaces of these two materials. In the couple heated at 425°C, both materials were plastically deformed and all surfaces remained clean and bright. It is suspected that the materials in the couple heated at 500°C were not in intimate contact since no plastic deformation was in evidence. All surfaces of this latter couple were oxidized.

C. EBR-II

1. Construction

The following tabulation presents the status of construction contracts as of February 14, 1961:

Building	% Completion	Estimated Completion-Date
Power Plant (Package 2)	100	-
Reactor Plant (Package 2)	100	-
Sodium Boiler Plant (Package 3)	86	*
Fuel Cycle Facility (Package 3)	77	*

*Construction schedule not available - completion cannot be estimated reliably.

The Power Plant and the Reactor Plant had been previously accepted from the Construction Contractor on an Interim Facility Transfer. Progress has been made by the Contractor on completing "punch list" work items, but some minor items remain to be finished.

Installation of equipment in the Fuel Cycle Facility by the Contractor is proceeding without hindrance as the building has been provided with heat and power. Major equipment being installed includes the high bay bridge cranes and associated equipment, the argon cell vacuum pumps, water pit pumps, the AEC filter banks, the fume exhaust plenums, the exhaust duct and filters for the sodium clean-up system, the air cell manipulator bridges, the degreasing unit, the air supply fans and equipment, and the recirculating cooling system.

Some progress is also being made on construction, including the first floor lighting, the bus duct and feed rail brackets, tile work, patching and painting interior walls and structural steel.

Installation of the high frequency motor generators which will supply power to the process furnaces in the Fuel Cycle Facility may be delayed because of the construction contractor's improper handling of this equipment. The motor generators and control boxes were shipped directly from the manufacturer's plant to the construction site and transferred to the contractor (Package 3). Although much rough construction work remained to be done in the Facility, the units were moved to the basement and uncrated. Subsequently, trenching operations resulted in at least some of these units being partially covered with gravel and crushed lava. When

the manufacturer was requested to evaluate the consequences of this condition, he advised that the units should be dismantled, thoroughly cleaned and all bearings replaced, and that this work should be done either at manufacturer's plant or under the direction of manufacturer's field engineer. This problem is not yet resolved. Figure 1 indicates the condition which caused manufacturer to withdraw his warrantee.

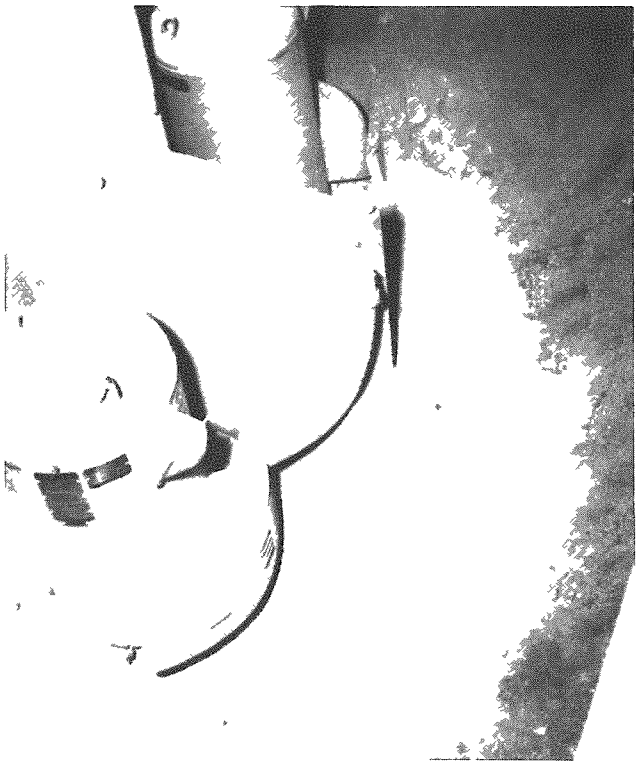

Figure 1

Basement of the Fuel Cycle Facility, Showing Dirt Piled Against the High Speed "Tocco" Units

The piping subcontractor has completed most of the major piping installation in the Sodium Boiler Plant and has removed his forces from the job site. The contract completion date is now delayed about 8 months and completion of the secondary sodium system without further intolerable delays is now dependent upon (a) receipt, at an early date, of the flexible seals which effect a containment seal between the two 12-inch secondary sodium lines and the Reactor Containment Vessel, or (b) performance of work out of sequence with that contractually specified. The flexible seals are a contractor-furnished item, and no inclination to perform work out of sequence has been shown. The lack of this equipment is now delaying completion of the secondary sodium piping system since pneumatic and helium mass spectrometer leak testing of the complete system cannot be performed until the seals are installed since the seals are an integral part of the system. Until these tests are satisfactorily completed, the system cannot be clad for induction heating, insulated, or painted.

The contractor has had almost twenty months to provide these flexible seals; yet the Laboratory still does not have a complete set of

shop drawings for this equipment. These seals perform a very critical function since they are a part of the Reactor Building Containment System. As a result of the contractor's lack of action and the urgency of removing this contractor from the Reactor Building so as to avoid interference with the EBR-II Experimental Program, the Laboratory has proposed to delete the flexible seal assemblies from the Construction Contract (by change order) and furnish them for installation under Package 4. The change order would terminate the contractor's work short of the Reactor Building and thus remove any effect of Package 3 upon the EBR-II Experimental Program schedule. It is hoped that the indicated delay can be minimized in this manner.

2. Installation of Equipment - Package 4

Installation work within the Reactor Plant falls into two general categories: (a) mechanical, comprising approximately 70% of the total; and (b) electrical, approximately 30%. For convenience in scheduling and supervising the work in the field, the mechanical work is divided into 43 major jobs, or packets, and the electrical into 5 packets. Work on 39 of the mechanical packets and all 5 of the electrical packets has been started. To date, approximately 70% of the total mechanical work has been completed.

During January, both the large and small rotating plugs had been returned to their normal operating locations within the primary tank support structure central ring. This represented the final installation of the large plug. The small plug, however, was removed once more (during the present period) for addition of the reactor vessel cover lifting columns, after which final installation was effected. Figure 2 shows the plug being lowered into final position. Also shown are the large ball bearings which support the weight of the plug, the index ring (used to indicate the angular position of the plug), and the bull gear for the plug which is driven by a drive unit mounted on the large plug. The wooden blocks taped to the top of the plug temporarily cover the seal trough heater holes. Figure 3 is a view of the top of the small rotating plug after final installation. The cluster of tubes in the center represent the upper ends of the control rod drive guide tubes; all control rod drives are now completely installed, as may be seen in Figure 4. The two large shafts clamped in place on the plug are the lifting columns of the reactor vessel cover lifting mechanism; this mechanism is also now completely installed. Figure 5 shows the completed installation of the control rod drive lifting platform on top of the small rotating plug. The central column appearing on the control rod drive lifting platform serves as the supporting member for the upper parts of the rod drive tube assemblies. Figure 6 is another view of the top of the small plug showing additional detail. In the background may be seen Instrument Control Center No. 1.

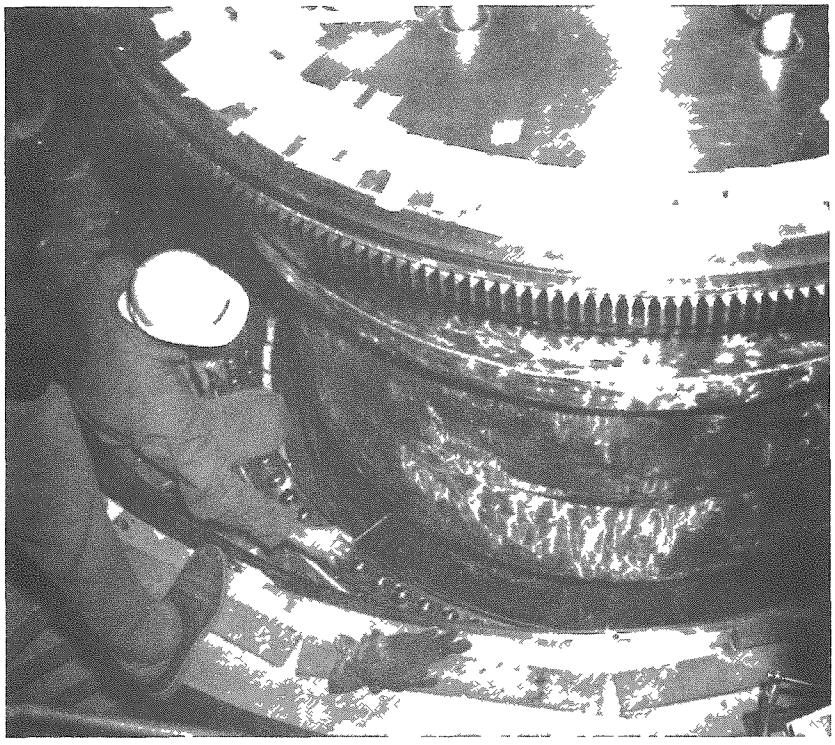

Figure 2

Small Rotating Plug Being Lowered
into Final Position

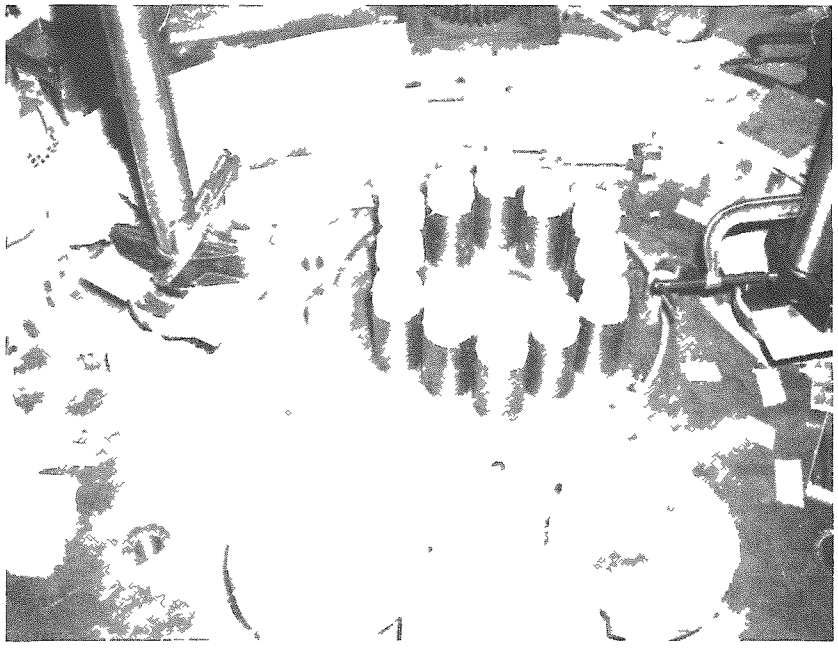

Figure 3

View of Top of Small Rotating Plug After
Final Installation

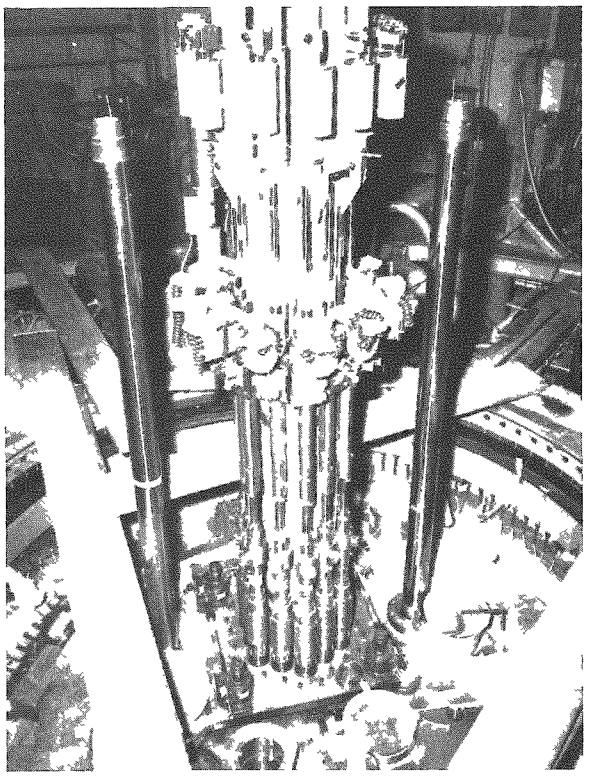

Figure 4

Control Rod Drives Installed

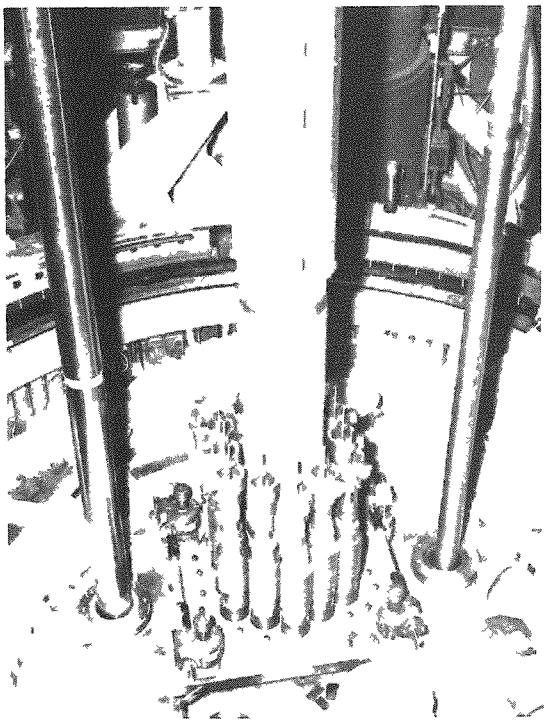

Figure 5

View of Control Rod Drive Lifting Platform on Top of Small Rotating Plug

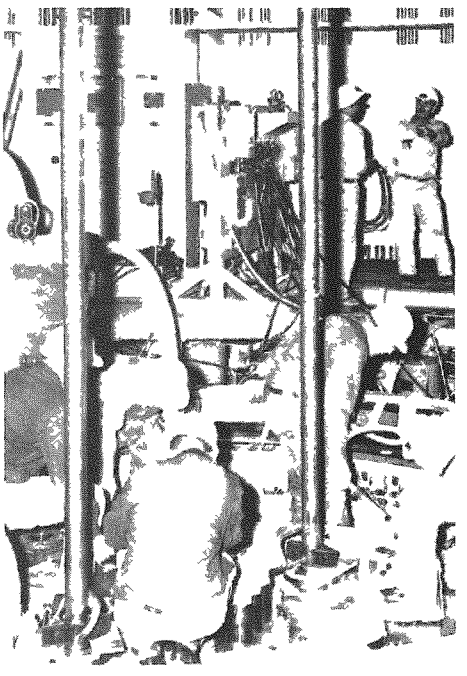

Figure 6

Another View of Top of Small Rotating Plug

The radial neutron shield surrounding the reactor vessel consists of five rows of stainless steel canned graphite blocks stacked three tiers high. Figure 7 shows part of the lower tier. Also shown (at upper left) is the lower end of one of the four "J" instrument thimbles strapped to the wall of the reactor vessel. A pair of reactor coolant inlet pipes passing through specially shaped shielding cans may be seen. The expanded metal work seen surrounding the shield is used to enable natural convection flow of bulk sodium for cooling of the shield cans. Installation of the entire radial neutron shield, all (eight) "J" and "O" instrument thimbles, and all reactor coolant piping is now complete.

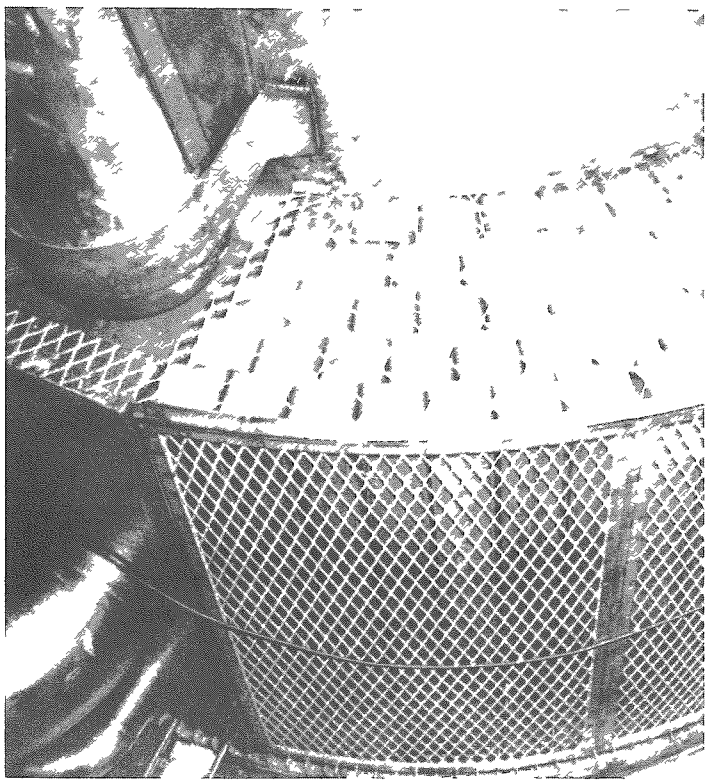

Figure 7

Part of Lower Tier of Radial Neutron Shielding

The major electrical installation effort has been in the Reactor Plant. A total of 25 electricians, 3 foremen, and 1 superintendent are installing equipment, cables, and wiring. Work is being concentrated on those items necessary to be completed prior to the Dry Critical Experimental Program. Electrical work is currently estimated at 62% complete and it appears that the required electrical work in the Reactor Building will be completed in early April.

3. Engineering

A number of special devices necessary for dry critical and post-dry critical investigations (both before and after sodium is in the system) are now in early phases of procurement or design. These devices include the following:

(a) One instrumented subassembly with thermocouples attached to the fuel element tubes to monitor temperature during the dry critical experiments.

(b) Two periscopes for remote observation of fuel handling within the primary tank during dry critical experiments and/or during sodium filling.

(c) A number of mirrors to be positioned temporarily within the primary tank for use in conjunction with the periscopes.

(d) Windows to be employed on certain primary tank nozzles to facilitate observations in the primary tank.

(e) Lighting facilities required for the above

(f) A temporary air circulating system to be used in the primary tank to assure reasonably uniform temperature distribution during isothermal temperature coefficient measurements and other dry critical and post-dry critical experiments.

(g) A closed circuit television system for observation in the primary tank prior to sodium filling is under investigation.

