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Argonne National Laboratory

REACTOR DEVELOPMENT PROGRAM

PROGRESS REPORT

April 1965

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Reactor Development Program
Highlights of Project Activities for April, 1965

EBR-II

A continuous power run of 12 days duration at 45 MWt and 14 MWe was completed as scheduled. During subsequent shutdown, difficulty was experienced with gripper actuation of two control rod drives which required replacement of the affected grippers. At the end of the month, the reactor was again in operation at 45 MW and half way through its next scheduled run which is expected to carry the maximum fuel burnup to 0.64 a/o. Preliminary inspection of fuel elements removed from the reactor after 0.5 a/o burnup maximum indicated them to be in excellent condition.

FARET

The A-E Title II design of the FARET Facility is estimated to be approximately 97% complete. All Title II design packages have been completed for transmittal to the AEC except Packages V, Instrumentation and Controls, and IX, General Facility, which are expected to be transmitted in late May or early June.

The Babcock and Wilcox Company has been selected to design and fabricate the reactor vessel. Authorization to proceed with design has been received.

The due date for vendor proposals for the cell shielding windows has been extended to May 1st.

AARR

A detailed scope of work for Title I, II and III effort was sent to the selected A-E firms (Burns and Roe, Inc.). Contract negotiations are in progress.

Two of the six samples of prototype fuel material being irradiated in the ETR G-12 facility were removed after completing one cycle of irradiation, with a maximum burnup of about 70% of the reference core life in AARR. Some discoloration was noted, but measurements have established that no plate blistering or gross dimensional distortion occurred.

Critical experiments with the 2/3 loading (810 fuel foils) are in progress.

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REACTOR DEVELOPMENT PROGRAM
PROGRESS REPORT

April 1965

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FOREWORD

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

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

The last six reports issued
in this series are:

October 1964	ANL-6965
November 1964	ANL-6977
December 1964	ANL-6997
January 1965	ANL-7003
February 1965	ANL-7017
March 1965	ANL-7028

TABLE OF CONTENTS

	<u>Page</u>
I. Liquid-Metal-cooled Reactors	1
A. EBR-II	1
1. Operations	1
2. Systems and Components	2
3. Plant Modification and Maintenance	4
4. Fuel Cycle Facility	5
B. FARET	8
1. General	8
2. Control Drives	9
3. Core Instrumentation	9
4. Fuel Assembly Sodium Flow Test Loop	10
5. Static Test Vessels	10
6. Sodium-Air Reaction	11
7. FARET Criticals	14
C. General Fast Reactor Physics	16
1. ZPR-3	16
2. ZPR-6	17
3. ZPR-9	17
4. ZPPR	19
D. General Fast Reactor Fuel Development	20
1. Development of Jacket Materials	20
E. General Fast Reactor Fuel Reprocessing Development	21
1. Skull Reclamation Process	21
2. Blanket Process	22
3. Materials and Equipment Evaluation	23
II. General Reactor Technology	24
A. Experimental Reactor and Nuclear Physics	24
1. Determination of U^{238} Capture Rate	24

TABLE OF CONTENTS

	<u>Page</u>
B. Theoretical Reactor Physics	25
1. Inelastic Scattering from Iron, Nickel, and Chromium	25
2. Diffusion Theory Calculation	27
C. High-temperature Materials	27
1. Ceramics	27
2. Liquid-Metal Corrosion	30
D. Other Reactor Fuels and Materials Development	31
1. Nondestructive Testing	31
E. Engineering Development	35
1. Two-phase Flow	35
2. Boiling Liquid Metal Technology	35
3. General Heat Transfer	37
4. Electric Master-Slave Manipulator Mark E4	39
5. Viewing Systems	39
F. Chemical Separations	40
1. Fluidization and Volatility Processes	40
2. Chemical-Metallurgical Process Studies	43
G. Plutonium Recycle Reactors	45
1. EBWR Facility	45
2. Fuel Development and Fabrication	47
III. Advanced Systems Research and Development	49
A. Argonne Advanced Research Reactor (AARR)	49
1. Critical Experiment	49
2. Heating of Materials in the Internal Thermal Column	52
3. Heat Transfer	53
4. Reactor Pressure Vessel	55
B. Magnetohydrodynamics (MHD)	56
1. Rotating Power-generating System	56

TABLE OF CONTENTS

	<u>Page</u>
C. Regenerative Emf. Cells	58
1. Bimetallic Cells	58
IV. Nuclear Safety	60
A. Reactor Kinetics	60
1. Fast Reactor Safety	60
B. TREAT	64
1. Operations	64
2. Large TREAT Loop	66
C. Chemical and Associated Energy-transfer Problems in Reactor Safety	67
1. Metal-Water Reactions	67
V. Publications	70

I. LIQUID-METAL-COOLED REACTORS

A. EBR-II

1. Operations

At the beginning of the month, the plant was in operation at a reactor power level of 45 MWt, with the turbine-generator producing about 14 MW of electrical power. Plant operation continued until a maximum reactor fuel burnup of approximately 0.5 a/o was attained on April 11.