4. Procurement

The tube bundle of the sodium-to-sodium heat exchanger is complete, including the upper head and the lower inner head (see Figure 8). A preliminary helium mass spectrometer leak test of the completed bundle was successfully performed. Assembly of the shield plug section has started. The first step of this procedure consists of attaching the lower section of the shield plug (below the stepped section) to the tube bundle and installing the steel, graphite, and boron-steel shielding material. Owing to the curvature of the two 12-in. secondary sodium lines, installation of the graphite requires considerable hand fitting.

Approximately 90% of the Datex Corporation drawings for the Fuel-Handling Center have been approved, or approved with comments. Datex has been instructed to proceed with construction on the basis of these drawings without waiting for approval of revised drawings. Datex has sufficient information to proceed with the construction of all components and assemblies. The Laboratory is expediting all phases of this work (including Datex subcontractors) to insure or improve the April delivery of the Fuel-Handling Center.

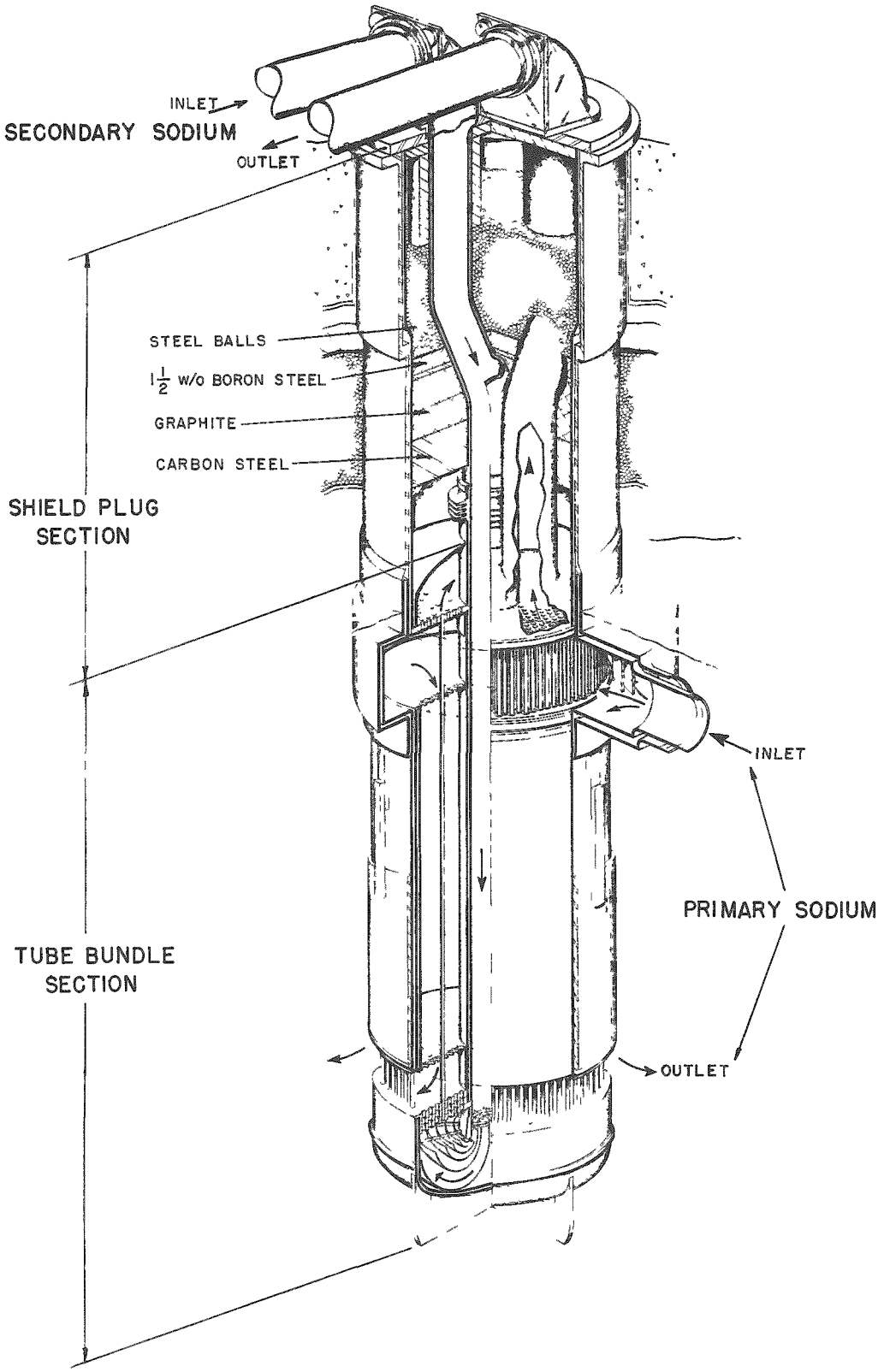

Figure 8

EBR-II
Sodium to Sodium Heat Exchanger

5. Component Development - Reactor

a. Storage Rack Mechanism - The storage basket has been received in Idaho and present plans are to ship the remaining pieces in the first week of March, 1961, which will complete all Package 4 shipments from the Lemont site to the field. All of the wiring on the unit is complete. A temporary power and control console has been fabricated and is being connected to the electrical equipment on the unit.

b. Leak Testing of Bolted Flanges for Primary Tank Nozzles - Several of the plugs installed in the primary tank nozzles have required shimming during installation to obtain the desired plumbness or angle as designed. This condition has been attributed to both deflections in the cover and out-of-squareness of the flanges with the nozzle vertical centerlines. The shimming thicknesses have been of the order of $\frac{1}{8}$-in. To determine if the copper "O" ring gaskets will provide effective sealing under these conditions a simple test fixture was constructed.

The test fixture consists of two blind flanges, of about 15-inch diameter, bolted together to compress a $\frac{1}{8}$-inch diameter copper wire "O" ring, approximately 10.5 in. in diameter. One of the flanges has a flat face while the other flange simulates the recessed bore for the gasket seats in the primary tank nozzle flanges. All finished surfaces are ground to a 30 micro-inch finish, which is the specified value on all primary tank nozzles. Three spare copper gaskets have been obtained for these tests.

Preliminary test results have been promising. The bolts on the assembly were tightened to effect a near uniform pressure on the copper gasket and the gap between flanges pressurized with 5 psig of helium and probed for leaks The gap was then evacuated and a helium leak rate test performed with an external helium pressure. Neither test indicated a leak. A second gasket was then placed in the groove and a shim of 0.040 in. was inserted between flanges at one point. The bolts were tightened until the flanges touched at the point diametrically opposite the shim. Again both helium tests revealed no leaks. It is planned to conduct a third test with a $\frac{1}{16}$-in. thick shim, and then to "soak" the assembly in an oven at about 200°F after which a final test will be made to assure that no leakage occurs. (The temperatures expected at these gaskets during operation are about 120°F-130°F.)

6. Component Development - Instrumentation

a. Pulse Counting Equipment - A new type of low dead-time neutron counting system has been developed at the Laboratory. This system is capable of measuring accurately the neutron flux at source level in a gamma field of approximately 10^4 r/hr The design is unique in that the preamp is physically attached to the counter shown in Figure 9 and required a special

circuit design to assure satisfactory operation in a high gamma field. The fission counter is a Westinghouse Type WX4245 with a 1 $\mu g/cm^2$ thick neutron sensitive coating of U_3O_8 enriched to 90% U^{235}.

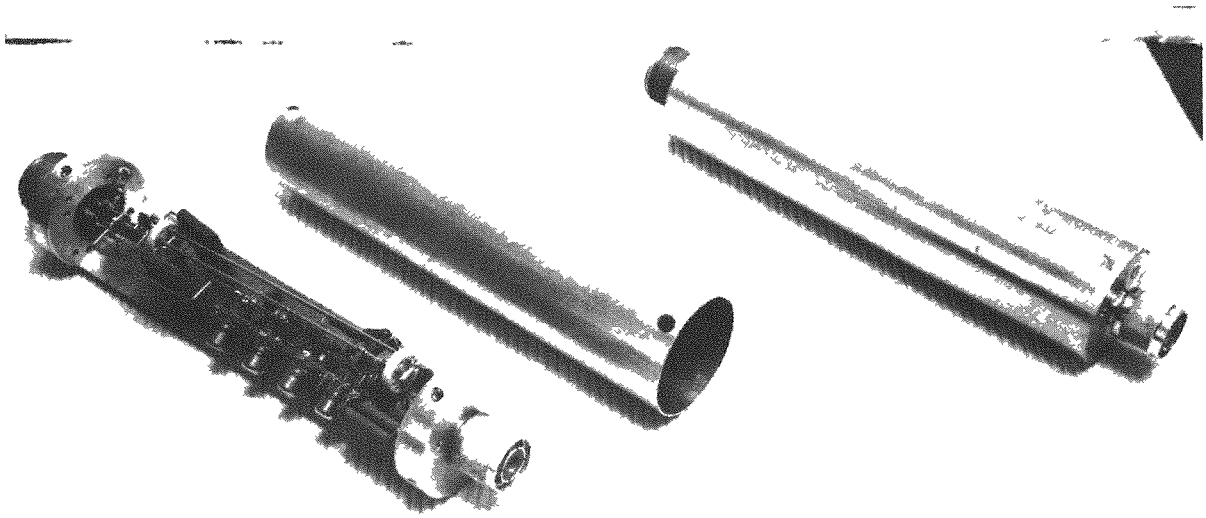

Figure 9

Neutron Counter with Preamp Attached

The design details of the preamplifier and the linear pulse amplifier are described in ANL-6292.* The response of this equipment will be calibrated with the Argonne Fast Source Reactor the flux spectrum of which is reasonably close to that anticipated in EBR-II.

b. Control Rod Reactivity Generator - A sinusoidal reactivity generator and wave analyzer are being designed to measure the transfer function of EBR-II. A new design is being investigated. The new design will possibly be linear instead of rotating and have a 2 cycle/sec speed with about a 4 in. throw. This drive would also be enclosed in a thimble with the rod not in contact with sodium. Gas cooling has been ruled out.

Preliminary electrical circuit layout of the reactivity generator instrumentation has been completed. Assembly drawings and details of external drive are approximately 85% completed.

c. Startup Instrumentation - Special instrumentation is being provided for EBR-II startup. Two movable thimbles will be installed in control rod locations. High temperature fission chambers capable of operating at 455°C will be used. Two separate channels of temporary

*R. J. Epstein and D. C. Thompson, "A Low Dead Time Neutron Counting System," January, 1961.

electronic equipment, high voltage supplies, preamplifiers, log count rate amplifiers, period amplifiers, scalers, and audio count rate circuits will be installed.

Modifications are being made to the system to incorporate a low level trip on each log count rate meter. Log count rate meter scales are being changed to read in counts/sec instead of counts/min The high temperature fission counters are not yet available for a complete checkout of the system.

d. Local Moderating Material for Neutron Detectors - A rather extensive literature search was concluded on the subject of zirconium hydride (ZrH_x) and its properties. The exact number of hydrogens associated with each zirconium atom is not known. It was proposed to use ZrH_x as a moderating medium in the vicinity of the low flux instrument-sensing chambers to increase the response during startup. Originally this called for replacing graphite in certain of the radial neutron shield cans with ZrH_x. Because these cans are submerged in sodium at 700°F under operating conditions, and because of the ever present possibility of leaks in the cans, information on the stability of ZrH_x at such temperatures and its compatibility with liquid sodium was required before proceeding.

It was later decided to abandon this scheme in favor of leaving certain of the cans inside the reactor vessel empty and placing ZrH_x inside the instrument thimbles and immediately surrounding the sensing chambers. The ZrH_x was to be contained in soft aluminum (2S) cans of special configuration. In this way, the ZrH_x and liquid sodium would never come in contact, and the search for information regarding compatibility with sodium was abandoned.

It was found that ZrH_x and aluminum were compatible at the temperatures involved (150°F at normal operating conditions and 700°F for periods of air coolant failure). For the hydride power available (Zircaloy-2 hydrided to 1.80 wt-%, or $ZrH_{1.69}$) the vapor pressures (H_2) at these temperatures were of such a small order as to result in negligible losses of the H_2 by permeation and diffusion. It was also determined that damage to the aluminum cans by dissolved hydrogen would be negligible or extremely slight.

e. Detector for Fuel Element Failure - In ANL-6234 (Monthly Progress Report, September, 1960), the detection of fuel element failure through the observation of delayed neutron emission by fission fragments in the EBR-II coolant was discussed. Sensitivity tests were performed at that time at the CP-5 reactor, and it was demonstrated in these measurements that the gamma radiation associated with the sodium coolant stream in EBR-II would produce an excessive background of photoneutrons through reaction with the deuterium present in hydrogenous moderators Consequently, a shift to graphite moderation was proposed and preparations were

made to optimize the design of a graphite moderating medium. It was anticipated that the appreciably larger slowing-down area in graphite would reduce the sensitivity of the counter and associated moderator to a fraction of that observed in the CP-5 measurements with paraffin.

A flexible moderating arrangement of graphite surrounding a centrally located tube simulating the coolant tube in the cold trap room of the EBR-II reactor has been assembled, and a sodium-heavy water photoneutron source has been prepared. It is now inserted in the graphite moderating system and various performance tests are underway. These measurements include a determination of the intensity and distribution of thermal flux in the graphite column as a function of the size of the assembly, the neutron sensitivity and γ-ray discrimination of two types of fission counters, and tests of the degree of suppression of gamma-ray pile-up by use of fast amplifying circuits.

The specifications for a system to be assembled at the EBR-II site for further testing will result from these measurements. Tests in situ are necessary because of the uncertainty in regard to the neutron background which will exist in the cold trap cell due to (γ-n) reactions of the sodium gammas with deuterium in the concrete walls of the cell. This background level will determine the minimum fuel element failure which can be detected by the delayed neutron technique. Another aspect of the development of a detector of fuel element failure is the design of circuitry which will reduce spurious warning signals to a minimum. In this connection, the statistics of detection of a small signal has been investigated and, based on these considerations, a prototype digitalized count-rate meter of superior stability and performance has been proposed.

7. Component Development - Steam Generators

Previous Progress Reports (December, 1960, ANL-6295 and January, 1961, ANL-6307) have described the difficulties encountered in fabricating the superheater units; namely, the tube-to-sodium tubesheet weld.

The details of the weld in question are shown in Figure 10 This figure shows the sodium inlet end details of an evaporator; however, the superheater is geometrically similar with the exception that baffle tubes are not used and dimensions are generally smaller. For example, the superheater utilizes a 0.596 in. OD tube with a 0.052 in. outer wall and a 1.109 in. pitch, whereas the evaporator utilizes a 1.438 in. OD tube with a 0.094 in. wall and a 1.938 in. pitch. The present welding problem was not evident during fabrication of the evaporators. The smaller tube diameter and thinner wall of the superheater tube seem to fall beyond the limit of the present welding techniques to produce consistently reliable welds.

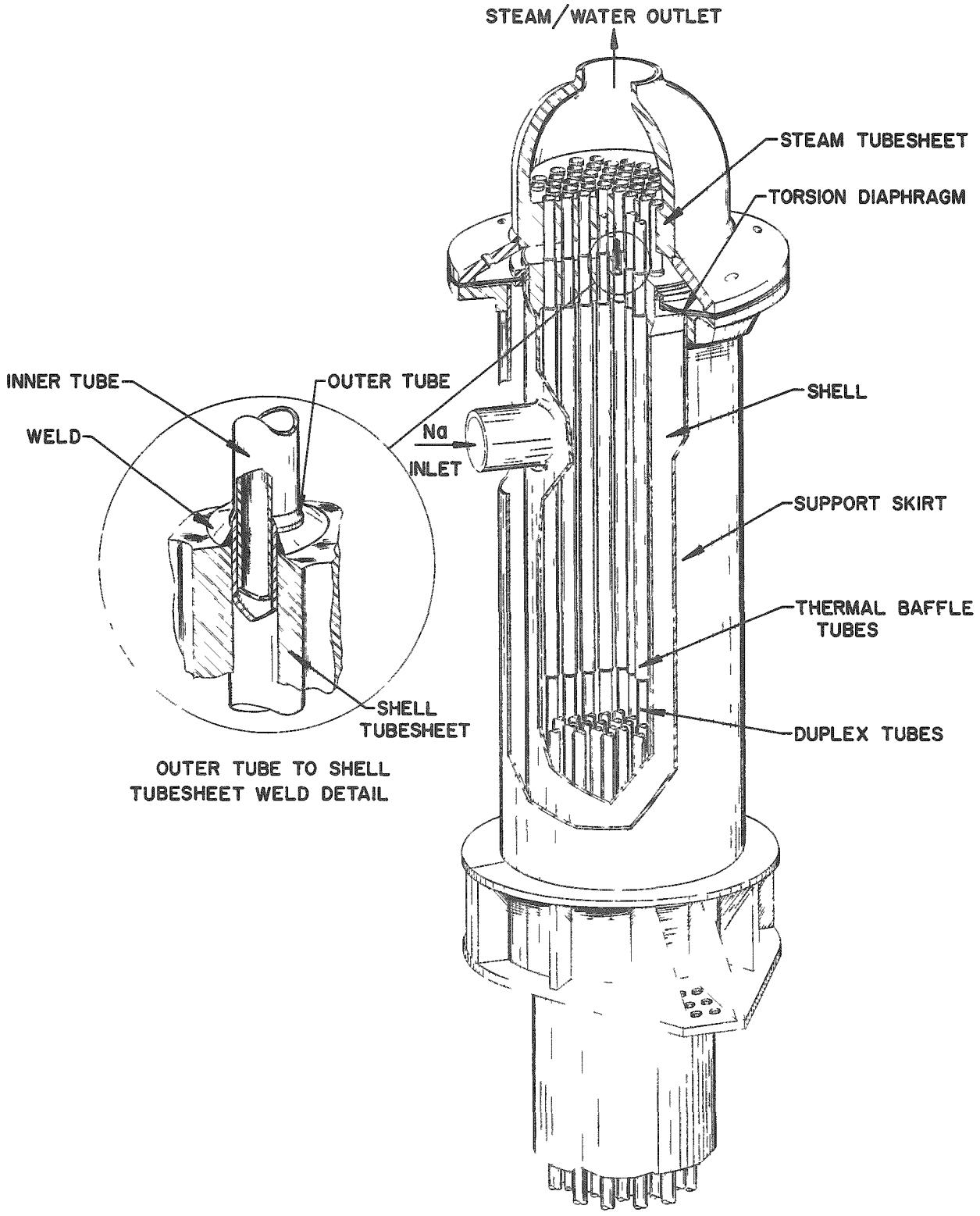

Figure 10

EBR-II Evaporator
Sodium Inlet End Details

The effort to develop a reliable and consistent welding procedure for the superheater tubes is continuing; however, the solution to the problem is as elusive as ever. It must be emphasized that the difficulty is not one of making a sound and reliable weld, but rather of doing so consistently. Basically, two fundamental welding procedures have been the most thoroughly investigated - the spray transfer and the dip transfer modes of metal deposition. The principal difference in the results of these methods, at this time, is that the spray transfer technique produces sound and reliable welds most of the time - with an occasional imperfect one. The imperfections usually consist of over-penetration and violation of the bond line. A typical example of an acceptable and an unacceptable weld made by the spray transfer procedure is shown in Figure 11. The defect is local and is not circumferential in nature. This is further illustrated by Figure 12 which shows a transverse cross-section of a test block. Note that all welds except one are sound and without defects.

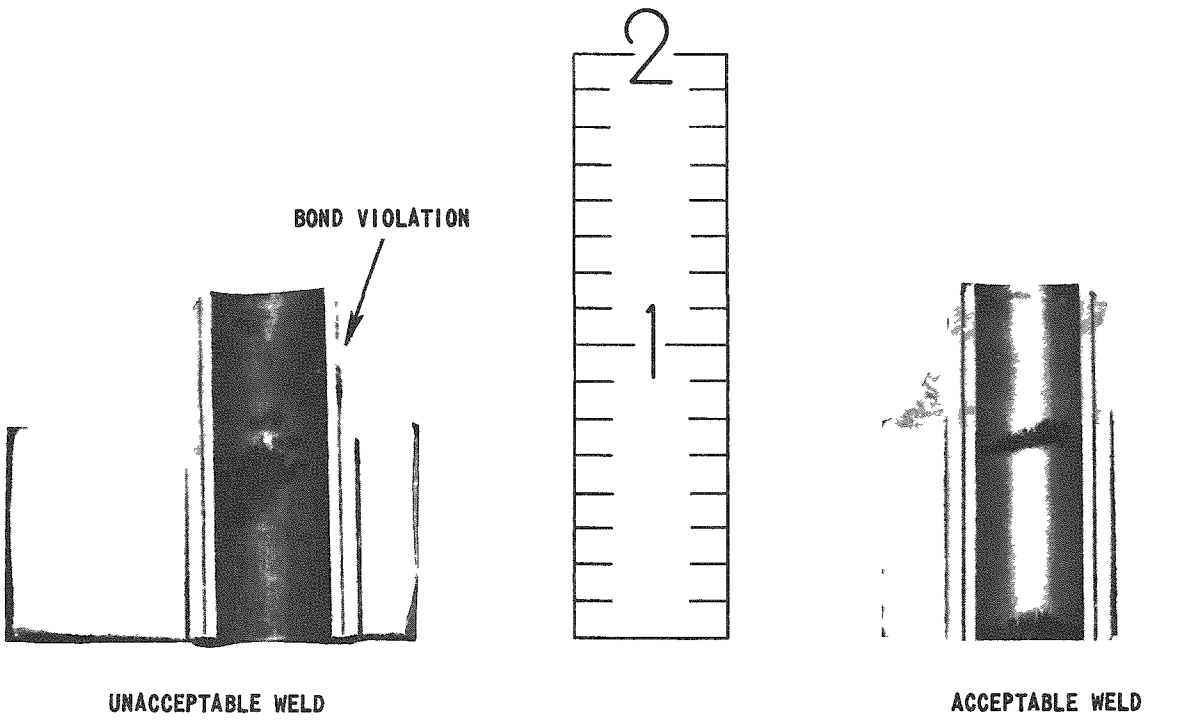

Figure 11

Superheater Tube-to-Sodium Tubesheet Weld Samples by Spray Transfer Method - Longitudinal Cross Section

The dip transfer method, discussed in the December Progress Report (ANL-6295), invariably incorporates a non-wetted area at the start of the weld cycle. Subsequent overlap of this area during the termination phase of the cycle does not wet the area. Defects of this type create a vertical leak path through the weld which would allow sodium to leak to atmosphere. A typical weld made by this technique is shown in Figure 13.

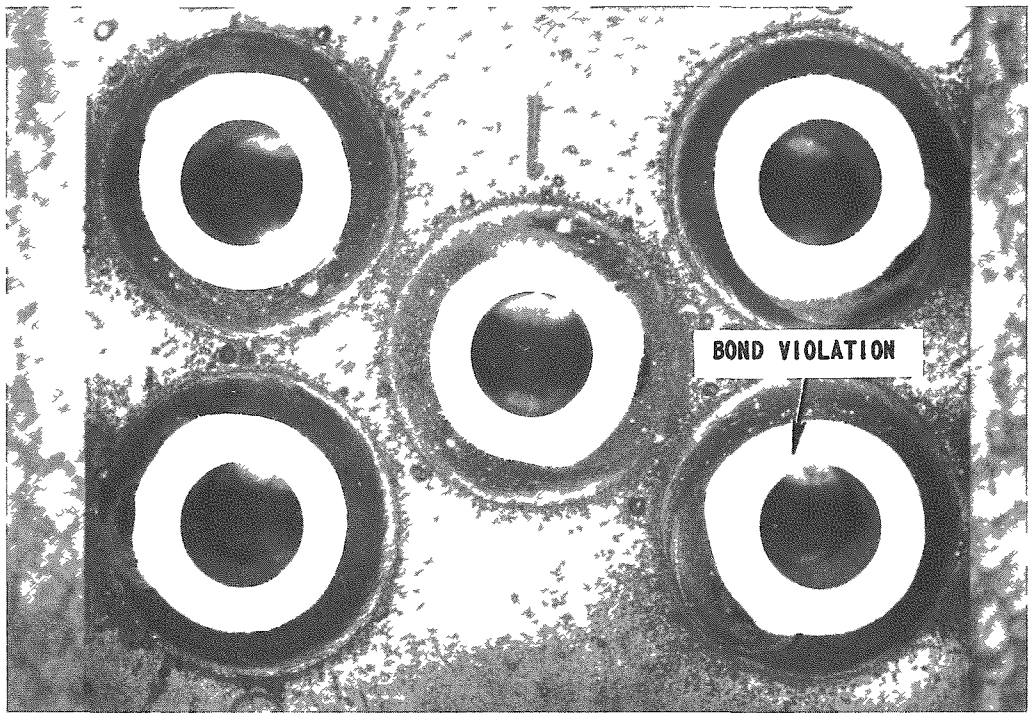

Figure 12

Superheater Tube-to-Sodium Tubesheet Weld Samples by
Spray Transfer Method - Transverse Cross Section

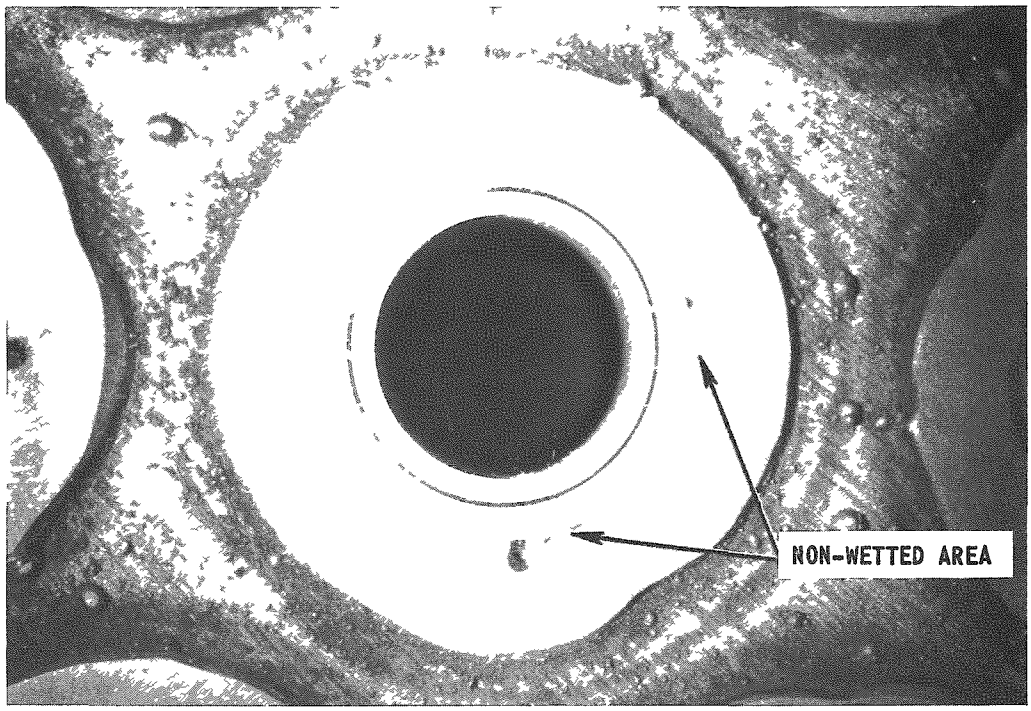

Figure 13

Superheater Tube-to-Sodium Tubesheet Weld Sample by
Dip Transfer Method - Transverse Cross Section

Alternate methods of superheating have been considered to avoid complete dependence on the solution of this problem and simultaneously avoid delay of the EBR-II Program. The use of evaporators as superheaters has been considered since an evaporator is quite similar to a superheater (see Figure 14). A spare evaporator is now available and all component parts, except tubes, are available for a second unit. A preliminary analysis was made of the performance of evaporator units as superheaters. The results indicate that it is feasible and reasonable to consider the use of evaporators as superheaters. Thus, a second evaporator will be assembled and the two units will be used as superheaters if the four superheaters now awaiting a solution of the welding problem have not been fabricated in time The use of evaporators for this service, however, is only an expedient and temporary means of resolving the present difficulty; work on superheaters for this facility will continue.

The results of the study are summarized in Figures 15 and 16. These results are based on the present evaporator design and a constant sodium inlet temperature of 810°F to the evaporating section of the steam generator. Possible modifications to the steam side to improve efficiency are yet to be investigated. Figure 15 shows the expected steam temperatures and the approximate per cent of moisture expected in the last stage of the turbine In addition, the expected moisture condition in the final stage of the turbine for 1250 psig, 840°F throttle steam conditions (design conditions) is shown for comparison. Figure 16 gives the calculated secondary sodium system pressure drop and sodium velocity in the 8 in. sodium inlet nozzle to the superheaters for the case of the two evaporators and the design condition (four superheaters).

With two evaporator-superheater units installed and at a reactor power level of 45 Mw, it is expected that a minimum steam temperature of 730°F will obtain. (It could possibly be made higher with modifications to the steam side to increase steam velocity. If a film coefficient equal to that for superheater units could be achieved, the temperature would be approximately 760°F.) This will result in a moisture content of about 18% in the last stage of the turbine. The secondary sodium system pressure drop will be approximately 26 psi. The sodium velocity at the 8 in. evaporator-superheater inlet nozzle is 13 fps.

A moisture content of 18% is considered reasonable for limited operation of the turbine. Sufficient information is not available to determine the precise effect of increased moisture content on the life of this machine. However, the EBWR unit has operated for more than 1.5 years at full power with a calculated moisture content of 17% in the final stage with no apparent deleterious effects.

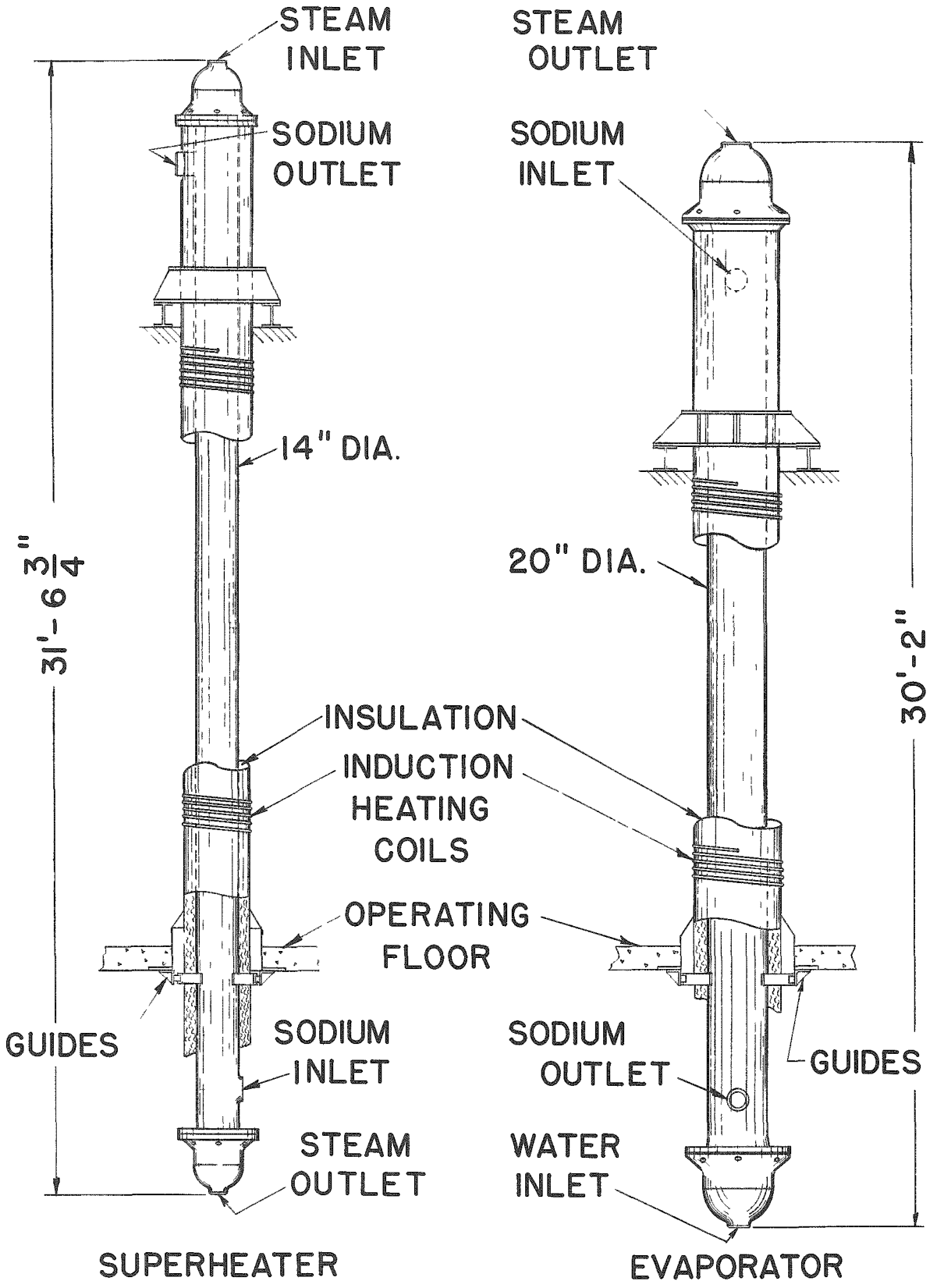

Figure 14

Similarity of EBR-II Superheater and Evaporator

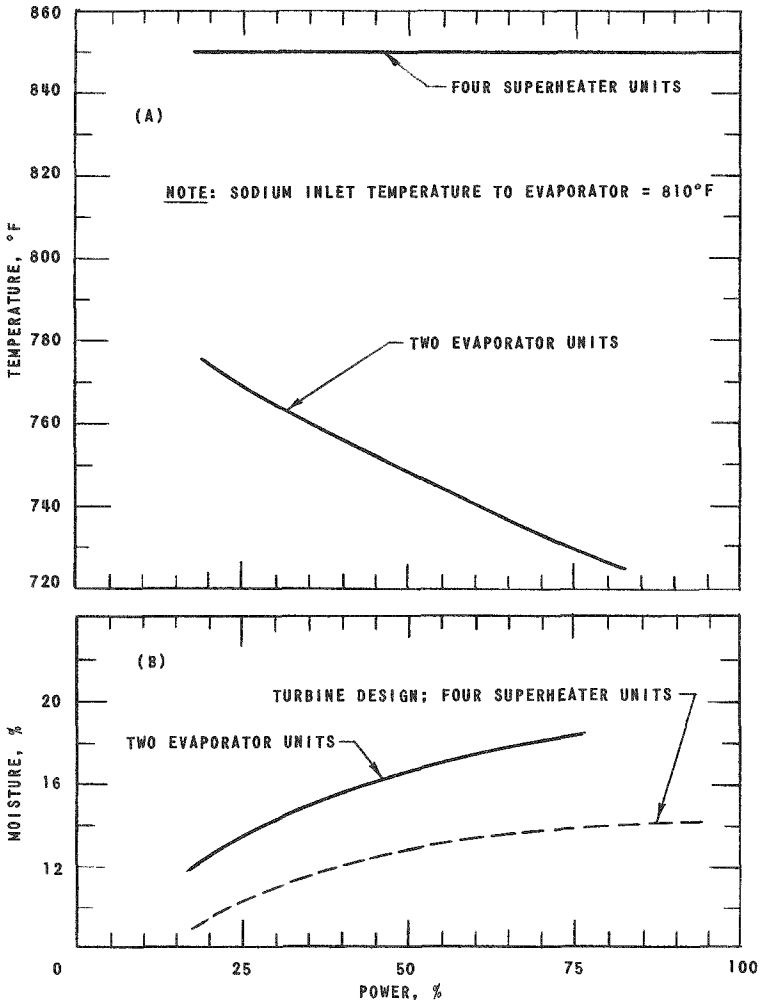

Figure 15

Calculated Performance of Evaporators as Superheaters
(a) Superheater Outlet Steam Temperature
(b) Moisture Content in Final Stage of Turbine

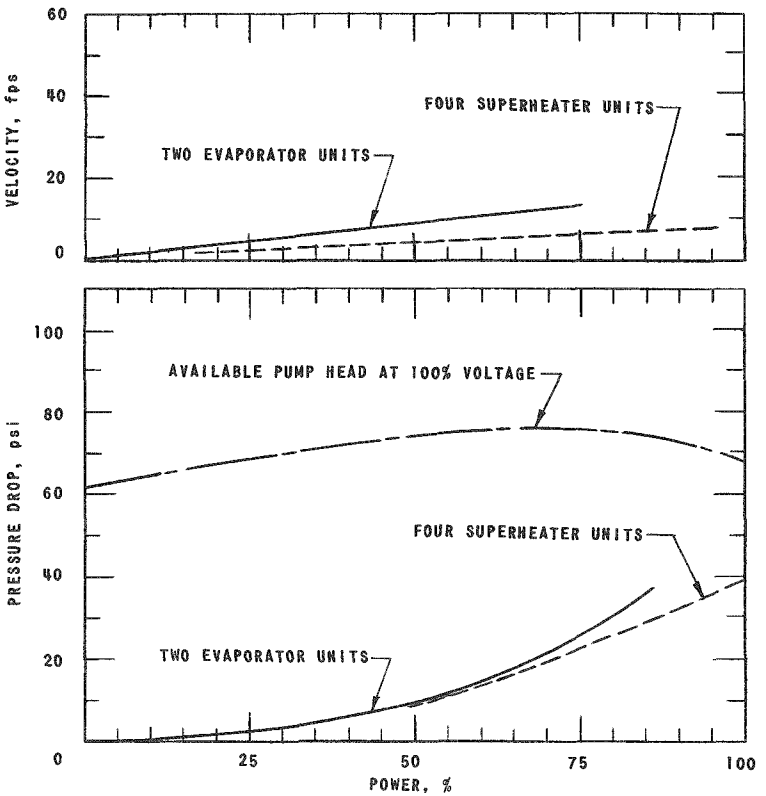

Figure 16

Calculated Pressure Drop in Secondary Sodium System and Sodium Velocity in Inlet Nozzle to Evaporator Units in Lieu of Superheaters

The use of evaporators as superheaters requires minor modifications to the support skirt, alignment guides, etc. It may also be desirable to increase the effectiveness of these units. This can be accomplished by installing a central core within each tube to increase the steam velocity which in turn reduces the steam side film resistance - the dominating thermal resistance.