The reactor power level was then decreased in steps while power coefficient measurements were made. The reactor was then shut down, and the plant put in standby condition preparatory to removal of fuel-surveillance subassemblies.

Restricted fuel handling was performed to load one core-type subassembly and one inner blanket-type enriched subassembly into the storage basket. At this time, two stainless steel, core-type dummy subassemblies were removed from the storage basket.

During the operations preparatory to unrestricted fuel handling, it was found that control rod drive No. 8 could not be disconnected from the control rod. Subsequently, difficulty was encountered in disconnecting control rod drive No. 11 also. While the control rod drive problem was being investigated and necessary repairs were made, the plant remained in standby condition. The work which was performed on the control rod drives is described in more detail in Section I.A.3

One fuel-surveillance subassembly had been installed in row 2 of the core and another in row 4. These were subsequently transferred to the Fuel Cycle Facility. The following were then loaded into the storage basket:

(a) one special stainless steel control subassembly for use in EBR-II kinetics measurements;

(b) one core-type subassembly containing reprocessed fuel;

(c) two inner blanket-type enriched subassemblies;

(d) one inner blanket-type alpha-measurement subassembly;

(e) four outer blanket-type alpha-measurement subassemblies;

(f) one experimental irradiation subassembly containing six encapsulated fuel elements of mixed uranium-plutonium oxides and four capsules containing materials specimens, all provided by General Electric Company, APED;

(g) one experimental irradiation subassembly containing 19 encapsulated fuel elements of uranium-plutonium fissium (and fuzzium) alloys provided by ANL Metallurgy Division.

After reassembly and checkout of control rod drives Nos. 8 and 11, the 14-in. rod drop times were measured for all control rods and for the safety rods, with 100% reactor flow, and found in good agreement with previous measurements.

Control rods Nos. 3 and 10 were calibrated, one against the other, at 100 kW. The results showed no significant change in total worth since the previous measurements. A series of negative transient studies were then conducted at essentially zero power (100 kW) by withdrawing approximately 3 lh with rod No. 7. The entire power trace was logged on magnetic tape and will be digitalized to provide information pertinent to the understanding of the delayed-neutron kinetics.

Power coefficient measurements were conducted over the range from 0 to 45 MW. At each incremental increase in power, the effects of ramp withdrawals of reactivity on power were registered on magnetic tape again, for later analysis and separation into a time-dependent feedback plot. The results of power coefficient measurements agreed essentially with those obtained earlier. Approximately 78.1 lh of reactivity were required to bring the system from zero power to 45 MW at full flow and with a bulk sodium temperature of 700°F, giving an average power coefficient of 1.74 lh/MW.

A comparison of excess available reactivity in the system for two zero-power conditions, one before and the other after the two fuel-assembly replacements, revealed that 22 lh of reactivity had been gained. About 8 hr of this gain is attributed to the two fresh subassemblies. The remainder is believed to be within the limits of reactivity change caused by mechanical readjustments within the reactor during loading.

The reactor is currently operating at 45 MWt. Power operation is continuing to attain about 0.65 a/o burnup in the reactor.

2. Systems and Components

The plugging temperature of the primary sodium remained below 250°F. The purification system was in operation with the cold trap bypassed until late in the month, when malfunction of a sodium-sampling valve caused the entire sampling system to fill with sodium. The purification system was then frozen to allow cleanup.

Measurement of Na^{24} activity at a reactor power level of 45 MWt was found to saturate at 0.57 Ci/liter.

Shortly after the reactor was shut down, sodium was found leaking from the packing of a vent valve in the secondary sodium system, indicating that the bellows had failed. The vent line was frozen to permit continued operation until a spare bellows assembly can be installed. After the vent line was frozen, the cold trap was placed in continuous operation to reduce the possibility of plugging the line.

With the plant in standby condition, it was difficult to control the reverse head on the main secondary sodium pump. To permit work on the secondary pump control system, the primary sodium temperature was reduced from 700 to 580°F during plant standby.

A 3-phase, 50-Amp autotransformer was substituted for the installed pump-control system as a temporary expedient to control low flow rates during plant standby. Meanwhile, a solid-state control device was installed in the control system for test purposes. However, its performance was unsatisfactory, and the autotransformer is currently giving better performance under low-flow conditions. Only limited experience has been gained with this modified system.