The use of evaporators as superheaters will require only minor modifications to the building or supporting structure owing to the external similarity of these units as is evident from Figure 14. (The arrangement of these units was shown in Figure 2 of the Progress Report for January, 1961, ANL-6307.)

8. Component Development - Fuel Reprocessing Facilities

a. Fuel Cycle Facility Design and Testing - Equipment for the Air and Argon Cells is being designed, fabricated, and tested. Materials to fabricate the melt refining off-gas system have been ordered. Bids on the two melt refining furnaces have been received and are being evaluated. The drawings and specifications for the service sleeve feed-throughs have been sent out for bids. The feed-throughs are replaceable units which slip into the sleeve ports cast in the concrete. They contain service lines to equipment located inside the Argon Cell. A contract for an interbuilding coffin has been awarded to O G Kelley & Co. The coffin will be used to transport fuel assemblies between the reactor and the Fuel Cycle Facility Building.

A manipulator simulator will be used to check out manipulator control systems A cable is being made to allow the simulator to be plugged into the Center Pivot Tower and Power Inlet The simulator will greatly simplify the job of checking the power and control circuitry.

Testing of Timken roller bearings lubricated with radiation-resistant greases continues. A 300-hour running test of a bearing lubricated with NRRG-159 grease which had been irradiated to 4×10^9 rad was completed The bearing did not overheat. Tests are now underway in which the lubricated bearings are alternately irradiated and run. These tests will more closely approximate cell conditions since the grease will be in a thin film as it is irradiated.

b. Development of Remote Refabrication Equipment - Design of the refabrication equipment for the EBR-II Fuel Cycle Facility is estimated at 68 per cent complete. Designs were completed, specifications written and procurement requisitions submitted for injection casting furnaces and vacuum pressure systems during the reporting period. A temperature compensated, linear spring suspension system was received from Mettler Corporation and is being tested for sensitivity, resolution, and accuracy as a part of the 20 kg remote weighing balance and its readout equipment.

Two radiation resistant filters are being tested for flow resistance, efficiency and capacity to withstand instantaneous pressure differential across the filter medium. If satisfactory, these will be incorporated between the injection casting furnaces and pressure-vacuum systems to minimize gross contamination of the latter systems located in the subcell.

Designs and specifications were completed on the Fuel Cycle Facility version of the pressure leak detector. These were similar to those described in last month's Progress Report but are further refined for manipulator operation. Provision was also made for mass spectrometer connection if desired.

9 Process Development

a Melt Refining Process Technology - A second melt refining experiment using highly irradiated fuel alloy has been completed Ten per cent enriched uranium-fissium pins were irradiated to an estimated total atom burnup of 0.2 per cent. After 28 days of cooling, the fuel was melt-refined for three hours at 1400°C in a lime-stabilized zirconia crucible. The overall yield of purified product was 69 per cent

One of the major objectives of these melt refining experiments with highly irradiated fuel alloys is to examine the effect of the intense beta and gamma radiation on the performance of the process. In the first high-level melt refining experiment with material irradiated to 0.6 per cent burnup and cooled for 40 days, a product yield of 72.4 per cent was achieved. The yields of 69 and 72.4 per cent are to be compared to an average value of 89.5 per cent for several experiments with unirradiated material. It is desirable to establish whether these lowered yields are caused by more extensive oxidation of the uranium in the presence of radiation.

A uranium material balance for the first melt refining experiment with highly irradiated material is given in Table VI. Data from a parallel experiment with unirradiated material are also included for comparison. The quantities of uranium in the various fractions are based on the determination of their weights and on the uranium concentrations obtained by chemical analysis. All the uranium charged to the melt refining crucible was accounted for in the ingot and skull fractions. It is reasonable to assume that the uranium retained in the crucible as a component of the melt refining skull is present either as unpoured metal or as uranium oxide Furthermore, molybdenum, a noble metal in the process, appears in the skull only in the unpoured metal fraction, and the unpoured metal has the same composition as the ingot. Thus, molybdenum and uranium analyses of the various fractions permit the calculation of the percentage of uranium present in the skull as unpoured metal and the percentage present as oxide.

Table VI: Comparison of Melt Refining Yields with Irradiated and Unirradiated Fissium Alloy

Conditions: Charge size: 388 g irradiated alloy
360 g unirradiated alloy
Time: 3 hours
Temp: 1400°C
Crucible: line-stabilized zirconia
Atmosphere: 750 to 800 mm argon

	Irradiated Material	Unirradiated Material
Relative Weight		
charge	1.00	1.00
ingot	0.724	0.867
oxidized skull	0.324	0.132
Uranium Concentration, weight per cent		
charge	95	95
ingot	94	95
oxidized skull	82	79
Total Uranium, relative per cent		
charge	100	100
ingot	72	87
skull	28	13
Molybdenum Concentration, weight per cent		
ingot	2.6	2.5
skull	1.8	1.6
Analysis of Skull Fraction, relative per cent		
unpoured metal	22	10
oxidized metal	5.9	2.8
Total Uranium Accounted for, per cent	100	100

Two significant facts emerge from this calculation. First, about twice as much uranium was oxidized in the experiment with highly irradiated material. Second, the ratio of unpoured uranium metal to oxidized uranium in the skull was the same in both experiments.

There appear to be only two possible explanations for the larger amount of uranium oxidized in the active experiment; one is atmospheric contamination, the other is a radiation-induced increase in the reaction rate. Subsequent experiments with irradiated material will show which of these effects prevails. If the increase in oxidation is induced by radiation, it may be possible to reduce considerably the time of liquation necessary to produce adequate fission product decontamination by oxidative slagging. A desired reduction in the amount of uranium oxidized would accompany a decrease in the length of liquation time.

Evaluations of the extent of oxidation and of fission product decontamination for the second run are awaiting analytical results.

The second melt refining experiment provided preliminary data on the collection of particulate and volatilized activity in a filter and vacuum pump in the off-gas system. When the fuel was still below the melting point, evacuation of the melt refining furnace caused little activity to appear in these components. After the fuel was molten, however, survey meter measurements showed substantial amounts of activity when a small quantity of gas was pumped out of the system. The major constituent in the activity was identified as iodine.

Since the argon atmosphere of the EBR-II Fuel Cycle Facility is expected to contain small amounts of nitrogen, the possibility of nitride formation during the handling and storage of fuel pins is of importance. At nitrogen concentrations higher than one volume per cent in argon, the nitridation rate of fissium alloy was independent of nitrogen concentration. Below the one per cent level, the nitridation rate decreased rapidly as the nitrogen concentration was reduced. In one experiment a high-frequency coil was used to produce a silent discharge to simulate roughly the ionization produced by radiation. The nitridation rate in this case was increased approximately threefold.

A seventeen-hour exposure at 300°C of fissium pins to a 5 per cent nitrogen-argon atmosphere resulted in a low pouring yield on subsequent melt refining (2 kg scale). Work is continuing to determine if short (about 3 hr) exposures can be tolerated. Electrical energy input to a single fissium fuel pin equal to the energy expected from fission product decay heating for 2 per cent burnup and 15 days' cooling resulted in a pin temperature of around 120°C when cooled by natural convection.

b. Processing of Melt Refining Skulls - Two demonstration runs of the skull reclamation process have been completed. In these runs skull material was removed from full-size melt refining crucibles by oxidation. The oxide was then reduced at 800°C by a dilute (5 per cent) magnesium-zinc solution in the presence of a molten chloride flux. The uranium was precipitated from solution, first as a uranium-zinc intermetallic from the reducing solution, and then as uranium metal from a magnesium-rich zinc solution; and uranium metal was isolated by retorting to drive off residual solvent metals. Reductions were essentially quantitative as indicated by the absence of uranium in the molten salt flux Uranium metal of good appearance was isolated from each run. Many runs of this type are planned to develop process operating techniques and equipment, to expose process problems, and to demonstrate adequate uranium recovery and purification.

The present philosophy of operation in the demonstration runs is that of allowing heels of constant volume to remain in several process

steps. Therefore, high product yields can be shown only by repetitive operations. Process losses in waste streams were 0.2 per cent in the molten salt flux, 2.6 per cent in the zinc-rich supernatant solution and around 4 per cent in the magnesium-rich supernatant solution. Considerable reduction of losses in the two supernatant metal solutions is possible by making the phase separation at lower temperatures.

Investigations are being continued on the use of fluxes for the reduction of fissium oxide by zinc-magnesium solutions. Previous work had shown that group II cations in the flux are beneficial in promoting the reduction. For satisfactory reduction rates, magnesium ion has been found essential to the flux. Zirconium oxide appears to have a deleterious effect on the reduction in the presence of an air atmosphere, but little or no effect in inert atmosphere. In the reduction of fissium oxide, the refractory zirconium oxide from the melt refining crucible is not reduced. The behavior of fission product zirconium oxide, which might be expected to behave differently, is being examined. The process variables involved in the reduction of fissium oxide are being optimized in a special set of experiments.

As a preliminary step in the skull recovery process, it may be possible to remove noble metals by contacting the skull oxide suspended in a flux with liquid zinc. Uranium oxides are not reduced in the absence of magnesium. In a preliminary experiment effective noble metal removal was realized (Ru, 91.0; Mo, 83.6; Pd, 93.3 per cent) with a uranium loss of only 0.003 per cent.

It is likely that this extraction of noble metals into zinc will become a process step preceding the oxide reduction. Contrary to previous expectations, separation of molten zinc and the oxide-flux slurry by withdrawal of the slurry does not appear feasible because of inability to keep larger oxide particles in suspension.

A program has been started to study the chemical basis of uranium oxide reduction from molten halide fluxes. The initial work on this problem will consist mainly of spectrophotometric measurements to determine the uranium species in solution.

A second run to determine the extent of fission product volatilization during skull oxidation again showed iodine as the major evolved activity. A trace of tellurium activity was also found. Volatilization of ruthenium and molybdenum could not be detected.

c. Blanket Processing - Work was continued on the direct interaction of uranium and zinc to form the intermetallic which is subsequently decomposed by addition of magnesium to precipitate the uranium and dissolve the plutonium. Subdivision of the uranium by hydriding and dehydriding to increase the reaction rate of intermetallic formation by increasing surface

area shows promise. The formation of the zinc-uranium intermetallic appears to be complete within 10 hours. While plutonium-magnesium solutions appear reasonably stable in tantalum, a 10 per cent loss of plutonium from a magnesium solution over a period of 320 hours at 800°C indicates that there may be a slow introduction of impurities. Carbon, which is known to precipitate plutonium from magnesium solutions, is suspected to be the impurity involved. Special efforts are being made to establish the identity and the source of the impurity.

d. Liquid-Metal Processes for Plutonium Fuel - The selective extraction of fission products by liquid calcium is being investigated as a possible basis for processing uranium-plutonium-fissium fuel alloys. The addition of sufficient calcium to a solution of uranium and cerium in zinc is known to precipitate uranium metal quantitatively, leaving the cerium in solution. It is believed that the behavior of plutonium will parallel that of uranium, thereby permitting the separation of both uranium and plutonium from cerium, and probably other rare earth metals. The solubility of uranium in zinc-calcium solutions and the coprecipitation behavior of cerium with uranium were determined at calcium concentrations up to 23 weight per cent, where the uranium solubility at 725°C is about 0.001 weight per cent and the coprecipitation coefficient for cerium is zero.

e. Materials and Equipment Evaluation - Corrosion studies in the cadmium-zinc-magnesium system indicate virtually no corrosion of 1020 steel at 750°C by cadmium containing up to 15 per cent zinc. The presence of magnesium appears beneficial in reducing any attack. Preparations are continuing for investigating the corrosion of refractory metals in zinc-based systems.

Various methods were evaluated as a means of determining the extent of graphite penetration by fused salts. Chemical analysis, autoradiography, metallographic examination, and a staining technique all produced useful information. In a 75-hour exposure of CS graphite to a molten halide at 800°C, between one and five per cent of the void spaces in the graphite were filled by the salt.

10. Fuel Development and Fabrication - Core I

a. Fabrication and Assembly of Fuel and Blanket Elements - Over 10,000 fuel pins have been cast, assembled, welded and inspected. These operations are now completed. A total of 97 fuel elements (containing 91 fuel rods per element) have been assembled, and 14 control elements (containing 61 rods per element) have been assembled. These operations are also completed. There remain only 6 safety elements to be assembled. It is believed that all production can be completed early in March.

b. Testing - Studies of eddy current methods of testing EBR-II fuel elements are still proceeding. It has been found that a small close-coupled coil has the same sensitivity as coils of larger inside diameter. This permits a more compact sensing element. In place of the regular cyclograph testing equipment a differential pulsed eddy current coil and circuit are used for the study of hot fuel. This equipment was originally designed to determine the sodium level of the fuel elements, but the study will be extended to cover the flux characteristics of the entire fuel element.

c. Reaction Rate of Uranium and Stainless Steel - Safety considerations for EBR-II, Core I operation require information about the rate of penetration of Type 304 stainless steel jacketing by molten uranium. To that end a technique has been developed based on immersing capsules of desired wall thickness into molten uranium. Inside the capsule is an insulated wire which is shorted when the uranium breaks through the stainless steel wall thus giving an indication on a high speed recorder. Penetration rates are being determined at temperatures ranging from 1150°C to 1250°C using 0.010 in. and 0.040 in. wall capsules. Capsules with heavier wall thickness have been attempted but owing to insufficient mixing in the melt a local concentration of eutectic built up at the liquid-solid interface thereby slowing down the penetration. The thinner walled capsules are now being used in an attempt to minimize this effect.

Preliminary data indicate that molten uranium penetrates the 0.010 in. wall in the order of 0.5 sec and the 0.040 in. wall in the order of 2.5 sec, both being at a temperature of approximately 1200°C.

d. In-pile Performance of EBR-II Fuel - Although many irradiation experiments have been made on the EBR-II reference fuel, none of the tests exactly duplicates EBR-II core conditions in terms of such items as burnup rate and temperature and burnup distribution. The final answer on the performance of stainless steel clad uranium-5 w/o fissium alloy as a fuel for EBR-II must therefore be supplied by EBR-II itself. In order to obtain the best data on the behavior in the reactor of production fuel pins it is evident that measurements will be needed both before and after reactor exposure which are much more precise than routine "go, no-go" inspections.

A sample of 105 production pins was therefore selected for detailed measurements. The diameters of the bare pins were measured at close intervals. Lengths and immersion densities were also determined. After the pins were sealed into jackets the measurements were repeated on the assembly. Each element is identified with a serial number. The elements will be loaded into known core locations where outlet coolant thermocouples are located so that as much information as possible will be available on the in-pile temperature history of the elements. The upper

end closure was modified slightly in shape so that the measured elements can be recognized in the EBR-II reprocessing cell and returned to the Lemont site for detailed postirradiation examination.

Before any measurements were taken, each pin was given an extended anneal at 500°C to convert them fully to the alpha phase, so that the dimensions and density would not change significantly during subsequent sodium bonding operations.

11. Core II Fuel Development

a. Phase Study of U-Pu-Fs - As part of a program to better understand the metallurgical characteristics of the U-20 w/o Pu-10 w/o Fs alloy now considered to be the reference alloy for the second loading of EBR-II, a study of the equilibrium structures of alloys of uranium containing 20 w/o plutonium and 5 to 15 w/o fissium is in progress. The effect of additional zirconium on the phases present at various temperatures in these alloys is also being investigated.

Metallographic examination of these alloys heat treated at 602°C for 354 hours was completed during this period. Only slight changes in the microstructure were evident from a comparison with the alloys heat treated at 625°C.

b. Casting and Jacketing Technology of U-Pu-Fs Fuel Pins - The 107 EBR-II type castings reported last month have been machined into metallurgical evaluation specimens (53 two-inch samples and 73 pins 14.22 in. in length). Density measurements were made on the 2 in. pins by weighing in air and immersed in bromobenzene. The average density was 16.89 gm/cm^3, varying between 16.76 and 17.00 gm/cm^3.

Vanadium, steel, niobium, Nb-1 w/o Zr, Inconel X and Type 304 stainless steel jacket components were received for final assembly and sodium bonding of the requested samples. The Inconel X and Type 304 stainless steel specimens incorporated a 0.0005 in. thick vanadium barrier foil which has caused considerable difficulty in assembly of the specimens.

The design of the specimens required a heliarc girth weld of the 0.009 in. jacket materials to extended top caps. Because of difficulties experienced in controlling the P & H Welder at the low amperages required, a new Vickers Welder was obtained. After adaptation of the welding control timers to this machine and some modification, it was found that stable welding arcs below 4 amperes could be maintained. The various materials required quite different welding conditions with amperages ranging from the low value up to 12 amperes required to make satisfactory welds.

12. The Dry Critical Experiments

The preparation of the Dry Critical Report has been completed (to be published as ANL-6299). It describes the experimental program, experimental procedures and safety considerations relating to these experiments. The following description is a summary of the planned program.

The Dry Critical Experiments are essentially zero power experiments conducted in the EBR-II reactor system in the absence of the sodium coolant. These experiments have two general objectives:

(1) They will be conducted to obtain information relating to the performance of the system without sodium coolant for future comparison with information derived with sodium coolant. By comparison it will then be possible to ascertain various sodium effects on the neutronics of the system.

(2) They will be conducted to determine and/or verify certain operational data to permit modification or improvement of the system prior to the introduction of sodium into the system. After sodium filling it will be difficult to alter, modify, or convert many of the system components.

The experimental program includes the following:

(a) Neutron source and instrument response.
(b) The approach to critical.
(c) Neutron flux and power calibration.
(d) Reactivity measurements.

The relationship between neutron source strength and neutron detector response is important both to the conduct of the dry critical experiments and to the basic design of the reactor system as it relates to normal operation. The dry critical experiment will permit establishing this information reliably and safely because of the allowable use of special in-core instrumentation which is located external to the reactor. The additional instrument sensitivity available will permit flux measurement at very low reactor multiplication and simultaneous determination of the response of the normal instrumentation. This will provide an assessment of the normal instrumentation at a time when modifications can be made, if necessary.

The critical approach is an essential sequence of operation to permit a safe approach to the critical reactor configuration. The procedure will provide useful information pertaining to the critical approach when sodium is present. The determination of the dry critical mass, when compared later with the wet critical mass, will provide a measure of the total reactivity worth of the sodium coolant in the EBR-II.

The neutron flux and power calibration experiments will:

(a) Afford a relationship between instrument response and dry reactor power level;

(b) Determine leakage neutron flux from reactor blanket and neutron shield;

(c) Determine fission distribution in the reactor blanket, and

(d) Afford a calibration for high temperature fission counters for the wet critical experiments.

Several types of reactivity effects will be measured in order to evaluate design parameters as well as to achieve understanding of the neutronics of the reactor. These measurements include:

(a) Reactivity worths of control, safety and fuel subassemblies,
(b) The dry isothermal temperature coefficient of reactivity, and
(c) Calibration of an oscillator rod.

The various subassembly calibrations are determined to establish the reactivity worths for reactor operation and shutdown. The dry isothermal temperature coefficient provides a very interesting measurement of the sodium coefficient under uniform heating. The static calibration of the oscillator rod will provide a verification of the design several months prior to operation.

In connection with the planned EBR-II dry critical program, estimates were made of the radiation levels to be expected from EBR-II subassemblies after low power operation. About a week of cooling is required to reduce the radiation at the surface of an assembly to 10 mr/hr after operation at a total reactor power of 10 watts for 2 hr.

13. Hazards Evaluation

A solution of the EBR-II flow decay problem has been obtained. The results show that 10 sec after the failure of power to both primary coolant pumps the flow has decreased to about 20% of the design flow rate. This value is considerably higher than the value reported in ANL-5719* for electromagnetic pumps.

In order to establish the coolant flow rate distributions which exist in EBR-II in every row under natural circulation conditions, row by row calculations were made. The results of the calculations show that under

---

*"Hazard Summary Report - Experimental Breeder Reactor (EBR-II)," ANL-5719, May, 1957.

natural circulation conditions the ratio of the core coolant flow rate to the total coolant flow rate increases about 3% from those at the design flow rate. Estimates show that only about 27% of the design power can be removed by natural convection before local boiling of sodium begins in some rows of the core, and practically complete blockage of the flow will occur in a row as soon as net boiling begins in the core.

III. STUDIES AND EVALUATIONS (040116)

A. Evaluation of Pebble Bed Reactor Designs

Two reports on pebble bed reactor designs prepared by ORNL, one a 10-Mwt prototype experiment and the other a 330-Mwe power plant design, were submitted to the Laboratory for review. It was requested that the review develop "quantitative and qualitative" answers to the following questions.

(1) Does the reactor concept appear technically feasible? Where are the questionable areas and how critical are they?

(2) Of the research and development work that would appear necessary to prove this concept, which of the research and development is most critical? What type of results should be obtained from the research and development in order to proceed further with the concept?

(3) What is your judgment of the estimated cost of research and development and the time and manpower requirement necessary to carry it through?

(4) How reasonable are the cost estimates presented?

(5) In view of the economic promise of the concept, in your opinion is a program leading to the design and construction of an experiment and a prototype justified?

The following are general conclusions from the review:

(1) Major uncertainties with the pebble bed reactor concept are associated with the performance of the fuel and the fuel handling machinery. Data on fission product release rates and irradiation stability of fuel balls are not yet available to determine the economic status of the concept. Much experimental work is required to evolve a satisfactory fuel handling scheme.

(2) Development of remote maintenance equipment and procedure are required for the reactor vessel internal structure.

(3) Because there is not available a sufficient amount of data some uncertainty exists in calculating exact heat transfer coefficients and pressure drop. Heat transfer and pressure drop tests with packed beds composed of large diameter spheres would be required for experimental verification. The accuracy of the pressure drop correlation is especially important because the reactor power is essentially inversely proportional to the square root of the friction factor.

(4) Additional design and experimental work is needed to establish the costs for a reactor system having a tolerable helium leak rate.

(5) Other design problems exist which, because they are not particularly unique to the PBR, will be solved on other reactors such as the EGCR.

B. Committee on Organic Reactor Technology

An ad hoc committee has been established at the Laboratory to evaluate the current status of organic reactor technology and to determine those areas of research and development work which are now lacking or are inadequate for the successful development of the reactor concept. A report is due on April 1, 1961. The Committee intends to examine the work already done, the problem areas, the work planned, and to define the areas requiring investigation. The Committee is composed of ANL personnel who are experienced in chemistry, petroleum processes, metallurgy, and the engineering phases of reactor design.

An information meeting was held in Idaho Falls to acquaint the members of the Committee with the organizations involved in the organic program, and the key problem areas. Visits were also made to OMRE, Phillips Petroleum Company at the National Reactor Testing Station, and Atomics International at Canoga Park, California.

IV. REACTOR SAFETY (040117)

A. Thermal Reactor Safety Studies

1. Fuel-Coolant Chemical Reactions

Knowledge of the nature and extent of chemical reactions with nuclear reactor core metals that may occur in pressurized water or steam is essential to safe operation of reactors. The principal laboratory procedure uses a condenser discharge to provide almost instantaneous heating and melting of metal wire in water or steam. The energy input to the wire indicates reaction temperature; the transient pressure measures reaction rate; light emission indicates time-temperature behavior; hydrogen generated gives extent of reaction; and particle size of the residue indicates the surface area exposed to reaction. A second method consists of heating the metal inductively and then subjecting it to a steam pulse to induce a metal-steam reaction.

The reaction of aluminum with uranium oxides is being studied by use of differential thermal analysis.

Studies of the kinetics of metal-water reactions under reactor incident conditions are being made in the TREAT reactor.

The series of runs with zirconium wires in the high pressure condenser discharge apparatus are continuing. Further runs in water at 310°C (1500 psi vapor pressure) have been made. Results continue to indicate that the reactions are identical to those carried out in water at 100°C (15 psi) and 200°C (225 psi).

A series of zirconium runs in water at room temperature have been undertaken in a reaction cell having a large vapor volume. This series of runs should be a critical test of current theoretical ideas of the cause of the increase in reaction at elevated water temperature. Hydrogen generated by reaction will not build up a significant pressure in the large reaction cell and thus should not suppress the reaction if it is limited by gaseous diffusion.

Differential thermal analyses were made to establish if thermite reactions occurred between aluminum and 10 percent U_3O_8 or between stainless steel and 10 percent uranium dioxide cermet fuel pins. Specimens were heated at 10°C per minute to a temperature of 1200°C in an argon atmosphere. No exothermic reaction occurred. X-ray diffraction analysis following heating indicated that only the original metals and uranium dioxide were present. Studies will be continued with pressed powder mixtures that were not sintered.

Two experiments with 64-mil, 93 percent enriched uranium wires were completed as part of the eleventh series of TREAT, metal-water meltdown tests. The wires gave 33.2 and 50.2 percent reaction with water using reactor periods of 440 and 152 milliseconds, respectively. Average energy input was 554 calories/gram uranium (integrated power of 100 megawatt seconds, peak power of 47 megawatts). These are the highest extents of metal-water reaction that have been obtained in TREAT runs made to date.

2. Kinetics of Oxidation and Ignition of Reactor Materials

Studies are being made of the oxidation and ignition kinetics of the metals uranium, zirconium, and plutonium in order to provide information leading to an understanding of the reactions. This knowledge should make it possible to minimize the hazards associated with handling these nuclear reactor materials. Isothermal oxidation on microscope stage, shielded ignition, burning curves, rate of propagation of burning foil, and burning temperatures are the techniques being used. In the continuing study of ignition and burning of uranium, zirconium, and plutonium, more emphasis is being placed on the burning process. Burning propagation rate studies provide a useful tool to observe the effects of many variables. Addition of halogenated hydrocarbons to air in which uranium or zirconium specimens are burning has been observed to lower markedly the burning propagation velocity. This effect is being investigated.

Efforts to elucidate the relationship between isothermal oxidation rates and burning curve ignition temperatures are continuing. Equations were devised to represent the self-heating of a uranium specimen and the heat exchange with the program furnace in the burning curve apparatus. The equations were solved for two particular cases using a stepwise hand calculation. A calculated burning curve for 8.5-mm cubes agreed with an experimental curve up to about 400°C, where apparently the kinetic rate decreases and cannot be represented by isothermal rate data extrapolated from lower temperatures. Better agreement is expected for specimens of higher specific area.

Studies of the ignition and oxidation of uranium powders are continuing. The effect of pre-oxidation of a coarse (-20 +25) spherical powder was shown to be similar to that reported for a fine powder last month, namely, initial lowering of the ignition temperature until the surface was oxidized, then a gradual increase in ignition temperature.

Plans have been formulated to study several homologous series of halogenated hydrocarbons to determine the mechanism of inhibition of burning propagation along foil strips. A two-color photoelectric pyrometer will be used for the simultaneous measurement of maximum burning temperature and propagation velocity.

Because some uranium blanket plates for use in ZPR-III have been observed to be deteriorating and crumbling, a few ignition experiments were performed to appraise the pyrophoricity hazards. Samples of the deteriorated material ignited at 115°C in either oxygen or air. This is slightly lower than was obtained for compacted irregular powder which had a measured specific area of 6,500 sq cm per gram. ZPR-III personnel have completed the inspection of all affected plates. The storage of the affected samples in argon in pint cans in a sand bath should avert hazard.

B. Fast Reactor Safety Studies

1. Stability of Fast Reactors: Calculation of Oscillating Temperatures in the EBR-II Core

A necessary part of reactivity feedback calculations for oscillator experiments in EBR-II is the calculation of the oscillating temperatures of a fuel pin and its associated coolant as a function of axial position. An elegant solution to this problem was given by Storrer in APDA-132* for the case of an unclad fuel pin. In Storrer's calculation axial heat conduction in the pin is neglected, and a sinusoidal time variation of the power is assumed, making it possible to separate out the time dependence. The pin temperature is determined as a superposition of two radial functions, one proportional to the heat source in the pin and the other to the local coolant temperature. These functions are the same for any axial position, their normalizations at any position being determined by the local heat source and coolant temperature. The latter is found by a simple axial integration when the two radial functions are determined. Storrer's solution was extended by a two-region clad pin and this solution was coded for the IBM-704 as Code IDO-102. Temperatures are multiplied by reactivity coefficients in this code to give the reactivity feedback. Some further modifications and extensions of this code to make it more suitable for use with EBR-II have been made, the result being designated as RE-233. These modifications include possible use of several representative pins to take account of sodium flow orificing in different subassemblies, and the provision for different pin characteristics in core and axial blankets.

While the above calculation promises to be completely satisfactory for the case in which the feedback is linear in temperature and the kinetics equations are linear, it cannot be used to deal with non-linearities which would cause the power to become a Fourier series in time rather than to be proportional to a single sinusoidal component. To deal with non-linearities the analog computer, which operates in the time domain, is suitable. Such non-linearities could arise for example from bowing, phase transformation in the fuel, or non-linearity in the kinetics equations. Because of the limited

---

*Fred Storrer, "Temperature Response to Power, Inlet Coolet Temperature and Flow Transients in Solid Cycle Reactors," APDA-132, June 5, 1959.

capacity of the analog computer compared to a digital computer a more simplified temperature calculation is necessary. For this reason a "radially lumped" pin model has been studied in which the two radial temperature functions in a pin discussed above (which for the Storrer-type solution are linear combinations of Bessel functions of complex argument) are approximated by the steady state solution. The function proportional to the coolant temperature in this case is a constant, the coolant temperature itself. Results obtained by this approximation by a four section finite difference calculation, give surprisingly accurate results up to $\omega = 5$ radians/sec. It thus appears that sufficient accuracy with this model could be obtained for a non-linear problem in which the driving frequency ranged up to 1 or 2 radians/sec and in which one or two harmonics were present.

2. Core Meltdown Studies: TREAT Program

In-pile meltdown experiments are being performed in the TREAT reactor in order to survey types of fast reactor fuel element failure and the associated movement of fuel element materials, as well as to determine the mechanisms producing such phenomena.

a. EBR-II, Mark-I - Dry Samples - Six 3% enriched EBR-II, Mark-I elements, contained in capsules with tantalum neutron absorbers at the sample ends to shape the neutron flux axially, were irradiated in TREAT during the report period. These are the first tests on EBR-II pins with axial flux-shaping to approximate the flux distribution expected in EBR-II. One sample has been examined. It registered a maximum cladding temperature of 850°C, and in agreement with results of previous experimentation on similar elements subjected to an essentially uniform axial neutron flux, no damage was noted either to sodium bond or stainless steel cladding. A detailed gamma scan was made to determine activation of the element as a function of axial position in order to confirm the results of the two steady-state calibration runs. The plot of activation vs. axial position was of the desired "chopped cosine" shape and had a maximum-to-minimum ratio of about 1.6. Instrument records of the remaining five elements exposed indicated that failure occurred; the total energy inputs to the five covered a range estimated to extend from that necessary to produce extensive failure to a figure approximately 70% greater than that necessary for extensive failure.

b. EBR-II, Mark I - Samples in Molten Sodium - The four EBR-II, Mark I elements to be used in the second series of experimentation on meltdown in stagnant sodium (see Progress Report, January, 1961, ANL-6307) have been encapsulated and received at TREAT. Two of these capsules contain a fuel pin with a chromel-alumel thermocouple mounted within the fuel rod a full 1.5 in. from the top. The other two capsules have the thermocouples mounted in the bond sodium at a point halfway between the top and bottom. Irradiation of these capsules will demonstrate the best method of registering fuel temperature during a transient.

c. Fermi-I - Dry Samples - Five half-length 10% enriched Fermi-I fuel elements subjected to flattened power transients were examined. Each sample was instrumented with two fast-response Pt - Pt Rh thermocouples spot welded to the cladding. Both thermocouples failed before the end of the transient in four of the experiments; in the remaining case (Sample 2) one thermocouple failed. None of the five specimens was observed to have the small cladding cracks previously noted for short (about 0.3-sec duration) power bursts. All samples showed the irregular appearance with occasional flattened areas characteristic of a fuel pin which melted during the experiments, the melted pin taking the shape dictated by the least strain in rapid cooling. A summary of the results is given in Table VII. In no case was there any indication of violent ejection of melted fuel.

Table VII: Summary of Results Obtained on Fermi-I Fuel Elements

Sample No.	Max. Recorded Temperature, °C	Nominal Reactor Constant Power Level, Mw	Reactor Energy Release, Mw-sec	Sample Condition
1	1205	4	53	Pinhole failure at a thermocouple weld.
2	1260	8	51	Pinhole failure at a thermocouple weld.
3	1305	16	42	No failure.
4	1160	16	48	Two pinhole failures at thermocouple welds; element badly warped.
5	1165	16	53	Extensive failure (not associated with thermocouple welds).

3. Theoretical Analyses

As a consequence of experiments in TREAT, the rapid expulsion of fuel by bond sodium from failed fuel pins is under investigation as a potential cause of a nuclear excursion. The analysis is currently aimed at the EBR-II reactor, and a new, more powerful time and space dependent heat transfer computing program is under development. Its completion will make possible theoretical estimates of the time and location of failure of fuel pins as a function of core radius during some abnormal condition leading to severe overheating of the core. Estimates can then be made of the rate of reactivity change resulting from this condition.

Studies have been continuing on the potential autocatalytic features of meltdown configurations during the early portions of an explosion. A two-dimensional perturbation code to assist theoretical studies of this nature has been devised and will be written as soon as the two-dimensional CURE-M adjoint code has been checked out.

An interesting aspect of the explosive nature of hypothetical nuclear reactor explosions is the dependence of energy yield on the shape of the power distribution in the core. Early, approximate, theoretical estimates of this effect are in some doubt. The powerful AX-1 coupled hydrodynamics-neutronics computing program is being utilized to study this effect accurately, taking various quantities such as the initial reactivity and the relative density of core and blanket as parameters.

4. Component Development

a. Meltdown Instrumentation Development-Pressure Transducers - A variety of tests have been performed in TREAT on the response to a high radiation environment of miniature pressure transducers of types of interest for meltdown experimentation. Data collected has incorporated results from transducers used as pressure monitors during actual meltdown experiments and from instruments run separately. The principal effort thus far has been expended on strain gage devices, and the following review is based on results on this type of transducer

Two types of sensitivity are apparent· an output signal component approximately proportional to reactor power, and a signal component approximately proportional to short-term reactor energy release. The power sensitive component has been ascribed to ionizing effects in lead insulation, and can be made negligible by use of a-c carrier excitation of the strain bridge. Carrier excitation did not produce any appreciable change in the energy-sensitive component. The magnitude of both components as a function of reactor power or energy varied widely from transducer to transducer, but were typically of the order of 10-25% of the full-scale output of the instrument.

Attempts to reduce the energy-sensitive signal by gamma shielding were unsuccessful. However, shielding with cadmium reduced this component by approximately a factor of three. The relative insensitivity of the energy-sensitive component to reactor energy release during the power tail of a transient led to a check of the decay of this response as a function of time after a transient. Typical decays showed a logarithmic decrease with an e-folding time of the order of 10 to 25 sec. This work is continuing.

b. Transparent Meltdown Facility Development - Tests were made at TREAT of the prototype transparent meltdown facility and its accessories. A gas handling system to be used for purging air from the inside of the facility and to provide an inert cover gas for the sample inside the transparent assembly was built and installed at TREAT. The handling coffin for use in insertion and removal of the facility from the reactor was built, checked out, and adjusted to provide proper vertical alignment with the reactor core slot. Camera checks were made of camera and light alignment, as well as transparent facility alignment and light reflections. Necessary alignment

adjustments noted were minor. A low power reactor calibration run was made with a 6% enriched EBR-II, Mark-I element inside the transparent assembly.

Shortly before beginning the prototype testing it was discovered that the alignment fins on the Zircaloy outer subassembly of the facility were too large. During subsequent grinding operations on the fins, leaks were opened in the outer subassembly, and plans for actual transient meltdown tests, using the facility with a metal cover over the outer window, were postponed pending construction of a new outer subassembly. As a result of the handling experiences and additional reviews of facility design features, it was decided to modify the outer subassembly design for greater strength, to make future outer units from stainless steel, and to incorporate a Zircaloy inner liner into the modified design in order to retain the resistance to attack from molten uranium afforded by the refractory metal. Fabrication of seven additional inner subassemblies is underway. Design changes in the outer subassembly shop drawings are being made.

c. Preparation for Experiments with Irradiated Samples - Irradiation in MTR has been completed on twenty 6% enriched, EBR-II elements set aside previously in preparation for meltdown experiments on pre-irradiated samples. The samples are now in storage to permit fission product activity to decay. Preparations are underway for MTR irradiations of twenty, half-length, 10% enriched Fermi-I elements.

d. Large Sodium Loop Development - Design work continued on the large sodium loop, based on the reference design completed earlier (see Progress Report, December, 1960, ANL-6295). A study of the design has been completed and suggestions on loop design features are being reviewed.

V. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Applied Nuclear and Reactor Physics

1. Experimental

a. BORAX-V Superheater Critical Experiment - In connection with the planned insertion of BORAX-V superheater elements into the ZPR-VII facility, a proposed layout showing how central and peripheral superheater regions might be located in the ZPR-VII system containing the Hi-C structural components has been completed. If either of these arrangements is adopted it will be necessary to provide additional grid and support members. Drawings showing these possible configurations have been prepared for inclusion in the Hazards Summary Report Addendum to be prepared.

As part of the hazards analysis, the effect of flooding of a superheater zone located both centrally and at the periphery of a water-moderated system has been calculated. For twelve BORAX-V superheater elements surrounded by a water reflected region of Hi-C fuel with water to uranium volume ratio of 3, the flooding of the dry superheat region would result in a reactivity increase of 6.8%. However, if the BORAX-V boiling region fuel is substituted for the Hi-C fuel, a higher water to metal volume ratio results and the reactivity associated with flooding is only 4.2%. If the same spacing of Hi-C fuel is used with a peripheral superheater, the reactivity effect due to flooding is only 2.7%.

The magnitude of these reactivity effects indicates a possible need for a large central safety rod in ZPR-VII capable of overriding an accidental influx of water into the voided central region. This would supplement the water dump and provide a second mechanism which could certainly shut down the system if the superheat zone became accidentally flooded during a reactor run. Preliminary calculations were inconclusive in regard to whether shutdown could be accomplished by control rods located outside of the superheater region. The requirement of a central rod is a tentative conclusion, and additional calculations of control rod worths are in progress.

b. Pile Noise Analyzer - In ANL-6269 (Progress Report, November, 1960), plans for improving the pile noise analyzer used in a number of measurements over the past years were discussed. This was prompted by a growing external interest in the use of this technique for the determination of prompt neutron lifetime. A novel integrator making use of a phantastron circuit to produce a square pulse whose duration is proportional to input signal voltage was described. Linearity and stability tests of this integrator have now shown that it performs remarkably well. This device is linear to within 0.3% of its full-scale reading and has a long term stability of the order of 0.2% drift over a one-week period.

The new pile noise analyzer has been built and is being subjected to extensive performance tests. The electronic tuned circuit analog has given the improved performance expected at very low frequencies, but its performance at higher frequencies has not been as good as desired. This is probably due to reduced operational amplifier gain at these higher frequencies. However, in practice this presents no real limitation on the performance of this equipment, since this is beyond the range normally used for the pile noise measurements. If necessary, measurements at the higher frequencies can be made quite well by inductance-capacitance tuned circuits.

c. Measurement of $\overline{\nu}$ for Thermal Fission of U^{235} - In last month's Progress Report, certain conclusions drawn from preliminary measurements at the thermal column of CP-5 were stated. These indicated the need for additional equipment in connection with the measurements to be resumed next month at the thermal column. Preparations in this direction are well underway. The necessary monitor chambers and associated electronic circuitry are being assembled and the collimators for the slowing-down tank are now under construction. The $MnSO_4$ solution counter assembly required for the detection of manganese activity is being fabricated, and modifications of the manganese bath system are nearing completion.

The solution counter assembly consists of an annular region surrounding a 4-in. sodium-iodide crystal. This annulus contains nearly 2 liters of manganous sulphate solution. Five low noise EMI 5255S phototubes placed around this solution view the Cerenkov light emitted by the Mn^{56} betas. The technique for detection and determination of absolute disintegration rates is beta gamma coincidence counting. Attention has turned recently to an evaluation of the beta-counting efficiency for the beta-gamma coincidence counting of the manganese solution.

Counting rates resulting for various experimental conditions have been observed and evaluated. These conditions included a comparison of reflectivity by aluminum and magnesium oxide, counting with slow amplification (Argonne circuit A-61) and faster amplification (Argonne circuit A-137), and a range of depth of solution. In terms of usable values, the best efficiency obtained for a 200 gm per liter aqueous solution of $MnSO_4$ was approximately 3% for a 250 ml sample when the EMI 6255S photomultiplier was operated at a voltage which resulted in a background counting rate of one count per second. Prevention of pile-up by using a moderately fast rising system results in an improved signal to background ratio over long-time-constant systems. An additional decrease in background may be achieved by placing the glass envelope of the photomultiplier tube at ground potential to attract positive ions, and by some modification in the shielding.

In a few important experiments in the $\overline{\nu}$ measurement, the specific activity of the manganese solution will be quite low. Consequently, some effort has been made to determine if the activated solution withdrawn from

the manganese bath can be mixed with a liquid scintillator in order to improve the signal pulse height relative to noise. A brief effort was made to test the miscibility of the manganous sulfate solution in various liquid scintillators. Phase separation occurred in xylene, toluene and dioxane when more than a few percent of the inorganic salt solution was added to the solvent. To achieve a worthwhile increase in the counting efficiency it would be necessary to introduce at least 20% of the solution into the scintillator. Attempts to suspend the solution in a thixotropic scintillating gel were also unsuccessful, and so this approach has been abandoned.

d. Modifications to ATSR - The Argonne Thermal Source Reactor (ZPR-IV') is fulfilling an increasingly important role as a source of neutrons for a variety of reactor physics experiments. It is currently being used for (a) an investigation of the performance of a mechanical velocity selector which will be used in connection with thermal spectra studies; (b) an experiment to determine the perturbation produced in the thermal and near-thermal spectra due to the presence of a beam hole; and (c) an experimental program to determine infinite dilution resonance integrals for a number of fission products. For these uses, a number of modifications are necessary and certain characteristics of the ATSR must be established. In addition, it has become desirable to repeat and extend the preliminary calibration measurements made when the ATSR was first placed into operation. This type of activity has assumed major proportions during recent weeks.

In an effort to provide better shielding for the ATSR so that operational restrictions on the use of the facility imposed by the radiation level in its vicinity can be minimized, several additions have been made to the shielding of the reactor. A reduction of a factor of two in the neutron background has been achieved by dissolving 35 kg of boric acid in the water which fills the shield tank surrounding all but one face of the core, by the addition of cadmium over the holes in the top of the reactor, by shielding the thimbles which contain flux level detectors, and by the addition of cadmium shielding to the leakage face of the reactor. The reduction of gamma flux was approximately proportional to the reduction in the neutron flux. It is concluded that any additional reduction of the leakage from ATSR will require extensive shielding changes.

Several tests have been made of the reactor with a cadmium-covered tube inserted in the center of the core. This required the design of a special fuel element and additional calibration runs to insure that the excess reactivity available was consistent with operational requirements and did not exceed the allowed margin from the hazards standpoint. Within this cadmium-lined void tube, measurements have been made to determine whether a $1/E$ variation of the neutron spectrum exists in the central hole. This is of importance in connection with the planned program of resonance integral determinations. Four detector materials having a dominant single

resonance were used in this investigation, Au^{197} (4.9 ev), As^{75} (45 ev), Co^{59} (132 ev) and Mo^{98} (490 ev). An appreciable deviation from $1/E$ spectrum was indicated by the initial measurements. Two sets of foil irradiations and radio-chemical procedures to determine the activation yielded results in good agreement among themselves. However, the indicated flux per logarithmic energy increment determined by cobalt and arsenic differed by a factor of two from that determined by gold and molybdenum. It is likely that self-protection in the gold foil accounts for some part of this difference, and so these measurements will be repeated with thinner gold foils.

The reactivity contribution of the plutonium contained in the plutonium-beryllium startup source has been determined. This source contains 81 gm of plutonium, and during startup it is displaced 12 in. from its location in the shield tank to a position in the light water reflector (which surrounds the ATSR core). There are two effects associated with this source. In addition to the neutrons added to the reactor by its insertion, the fissionable isotope contributes a positive reactivity effect owing to induced fissions and the subsequent increase in the number of neutrons in the system. The magnitude of this latter effect depends strongly upon the flux existing at the location of the source and its proximity to the core.

Analytical investigation of the kinetic performance of the reactor following the step insertion of the neutron source shows that, following an initial transient, the neutron level in a reactor that is just critical will increase linearly. On the other hand, the step insertion of a positive reactivity will, after the initial transients, cause the neutron level to increase exponentially with a positive period. For a plutonium-beryllium source the increase in neutron level will exhibit a combination of these two effects.

e. Neutron Physics

(1) Fast Neutron Capture Cross Sections - The neutron capture cross section of I^{127} has been measured by an activation technique. The neutron energy range covered is from 200 kev to 1700 kev. The measured cross section agrees reasonably well with the measurements of Bame and Cubitt[1] but is larger than the ORNL liquid scintillator results[2] and smaller than the Los Alamos liquid scintillator results.[3] The absolute value of the cross section is based on the U^{235} fission cross section, and on the Au^{197} activation cross section measured at this laboratory. The tentative value for E_n = 200 kev is $\sigma_i(n,\gamma)$ = 2.53 barns.

---

[1] Phys. Rev. 113, 256 (1960).

[2] "ORNL Liquid Scintillator," Phys. Rev. (To be published).

[3] Diven, et al., Phys. Rev. 120, 556 (1960).

(2) $\overline{\nu}$ Measurements Relative to $\overline{\nu}$ (U^{235}) - Apparatus has been set up to measure the average number of neutrons emitted ($\overline{\nu}$) in the neutron-induced fission of several fissile nuclei relative to the number emitted in the thermal neutron-induced fission of U^{235}. The incident neutron energy is variable from thermal to about 1.6 Mev. Thus the quantity $d\overline{\nu}/dE$ can also be obtained. Both $\overline{\nu}$ and $d\overline{\nu}/dE$ for U^{235} and U^{238} fission are of importance in reactor design.

Preliminary measurements at 100 kev and 1.6 Mev incident neutron energies resulted in values of $\overline{\nu}(U^{235})/\overline{\nu}(U^{238})[E_n = 1.6 \text{ Mev}]$ and $\overline{\nu}(U^{235})$ 100 kev/$\overline{\nu}(U^{235})$ 1.6 Mev having a standard deviation of about 3%. These measurements indicate that the method is capable of precision of about 1%. With that objective in mind, work with thorium, U^{235}, U^{238}, and Cf^{252} is in progress.

f. Accelerator Instrumentation

(1) Tritium Target Assembly - A target assembly designed for the production of monoenergetic neutrons by the p-n reaction in tritium is being fabricated. It is hoped that this target unit will provide a neutron source with an energy spread of less than 35 kev and accept an incident proton beam of about 20 μamps without destruction of the requisite thin target windows. Owing to the low Q value the use of the tritium p-n reaction as a source of neutrons will extend the effective neutron energy range of the existing accelerator. Moreover, the reaction is free of gamma radiation thereby removing some of the more troublesome background problems present in neutron cross-section measurements.

(2) Ion Bunching System - Most of the problems experienced with the ion bunching system, utilized in cross-section measurements employing time of flight techniques, are associated with the high frequency sweep portion of the system. Previously the failure of a supplier to deliver special insulators within six months of the prescribed delivery date substantially delayed the program. The same insulators are now failing at an alarming rate. Moreover, they appear to be constructed of very poor materials for use in radio frequency fields. The net result of the difficulties is that it has been necessary to fabricate special insulating units in the Laboratory's shops. These appear to have considerable promise at this time. Unfortunately, instrument problems of this nature consume a very large amount of research talent.

2. Theoretical

a. CP-5 Hot Spot Factor - Neutron flux peaking effects resulting from the use of longitudinally seamed fuel tubes in CP-5 have been calculated using two-group diffusion theory. A correction factor for the theoretical results has been obtained by comparison with experimental data for one configuration (see Progress Report, January, 1961, ANL-6307).

Since the total U^{235} content of each seamed tube is to be kept the same as the U^{235} content of the equivalent extruded tube, the power density within the fuel meat will of necessity increase a constant amount over that of the extruded tube for the same total reactor power. The factorial increase in power density within each fuel tube is given below for two different seam widths.

Seam Width	Inner Tube	Intermediate Tube	Outer Tube
0.79 cm	1.048	1.040	1.035
0.95 cm	1.058	1.049	1.042

In addition to this constant power density increase, the thermal neutron flux peaks at the seam-meat interface. The magnitude of this effect varies with the seam position, but in no case is the peaking factor greater than 1.10. Away from the interface, the magnitude of this effect decreases rapidly. The local power peak near a seam will not give rise to an appreciable clad surface temperature rise since the seam, acting as a thermal fin, provides a large effective heat transfer area at the position of maximum power.

A conservative heat transfer calculation has indicated that the temperature peak is located within the meat at a circumferential distance of 1.27 cm from the seam-meat interface. The magnitude of the temperature rise at this point, compared to an extruded tube at the same total power, is a maximum of 3.0° C for the 0.95 cm wide seam and 2.6° C for the 0.79 cm wide seam. In order to maintain the same maximum clad temperature when using seamed fuel tubes, the total reactor power would have to be maintained at 95% of the maximum power level allowable using extruded fuel tubes.

b. Mathematical Numerical Analysis - It has been found possible to extend the iterative technique for interpolation by ratios of functions to include osculatory interpolation, i.e., interpolating functions for which not only the functional values, but also the first m_i derivatives are specified at the set of points x_i. The method has been tested by hand computation, in fitting the function and its first derivative at a set of points, and several variants have been programmed for the LGP-30 computer. Tests of the programs in interpolating for tan x in the vicinity of $\pi/2$ have shown that a two-point osculatory rational interpolation is more effective than a three-point rational interpolation using only function values. However, the higher order interpolations do not increase in effectiveness, owing to the effects of roundoff errors. Further analysis is planned to determine whether this source of difficulty can be ameliorated. A preliminary report on the new algorithm has been submitted to the Society for Industrial and Applied Mathematics for presentation at their April meeting.

c. Resonance Integral and Age Compilation - A second draft of the resonance integral compilation was completed.

A preliminary listing of neutron age values was made. Reports concerning neutron age investigations using non-fission sources as well as fission sources are being examined for data to be included in the listing.

d. Reactor Physics Constants Center - Drafts of several sections for the Revised 1961 edition of Reactor Physics Constants (ANL-5800) are in final form and preparation of the remaining material is proceeding.

e. ZPR-VII Data Analysis - Calculations have been completed for flux-trap experiments using 25:1 THUD fuel in lattices of $2a_0$ or $6a_0$ spacing and D_2O moderator. Approximations are being developed to study the flux traps using 15:1 Babcock and Wilcox fuel and $1a_0$ spacing. These flux traps were constructed with very thin core regions, some being less than 3 cm thick. Actually the value of k_∞ of these cores is less than unity because of the strong resonance absorption in the thorium. However, it has been shown, using three-group theory with extrapolation distance boundary conditions, that such a reactor can be made critical provided that η_f is greater than a quantity which is a function of the migration areas in the moderator and is greater than unity. This means that most of the fast neutrons produced escape from the thin core and slow down in the moderator below resonance energies before being absorbed in the core.

B. Reactor Fuels Development

1. Ceramic Fuels

a. Refractories for Reactor Use - Work on the lanthana-urania system was continued. Air sintered compacts were subjected to the following heat treatment:

Temperature (°C)	Time (hrs)	Furnace Atmosphere
1675	310	Combustion
1450	236	Air
1800	4	Hydrogen

Weight loss, density, and microstructural changes resulting from the oxidizing heat treatments are under investigation. Most of the compacts disintegrated when subjected to the 1800°C hydrogen heat treatment. The failure was possibly caused by oxygen release of the solid solutions during the reducing heat treatment.

Investigation of the pycnometric densities of sinterable powders used for pellet fabrication was initiated. Chemical, metallographic and X-ray investigations of sintered compacts subjected to the previous heat treatments are continuing.

b. Urania-Thoria Bodies - Computer calculations were completed for X-ray parameter data of urania-thoria solid solutions prepared by the air sintering of U_3O_8-ThO_2 mixtures reported last month. Solid solutions in the range of 2 m/o to 70 m/o UO_2 were subjected to (1) air sintering, (2) oxidation following air sintering, (3) reduction following air sintering, and (4) oxidation of samples reduced to stoichiometry. Correlation of the X-ray data with stoichiometry is in progress. The lattice parameters calculated for air sintered samples reduced to stoichiometry in H_2 are in agreement with those reported by Lambertson, Mueller, and Gunzel, J. Am. Cer. Soc., 36, 397-399 (1953).

Experimental apparatus has been assembled for determination of the thermal expansion of these solid solutions to 1600° C. Calibration of the dilatometer is being made using 90° oriented single crystal sapphire as the reference material.

c. Uranium-Thorium Sulfide - Further data showing the effects of argon and vacuum processing of US 15 (see Progress Report, November, 1960, ANL-6269) are given in Table VIII.

Table VIII. Effects of Argon and Vacuum Processing of Uranium Sulfide

Treatment (1800° C-1 hr)	Chemical Analysis (w/o US phase)	Lattice Parameter, a (Å)	Average Grain Size (μ)	Modulus of Rupture (psi)
Argon Homogenization		5.4853		
Argon Homogenization, Argon fire	97.8	5.4815	121	24,100
Argon Homogenization, Vacuum fire	98.8	5.4882	78	15,800
Vacuum Homogenization		5.4878		
Vacuum Homogenization, Argon fire	97.8	5.4837	71	21,500
Vacuum Homogenization, Vacuum fire	98.5	5.4891	24	15,400

It appears that the atmosphere during firing, rather than homogenization, determines the purity of the final product as given by w/o US phase. Polished sections showed that the argon-fired material contained considerably more grain boundary and surface secondary phase than did the vacuum-fired material for almost identical purity of product. Firing in argon also caused an increase in grain size. In all cases argon firing lowered the lattice parameter of the material whereas vacuum firing raised it. Finally, it was found that the modulus of rupture correlates most closely with density.

Preliminary to irradiation testing of US, its compatability with jacketing materials and its resistance to NaK attack are being determined simultaneously. A US sandwich containing sheets of V, Nb, Zr, and stainless steel is being run in NaK at 800° C for two weeks. Pellets made from the same batch will undergo water and steam corrosion tests.

Other property measurements are being set up. The vacuum interferometer apparatus has been made ready for thermal expansion measurements to 1000° C. An apparatus to measure the thermal conductivity up to 1000° C in argon, or vacuum if necessary, is now being constructed.

d. Poisoned Glasses - The feasibility of coating Zircaloy with a porcelain enamel containing a high cross section element such as gadolinium was investigated. A coating of this type could, for example, be used as a means of introducing burnable poison into EBWR Core II. The coating would be applied to the inside surface of the Zircaloy shrouds which contain the fuel elements.

Two commercial enamel frits suitable for coating stainless steels were selected for initial trials. Batch compositions of each of these frits were ground into slips, omitting the Gd_2O_3. The enamel slips were sprayed on both surfaces of 25 mil thick Zircaloy specimens which had been vapor blasted with 200 M Al_2O_3. After drying, the coated specimens were fired in air from 3 to 5 minutes at 1030° C, and similarly at 860° C. These heat treatments resulted in a good quality coating with no obvious imperfections. On the basis of these results, a batch composition containing 5% by weight of Gd_2O_3 was ground into slip. This slip was applied to specimens as above and fired for 3 minutes at 880° C. The resultant coating was one mil in thickness and had an excellent appearance. This work is considered to have established the feasibility of applying a porcelain enamel coating to Zircaloy.

e. Properties of Plutonium Carbide - A program to characterize high purity and high density plutonium-carbon alloys near the monocarbide composition is being continued because of its interest as a potential fast reactor fuel.

The density of alloys prepared by arc melting and casting which range in composition from 42.1 to 59.6 a/o C have been measured using the displacement method in CCl_4. These values were above 98% of the theoretical density.

The phase boundary and the identity of the low temperature phase found in the 44.5 a/o C thermal expansion specimen was investigated by metallography and X-ray diffraction. Specimens from cast alloys containing 42.1, 45.4, 46.8 and 50.4 a/o C were heat treated at 400° C for 25 days and fast cooled. The phases present in these specimens are given in Table IX.

Table IX: Phases Present in Cast Specimens that Were Heat Treated at 400° C for 25 Days

Composition (a/o C)	Phases	
	As Cast	Heat Treated
42.1	PuC + αPu	PuC + Pu_3C_2(zeta)
45.4	PuC	PuC + Pu_3C_2(zeta)
46.8	PuC + Pu_2C_3	PuC + Pu_2C_3
50.4	PuC + Pu_2C_3	PuC + Pu_2C_3

These results tend to confirm the general configuration of the LASL phase diagram (1960 Grenoble Conference) in the region of the monocarbide composition. The PuC, PuC + Pu_3C_2 (zeta) phase boundary occurs at about 45 a/o C in the LASL phase diagram while this work shows that this phase boundary is between 45.4 and 46.8 a/o C. The dilation curves of the 44.5 a/o C specimen, discussed in previous Progress Reports, can now be fully explained. On heating, the PuC + Pu_3C_2 phases expand uniformly to 400° C. Above this temperature there is an abrupt increase in the rate of expansion with increase in temperature as the Pu_3C_2 (zeta) phase dissolves in PuC. At 560° C the solution of Pu_3C_2 is complete and the dilation curve has about the same slope as the curves of higher carbon PuC specimens above this temperature.

Metallographic examination of a 44.4 a/o C specimen furnace cooled at a rate of 2° C per minute from 1000° C revealed that PuC was retained to room temperature.

Alloys containing less than 46.8 a/o C have been observed to crack when thermal cycled in a dilatometer and can be expected to lead to breakup under reactor conditions. This can be attributed to cyclic stresses resulting from the large expansion and contraction associated with the solution and precipitation of Pu_3C_2 on thermal cycling.

2. Properties of Metals and Alloys

a. Thermal Properties of U-Fs - Under subcontract to the Laboratory, the University of Denver is determining the specific heats, heats of transformation, and heats of fusion of U-3, 5, and 8 w/o Fs alloys. Data have been completed during the past month on the 5% fissium alloys in this series except for determination of the heat of fusion. Difficulty has been encountered with the drop calorimeter because, upon dropping, the weight of the sample has sheared the bottom from the yttrium capsules and a change in design has been necessitated. Only one sample of the 5% alloy is available and a cautious approach is necessary so that data can be obtained for this material. Data have been taken to 900° C on both 3% fissium alloys and one of the 8% samples. One of the 3% samples was cut into smaller size pieces

to be accommodated in the diphenyl ether calorimeter and it is planned to run a duplicate on this material through the first transformation. A similar series may also be possible on the 8% alloy.

The data obtained seem self-consistent and indicate a small decrease in the heats of transformation with increasing fissium content.

3. Nondestructive Testing Techniques

a. Ultrasonic Techniques - As mentioned in last month's Progress Report, velocity measurements have been made on a fused silica sample which is being sent around the country for velocity correlation and comparison purposes.

Based on a measurement of 2.01065 cm for the sample thickness, the velocity through the sample was found to be (5.966 $\pm$ 0.002) cm/sec. This would give a transit time of 6.740 microseconds. The formula used to calculate the velocity was:

$$V = \frac{2Lf_0}{n_0\,(\phi/2\pi)}$$

where L is the sample thickness, f_0 the measuring frequency, n_0 is the number of wavelengths in the sample at f_0, and ϕ is the phase shift in the wave upon reflection at the boundary between the sample and a buffer rod.

b. Neutron Techniques - The photographic detection of neutron images is presently being investigated with at least one long range objective being to apply neutron radiography as a useful inspection method for nondestructive testing.

Further study has now been made concerning the use of a Van de Graaff generator as a neutron source for radiographic purposes. The Li^7 (p,n) $\longrightarrow$ Be^7 reaction was employed to produce a fast neutron source in a thick lithium target. The accelerating voltage was 2.5 Mev and the proton current was 10 microamperes. At a distance of 12 inches from the target the neutron flux of the unmoderated beam had a photographic effect approximately 10% of that of the monochromatic thermal neutron beam being used at CP-5. Comparative speed measurements on several converter materials obtained using this beam do not follow the speed trends found with the thermal neutron beam at CP-5. These differences were observed because most of the photographic effect of the Van de Graaff unmoderated neutron beam appears to result from resonance neutrons. This was confirmed by relative speed measurements taken with and without a cadmium filter in the beam. Very little difference was observed between these two sets of measurements, in spite of the fact that the cadmium filter essentially cuts out of the beam all neutrons of energy less than about 0.4 ev.

Of the metal converter screens used, the resonance neutrons yielded relatively high photographic speeds particularly with silver, rhodium and gold (these materials all show several absorption peaks in the resonance region).

C. Reactor Materials Research and Development

1. Irradiation Damage in Steels

a. Transmutations Under High Irradiation - A general mathematical solution of the transformation of the isotopes of iron during a long exposure to thermal neutrons was started. For an SA-212B composition exposed to 1.5×10^{22} nvt thermal neutrons, the calculations show that about 3% of the Fe^{56} isotope is transformed to the Fe^{57}. A similar percentage of Fe^{57} is converted to Fe^{58}. The conversion process reduces the relative amount of the Fe^{56} isotope and builds up the Fe^{58} concentration. The process slows down at the Fe^{58} point because of the low absorption cross-section of Fe^{58}.

b. Magnetic Properties of Irradiated Steels - An improved bridge circuit utilizing the best of several experimental search coils (1000 turns of No. 28 gage enameled wire wound on a 2.5 cm long core with a 0.76 cm diameter) was built. The new bridge circuit utilizes 100 ohm resistances in the arms instead of the 1000 ohm components of the preliminary bridge. Tests with previously prepared samples of annealed and fully hardened SA-212B test bars (described in the December, 1960, Progress Report, ANL-6295, page 51) showed that the bridge circuit exceeded expectations. The unbalance voltage between the two arms of the bridge was reduced to 0.5 millivolt with an input of about 1.5 volts and 100 cps.

D. Reactor Components Development

1. Development of Manipulators for Handling Radioactive Materials

a. Force-reflecting Electric Servo Development - Work has continued on the problem of reducing the viscous drag in a force-reflecting electric servo system in which the master and slave motors are driven by separate amplifiers (see January, 1961, Progress Report, ANL-6307, p. 55). A one motion system has been successfully built and tested in bread-board form. The desired reduction in viscous drag was achieved by introducing the proper amount of series capacitance. The ratio of the reflected force to output force is reasonably constant over the useful speed range. The system is being investigated for possible use in master-slave manipulators that have dissimilar master and slave arms.

b. Force-reflecting Electro-Hydraulic Servo Development - Studies have continued on the development of an electro-hydraulic servo valve for use in a force-reflecting servo for a master-slave manipulator having a load capacity of 100 pounds and a maximum velocity of 36 inches per second.

An hydraulically powered system will have less inertia than an electric motor system, and possibly other advantages, such as elimination of reduction gearing, and lower powered electronics which makes the system more feasible for radio-control.

The ANL Model 12 electro-hydraulic servo valve was first operated in August, 1960, in a force-reflecting system using dual valves with integral electro-mechanical transducers and dual actuators. Since that time, extensive performance tests have led to modifications which have improved the stiffness and transient response of the system. Analytic work has paralleled the tests and has provided an understanding of the operation of the valve which will aid in the development of valves of improved performance.

The valve operates from a constant-flow supply and provides to an actuator a differential pressure which is proportional to the electric current in the electro-mechanical transducer. The valve has two stages, both of which operate on the open-center poppet principle. The full-signal differential output pressure of 1,000 psi is obtained with an electric input to the transducers of 0.2 watt. The valve has performed satisfactorily even when the hydraulic oil was considerably contaminated. The valve has a force-reflection sensitivity between one and two percent.

The reflection of transient forces have been improved by the addition of networks in the master and slave amplifiers which increases the gain of signals above 2.5 cps.

E. Heat Engineering

1. Void and Velocity Distributions in Two-Phase Systems

The mercury loop (see Progress Report, January, 1961, ANL-6307) was made to operate properly and the equipment necessary to determine values of the stream kinetic energy were assembled. One major change made in the loop was to replace the air system by nitrogen. This was done to stop the oxidation of the mercury and trace elements present.

To measure the stream kinetic energy a pitot tube is used with a static tap at the pipe wall. The pressure differential is measured by a pressure transducer. The transducer signal is integrated for a period of 60 sec and an average signal calculated. The integrating process is necessary since the kinetic energy at a point in the stream fluctuates between a large value for the mercury and a very small value for the gas.

2. Steam Separation

The problem of separating moisture from the steam leaving a reactor becomes increasingly important as the power output of reactors is increased. When the decision was made to increase the power of the EBWR to 100 Mw a program was initiated to evaluate the degree of natural separation (gravity forces only) that could be expected.

The General Electric Atomic Power Equipment Department in San Jose, California had constructed a large circulating loop and pressure vessel, capable of circulating up to 363,000 kg/hr (800,000 lb/hr) of water with steam flow rates of up to 30,000 kg/hr (65,000 lb/hr) at pressures up to 70 atm (1000 psia). A subcontract was drawn up with General Electric for a series of steam separation tests at 42 atm (600 psia). The tests were completed in March, 1960, and the results indicated that it would be difficult to obtain steam by natural separation with less than 5% moisture while operating at 100 Mw.

In a subsequent report by General Electric* it was reported that foaming was definitely occurring in the free surface during their tests. If foaming had been occurring, the 42 atm data obtained from this system by the Laboratory would be extremely questionable.

Analysis of their published data indicated that the existence of a foam layer could not definitely be established, nor could it definitely be eliminated. This conclusion stems from the fact that the effects of adding antifoam agents could not be assessed accurately owing to the non-reproducibility of the data. If, for the sake of argument, the non-reproducibility is ignored, the data show the addition of antifoaming agents had no measurable effect upon the "moisture gradient" through the interface of the two-phase mixture. In other words, the thickness of the measurable disengaging zone was not altered by the addition of antifoam agents.

It will be necessary to take more positive measures in order to establish the existence or non-existence of foaming in this experimental facility.

---

* "Free-Surface Separation of Steam and Water for Application in a Marine Reactor at 1000 psig," GEAP-3489, July 13, 1960.

F. Separations Processes

1. Fluidization and Fluoride Volatility Separation Processes

The volatilities of uranium and plutonium hexafluorides are the basis of schemes being considered for reprocessing power reactor fuels. Emphasis is now on the development of the Direct Fluorination Volatility Process applicable to the processing of typical Zircaloy-clad oxide fuels.

The direct fluorination of high-density, sintered uranium dioxide pellets in an inert bed of fluidized solids is being investigated to study problems of heat removal, reaction rate, fluorine utilization efficiency, and general behavior of the system.

Reactions of stainless steel with various gaseous reagents are being studied in fluidized beds with application to decladding of stainless steel-clad and processing of stainless steel-matrix fuels.

The fluid-bed conversion of uranium hexafluoride into uranium dioxide is being studied. The process involves either a single-step pyrohydrolysis and reduction using steam and hydrogen simulteneously, or a two-step process using alternatively steam or hydrogen, followed by the other reactant.

Studies are being made on the fluorination, transportation, separation, and decomposition of plutonium hexafluoride on a multigram scale.

a. Direct Fluorination of Uranium Dioxide Fuel - The study of the direct fluorination process applied to uranium dioxide fuel is directed toward demonstration of process control for the fluorination of typical pellet fuel in a three-inch diameter air-cooled reactor. Achievement of maximum uranium hexafluoride production rates is currently being sought. In order to provide adequate temperature control at high fluorination rates, an automatic fluorine control system was installed and tested during two shakedown runs. One control valve provides a constant pressure from a fluorine storage tank and another valve throttles the fluorine to the reactor in response to the reactor temperature. Quick response and ability to avoid temperature excursions have been shown. In one of the shakedown runs a charge of inert-fired pellets containing initially a relatively large amount of uranium fines (primarily UO_2F_2) produced caking early in the fluorination. This effect is attributed to intermediate fluoride formation and was not observed in the other run using a normal startup procedure (initially no fines present).

Fluorination of a three-inch deep bed of hydrogen-fired pellets was carried out in a two-inch reactor at 500°C with an inert bed of pure zirconium fluoride to determine whether zirconium fluoride could serve as a satisfactory fluid bed. Zirconium fluoride fluid-bed fluorination with

inlet fluorine concentrations of 20 m/o and 40 m/o for 1-hour periods not only showed no unusual reactions but gave rates and efficiencies which corresponded closely with those obtained under similar conditions with other inert bed materials (calcium fluoride and magnesium fluoride). It was concluded that no operational difficulities appeared in the use of a pure zirconium fluoride inert bed.

Heat transfer tests have continued in a mockup system to determine effective radial thermal conductivity of packed-fluid beds similar to those encountered in pellet fluorination. With $\frac{3}{8}$-inch brass pellets and with glass beads (20 percent 120 mesh, 70 percent 140 mesh, 10 percent 170 mesh), thermal conductivities and wall film coefficients were obtained as a function of fluidizing gas velocity. An average thermal conductivity of the bed was about 10 Btu/(hr)(sq ft)(°F/ft) with fluidization in the voids of the pellet bed. Without fluidization the conductivity was less than one Btu/(hr)(sq ft)(°F/ft). The range of individual values of conductivity varies about ±40 percent. The incipient fluidization in the system occurred at a superficial velocity (based on the tube diameter) of 0.2 ft/sec; measurements were made at gas rates from 0.088 to 1.0 ft/sec.

b. Processing Stainless Steel-Clad Fuel Elements - A two-zone fluid-bed reactor is being used to study chlorination reactions for the decladding and dissolution of stainless steel-clad uranium dioxide fuel elements. The chlorination products formed in the lower zone pass into the upper zone where they are converted to the tetrafluoride by reaction with hydrogen fluoride. Fluid-bed experiments were recently completed on 4-inch long by $\frac{3}{8}$-inch OD (35 mil thick) closed end stainless steel type 304 specimens. Overall penetration rates of 4.5 mils/hr were achieved in 4.5-hr chlorination tests at 580°C. This may be compared to the 9 mil/hr penetration rate reported previously on short, open sections of tubing. Apparently reaction at the open tube ends was a large contributing factor to tightly adhering nickel chloride and chromium chloride films which limited the chlorination reaction rate. A pyrohydrolysis step is being investigated as a means of converting the films to a brittle scale. A pyrohydrolysis of a chlorinated specimen carried out in a tube furnace for 10 to 15 minutes with steam at 600°C, followed by light brushing of the specimen, permitted the initial chlorination rate of 9 mils/hr to be attained. An alternate test mixture (three parts water, one part chlorine and two parts nitrogen) gave essentially no reaction (0.06 mil/hr) at 600°C.

A 20-pellet (121 g) charge of uranium dioxide was 92 percent reacted by a 69 mole percent chlorine in carbon tetrachloride mixture at 550°C in 2.5 hours in the 1.5 inch diameter fluid-bed reactor. No excessive heat effects were noted.

A tube-furnace study on the reaction of chlorine-carbon tetra chloride mixtures with uranium dioxide indicates that the reaction rate is

dependent on the ratio of gases. A maximum rate of 600 mg/(sq cm)(hr) was achieved near 50 mole percent chlorine at a linear velocity of 0.23 ft/sec and a temperature of 550°C.

c. Conversion of Uranium Hexafluoride to Uranium Dioxide by a Two-Step Fluid-Bed Process - The first step in this process is the steam hydrolysis of UF_6 to UO_2F_2. A three-inch diameter Monel fluid-bed reactor is being used to study the reaction of uranium hexafluoride with steam to form uranyl fluoride. The continuous operation period was extended from four hours to seven hours when the unit was operated at a feed rate of 100 g/min of uranium hexafluoride at 200°C with a recycle seed particle rate of 10 to 15 percent. However, the formation of fines again caused shutdown.

The second step involves the reduction of UO_2F_2 to UO_2. A study of the effect of the hydrogen to steam ratio on the reduction of uranyl fluoride to uranium dioxide is in progress in a three-inch diameter Monel fluid-bed reactor. A three-kilogram charge was converted to specification-grade (<300 ppm residual fluoride) uranium dioxide in five hours at 650°C using a gas composition of two parts hydrogen to one part steam.

d. Fundamental Fluidization Studies - The study of multistage fluidization has been completed and published as ANL-6267. One of the advantages of fluid-bed operation is high mass transfer rates. Accordingly a mass-transfer study was made to investigate the effect of column parameters on mass transfer rates in a fluid-bed system. A silica gel-water system was chosen as a convenient method of evaluating column variables such as gel particle size, fluidizing velocity and flow rate of solids. Murphree vapor efficiencies of the order of 100 percent were obtained for all cases except at the highest superficial velocity tested, 0.98 ft/sec, where the efficiency dropped to 96.5 percent. High efficiencies were attributed to the small particle sizes of the gel and the excellent contact of gas and solids in the movement of particles from stage to stage ("raining solids" phenomenon). A modified mass transfer coefficient of 35.4 lb of water/(sec)(cu ft of bed)(lb of water/cu ft of air) was determined.

e. Plutonium Fluoride Studies - Further work on the plutonium hexafluoride decomposition kinetics at 140° and 170°C has substantiated the conclusions, made from the previously reported results obtained at 160°C, that the rate equation can be expressed as a concurrent zero order and first order reaction ($-dp/dt = k_0 + k_1 p$).

Experiments are being performed to determine to what extent plutonium can be removed by fluorination from solid substrates representative of those which would be obtained after the fluid-bed decladding step of the Direct Fluorination Volatility Processing of uranium oxide fuels. Of the preliminary experiments, the best removal has been obtained upon reacting

a plutonium oxide-uranium oxide mixture with fluorine in a bed of calcium and zirconium fluorides. The uranium was removed at 400°C in two hours, and after eight hours of subsequent fluorination at 500°C, the plutonium concentration in the residue was 0.02 weight percent.

Additional experiments have been carried out on the rate of thermal decomposition of plutonium hexafluoride in a flowing gas stream. The purpose of these experiments is to study further the factors affecting the decomposition of plutonium hexafluoride under conditions simulating those in the effluent stream of a fluorination reactor. The results indicate that the rate of decomposition of plutonium hexafluoride increases as plutonium tetrafluoride is deposited in the vessel. The rates of decomposition at 150° and 200°C in a vessel packed with nickel wool, having a surface to volume ratio of 19 times greater than that of a similar nonpacked vessel, were 10 to 50 times greater than the rates obtained in the nonpacked vessel. These rates are attributed to the combined effects of the deposition of plutonium tetrafluoride and the increased surface area of the packed vessel.

Plutonium tetrafluoride which had previously been thermally decomposed in a vessel packed with nickel wool was fluorinated at temperatures ranging from 220° - 450°C. The fluorinations resulted in a recovery of 98 percent of the plutonium as plutonium hexafluoride.

f. Calcination in Small Diameter Columns - Work has started on a scheme for reducing the amount of off-gas that needs be handled in a fluid-bed calciner. Plans call for the introduction of the atomized feed at the bottom of the fluid-bed; thus the atomizing air plus the feed decomposition gases provide all the necessary gas for fluidization. A $2\frac{1}{2}$-inch Vycor column operated smoothly in preliminary runs.

2. Chemical-Metallurgical Process Studies

Continued progress is being made in the development of new pyrometallurgical methods for reprocessing nuclear reactor fuels. Supporting studies are providing physicochemical data on liquid metal systems and thermochemical data on materials of interest to high-temperature chemistry.

a. Liquid Metal Solvent Studies - The solubility of the rare earths thulium and lutetium in liquid cadmium may be represented by the empirical equations

thulium (324° - 524°C): $\log (\text{atom percent}) = 3.488 - 2514\ T^{-1}$.

lutetium (340° - 528°C) $\log (\text{atom percent}) = 3.781 - 2632\ T^{-1}$.

The density of the delta phase (TiCd) in the titanium-cadmium system was found to be 7.1 g/cc. This value compares well with the theoretical value of 7.05 g/cc computed from the structure of the substance as determined by X-ray diffraction methods.

The solubility of uranium in liquid zinc has been measured above 800°C. The results may be represented by the empirical equation

$$\text{uranium (840° - 900°C): } \log (\text{weight percent}) = 5.87 - 5550\ T^{-1}.$$

This equation represents the solubility of the delta uranium-zinc phase (U_2Zn_{17}).

The distribution coefficient of strontium between the immiscible liquids lead and zinc has been measured. Values of the coefficient (weight percent strontium in zinc/weight percent strontium in lead) are 0.12 at 701°C and 0.15 at 734°C.

b. Calorimetry - Using fluorine bomb calorimetry, a preliminary value of -507 ± 2 kcal/mole has been obtained for the standard heat of formation of uranium hexafluoride, UF_6 (g); this value compares well with the literature values. However, it is expected that after refinements in the calculation are made, the value obtained in the present study will be of considerably greater accuracy.

Initial calorimetric experiments have been completed with boron nitride using the fluorine vessel and bomb for spontaneously reacting substances. Exploratory combustions of vanadium, tantalum, and niobium in fluorine have been started.

The 1500°C furnace for the high-temperature enthalpy calorimeter has been assembled and the high vacuum system for the furnace has been installed.

G. Advanced Reactor Concepts

1. Basic Radiation Effects Reactor

The objective of the BRER effort is the preparation of a conceptual design for a fast reactor which gives a maximum ratio of fast neutrons to gammas and thermal neutrons. The reactor is to be used for certain basic radiation damage studies and will feature experiments at temperatures ranging from 4°K to 300°C - 400°C. Fast fluxes ranging from 2×10^{13} to 2×10^{11} are a requirement as a various neutron energy spectra ranging from a compact distribution at the highest possible energies to a distribution peaked in 10 - 50 kev.

A preliminary design report has been completed, reviewed, and prepared for publication. The only significant change made to the design is the use of NaK in the secondary heat exchange loop rather than the isopropyldiphenyl previously specified. This change was made to eliminate any chance for leakage of moderating material into the fast core by heat exchanger failure

2. Biogeonuclear Reactor

A literature search has revealed that neutron activation techniques for the analysis of plant, soil, rock, and water samples can be used to a limited extent Thermal neutron fluxes of the order of 10^6 - $10^{\sim}$ $n/cm^2/sec$ could be useful for activation analysis of rare earth elements, but in general values as high as 10^{12} $n/cm^2/sec$ are needed. Geological and associated biochemical methods could benefit from three types of neutron sources.

(a) Plutonium-beryllium source with a thermal region for activation and sample screening.

(b) Portable reactor of modest cost for on-the-spot determination or detection.

(c) Stationary reactor with hot cells for activation analysis requiring high sensitivity such as determining trace constituents.

A truck-trailer mounted reactor (Item b) appears attractive however it would require more development than the other two schemes.

3. Fast Reactor Test Facility

A FARET facility would represent an important link in the fast reactor development as a proving ground for the testing of future fast reactor systems or individual components. Currently, the integrated plant complex with a common heat rejection system and control room appears most attractive because of the variety of coolants under consideration (sodium, boiling sodium, sulphur, lithium, lead, bismuth, rubidium, and mercury). Also a common facility would be less expensive than individually contained reactor sites for each experiment. More uniquely, the complex would permit experiments at relatively high power as contrasted with the usual critical experiment.

Preliminary studies have been made in the advanced fast reactor field to establish ground rules for the work The results are being tabulated for the development of the design parameters.

4. Compact High Power Density Fast Reactors

The purpose of this work is to develop an ultra compact high-power density fast reactor using sodium vapor as a working fluid in a direct cycle system.

Experimental programs are needed:

(a) To find a structural material to function in 2000°F sodium vapor.

(b) To investigate the thermodynamic properties of sodium vapor.

(c) To develop and test a turbine to function in the high temperature sodium environment, to develop a practical method of cooling the turbine blades, and to develop a method to measure the temperature drop across the blades to verify the practicability of blade cooling design.

(d) To develop associated elements of the system, i.e., shaft seals, bearings, and instrumentation.

A tentative test loop was designed to provide a method for studying all these items in one rig; however, discussions indicate that no one test rig will satisfy all needs. Therefore it is proposed to set up a program in three phases. The first phase is to test available materials in the existing sodium corrosion loops. Twelve materials have been received for test. Among them are several alloys of Hastalloy and Stellite, a sample of rhenium, Vascoly-Ramet's Tantung alloy, tantalum, tungsten, and three samples of Kennametal Company's Kentanium TIC alloys. In addition, several columbium, tungsten, and molybdenum alloys are on order. Paralleling this corrosion work will be a simple natural circulation loop for heat transfer work on high temperature liquid sodium. The second phase of the test program will be similar except that sodium vapor at 2000°F will be the working fluid instead of liquid sodium, using materials proven in the first test.

After selection of suitable materials, the final phase of the test program would be the development and testing of components such as a high temperature sodium turbine.

5. AHFR Hydraulic Test Loop

Testing of a 30-mil fuel element mockup has been partially completed, and a point has been recorded where transverse bowing of the inlet edge of the fuel plates begins. The average water velocity in the coolant channels at this point was 5.74 m/sec and compares well with the calculated critical velocity of 5.2 m/sec. The total flow rate through the element when distortion became apparent was 640 liters/min. When this point was recorded, water temperature was 82°C, inlet pressure 9.5 atm, and pressure drop across the assembly 1.89 atm. The distortion consisted of adjacent plates bowing parabolically together, reducing the channel width by approximately one-half at the vertices. This condition existed at several random locations along the inlet end of the element. No longitudinal buckling or vibration was detectable, and transverse bowing ceased as soon as flow was shut down. Removal and inspection of the element revealed no permanent distortion or other damage.

The element tested was similar in construction to the 50-mil channel previously reported; however, it featured rounded "buttons" on all four mounting surfaces which maintained a constant 30-mil coolant passage between the element and its holder. Also, the keys which support the individual fuel plates and provide accurate plate spacing are rounded to provide better flow streamlining in each channel. As in previous mockup tests the fuel assembly was constructed of 1100-H14 aluminum alloy and consisted of 37 fuel plates, each 32 mils thick and spaced at 32 mils to form 36 coolant channels. Total length of the element is 48.6 cm.

A support comb is being fitted across the inlet of the element at the points where bowing is most severe. With the inlet ledges thus stiffened, further tests will be conducted to determine whether longitudinal buckling can still occur Final tests are also contemplated at water temperatures up to 145°C in order to establish the effect or relevance of cavitation.

6 Packed Bed Reactor Studies

A review was made of the properties of possible fuels for a packed bed irradiation test. On the basis of attainable fuel (U^{235}) density and corrosion resistance thoria-urania appears preferable to urania or an urania-zirconia-calcia composition. A literature survey was also made to obtain thermodynamic and transport properties of steam at low pressure and high temperatures (to 800°C). Using these values and friction data previously obtained a parametric study of packed bed characteristics was made over the range of interest for neutron flux, fuel density, and coolant (steam) temperature

In a further effort to establish the bed geometry the decay heat cooling problem was investigated. Although data on thermal conductivity and emissivity (for thermal radiation) for the exact bed materials contemplated do not appear to exist, some conservative estimates indicate that it will be feasible to consider decay cooling by radiation alone. The bed will have to be arranged to "see" a reasonably cool surface, however.

An investigation of induction heating of iron particles for a heat transfer study on packed beds was made. Results of heating a sample indicate a heat flux one-twentieth that desired. To complete the feasibility study, a new coil and transmission line are being fabricated for optimum coupling, and a simple apparatus to sort the particles mechanically to a more uniform particle size is being constructed. The information necessary for specifying the feasibility of high-density heat flux will be obtained by the new coil configuration and by submitting samples to induction heating equipment manufacturers for their evaluation.

7. Direct Conversion

Experimental and theoretical studies on the cesium diode have continued. The effective constants "A" and "ϕ" in Richardson's equation $I = A T^2 e^{-\Sigma\phi/KT}$ have been determined in the conventional way by varying the emitter temperature from 1400°C to 2000°C in vacuum. Subsequently the cesium containers were broken and the cell was filled with Cs vapor. The Cs vapor pressure was regulated by the temperature of the oil surrounding the Cs trap. The emitter current was plotted for emitter temperatures from about 1400°C to 2000°C. In each plot the applied voltages were varied from -4 volt to +8 volt between collector and emitter. The results are now being processed to determine the maximum power output for each temperature.

The single sheath model is being used for theoretical investigation of the current-voltage characteristics of the cesium diode. For a uniform plasma with a large ion mass the characteristic obtained reduces to that of Lewis and Reitz.* The electron temperature and plasma density become functions of the diode current.

It is very desirable to convert the low voltage-high current a-c, as delivered by the plasma cell, into a high voltage a-c output. A transistorized converter circuit has been designed and tested. The circuit consists of two 2N457 transistors in a push-pull arrangement with a transformer to produce the switching voltage, and two 2N278 transistors biased by the switching voltage to permit the flow of the cell current in opposite directions at each half cycle. A second transformer winding increases the voltage by a factor of 70 or 140 and delivers essentially a square wave (a-c) of about 500 - 700 cycles/sec. The converter efficieincy is about 70% at 1.5-2 volt d-c input.

---

*Lewis and Reitz, "Efficiency of the Plasma Thermocouple," J. Appl. Phys. 31, 723 (1960).

VI. PUBLICATIONS

Papers

EXPANSION LOSSES IN TWO-PHASE FLOW

P. A. Lottes

Nucl. Sci. and Engr. 9, 26 (1961)

ATMOSPHERE HANDLING AND CONTROL FOR A GLOVEBOX FACILITY

L. F. Coleman, J. H. Schraidt, R. L. Breyne, and J. V. Natale

Proceedings of the Eighth Conference on Hot Laboratories and Equipment, San Francisco, California, USAEC Report TID-7599, Book 2, (1960) p. 494.

A RADIATION STABLE HEAVY-DUTY ELECTROMECHANICAL MANIPULATOR

J. E. A. Graae, D. C. Hampson, I. Pollack, M. Levenson, J. H. Schraidt, and G. J. Bernstein

Proceedings of the Eighth Conference on Hot Laboratories and Equipment, San Francisco, California, USAEC Report TID-7599, Book 1, (1960) p. 239.

LOW COST GLOVEBOXES

R. F. Malecha, H. O. Smith, J. H. Schraidt, J. V. Natale, N. E. Ross, and H. O. Brown, Jr.

Proceedings of Eighth Conference on Hot Laboratories and Equipment, San Francisco, California, USAEC Report TID-7599, Book 2, (1960) pp. 485-493.

PRODUCTION OF REFINED UF_6 FROM ORE CONCENTRATES BY FLUIDIZATION AND FRACTIONAL DISTILLATION TECHNIQUES

S. Lawroski, A. A. Jonke, N. Levitz, E. J. Petkus, A. H. F. Litty, W. A. Rodger, G. J. Vogel, R. K. Steunenberg, O. Sandus, W. J. Mecham, R. C. Liimatainen, R. W. Kessie, L. Trevorrow and R. C. Vogel

Progress in Nuclear Energy, Series III, Vol. 3, Process Chemistry, New York: Pergamon Press, 1961, pp. 98-125

DEVELOPMENTS IN MELT REFINING OF REACTOR FUELS

L. Burris, N. R. Chellew, S. Lawroski, G. A. Bennett, A. A. Chilenskas, M. Ader, H. M. Feder, J. B. Knighton, I. O. Winsch J. Wolkoff and W. A. Rodger

Progress in Nuclear Energy, Series III, Vol. 3, Process Chemistry, New York: Pergamon Press, 1961, pp. 319-328.

Papers

DEVELOPMENTS IN PYROMETALLURGICAL PROCESSING
J. H. Schraidt and M. Levenson
Progress in Nuclear Energy, Series III, Vol. 3, Process Chemistry, New York: Pergamon Press, 1961, pp. 329-339.

PURIFICATION OF REACTOR FUELS AND BLANKETS BY CRYSTALLIZATION FROM LIQUID METAL SOLVENTS
H. M. Feder and R. J. Teitel
Progress in Nuclear Energy, Series III, Vol. 3, Process Chemistry, New York: Pergamon Press, 1961, pp. 355-390.

FLUIDIZED-BED CONVERSION OF FUEL PROCESSING WASTES TO SOLIDS FOR DISPOSAL
J. W. Loeding, A. A. Jonke, W. A. Rodger, R. P. Larsen, S. Lawroski, E. S. Grimmett, J. I. Stevens and C. E. Stevenson
Progress in Nuclear Energy, Series III, Vol. 3 Process Chemistry, New York: Pergamon Press, 1961, pp. 393-413.

ANL Reports

ANL-6212 SOME PHYSICS CALCULATIONS ON THE PERFORMANCE OF LARGE FAST BREEDER POWER REACTORS
S. Yiftah and D. Okrent

ANL-6223 PURIFICATION OF REACTOR FUELS USING LIQUID ZINC
J. B. Knighton, L. Burris, Jr. and H. M. Feder.

ANL-6248 TERMINAL REPORT ON THE BOILING SLURRY REACTOR EXPERIMENT
G. A. Freund, J. D. Lokay, G. C. Milak, and J. C. MacAlpine

ANL-6256 THE EFFECT OF THE LIQUID VISCOSITY IN TWO-PHASE, TWO-COMPONENT FLOW
Melvin Jerome Fohrman (Thesis Submitted to the Faculty of Illinois Institute of Technology in Partial Fulfillment of the Requirements for the Degree of Master of Science in Mechanical Engineering)

ANL-6261 A LUNAR POWER PLANT
J. C. Carter, H. H. Hummel, M. J. Janicke, and J. F. Marchaterre