The impurity concentrations in the primary argon changed little during the month. During the latter part of the month, chemical analysis showed about 1.5% nitrogen and less than 0.1% hydrogen. The continuous gas chromatograph indicates about 0.3% nitrogen and less than 10 ppm hydrogen in the secondary argon.

The turbine-driven feedwater pump was put in service during the 45-MW power run. Initial operation was satisfactory. However, the relief valve in the pump suction line failed so that the makeup-water system could hardly keep up with the loss. The pump was shut down because of severe speed oscillations, the cause of which was not apparent. The motor-driven feedwater pump was placed in operation pending inspection and repair of the turbine-driven feedwater pump.

The feedwater pressure-regulating valve associated with the motor-driven feedwater pump has continued to cause difficulty in feedwater control at low plant-power levels. However, performance has been improved after maintenance of the valve.

Several attempts were made to operate the turbine-driven condensate pump, but excessive backpressure developed in the exhaust system each time. It appears that feedwater heater No. 1 may not be able to accommodate exhaust steam from the pump turbine along with low-pressure extraction steam from the main turbine and steam from the low-pressure flash tank of the blowdown system. Another attempt will be made to operate the condensate pump with the low-pressure extraction line blocked and with minimum blowdown.

A wiring modification for the 100-kW diesel generator was completed. This modification assures automatic transfer of emergency loads to the 400-kW diesel generator unit if normal electrical power supply is lost and the 100-kW unit stops because of low lubrication oil pressure, water overtemperature, or overspeed. The revised circuitry was tested by simulating a power failure. Both diesel generators performed properly, and the automatic transfer was made satisfactorily.

3. Plant Modification and Maintenance

a. Inspection and Repair of Control Rod Drives Nos. 8 and 11.

During the preparatory fuel-handling operation of opening the control rod drive jaws to release the control rods from their drives, No. 8 control rod drive sensing rod did not drop, although external indications showed that the jaws were open. The No. 8 drive was partially disassembled, and an unsuccessful effort was made to release the sensing rod. Raising the drive and observing the nuclear instrumentation verified that the rod had not been released. Finally, by pulling up on the sensing rod and vibrating the drive, the control rod was released. The drive then was removed from the primary tank at 700°F and a spare installed in its place. Inspection of the drive indicated that one of the two small cam pieces that caused the jaws to function was missing. The missing cam was found in the jaw mechanism. The cam had overtraveled the upper cam surface of one jaw, causing deformation of the upper edge of the jaw and the failure of the two small No. 8 screws holding the cam to the jaw sleeve.

The No. 11 control rod jaw mechanism worked with extreme difficulty. Initially, it was difficult to release the drive from the control rod; however, on a second attempt the jaws would not release the control rod at all. The drive was partially disassembled and the rod released in a manner similar to that used for release of the No. 8 control rod. The No. 11 control drive jaw mechanism was tested outside of the primary tank after cleaning. Performance was not as good as one would expect. Inspection of the No. 11 control drive jaw assembly revealed only burnishing of one cam piece and the matching jaw. The gripper sleeve showed evidence of galling. A second spare control drive shaft assembly was prepared for installation in control rod position No. 11. Certain parts from this replacement drive had been used in repair of the fuel-gripper jaws, and the old gripper jaws and cam pieces from the fuel gripper were still available. With these pieces and a new sleeve the spare unit was assembled. In addition to the two small screws holding each cam piece to the sleeve, a butt weld was made between the two screw holes. The butt weld is intended to hold the cam pieces firmly to the sleeves. After assembly of the control drive motors, the fuel-handling functions of all 12 control drives were checked out. All jaws satisfactorily released the control rods.

b. Argon System Shielding for Fuel Unloading Machine (FUM). The installation of the FUM argon system shielding has been completed. A valve drive was modified to extend through the shielding and up just under the deck plates. This will allow operation of the valve without moving a deck plate and a shielding plug.

c. Turbine-driven Feedwater Pump. The covers of the pump and the turbine were removed for inspection. The pump end was in good condition, but the turbine end had sustained some damage. The turbine wheel was removed to permit further inspection; two buckets were missing from the first row. Both rows of buckets on the wheel, as well as the stationary intermediate buckets, had been damaged by the two loose buckets.

Replacement parts were ordered and a factory representative confirmed that the damage was confined to the turbine interior, with shaft and bearings unaffected. The failed turbine wheel was returned to the factory for inspection to determine the cause of the failure.

d. Other Modifications or Maintenance. The installation of the improved desuperheater spray strainer was completed, and it is now in service. Cooling water was supplied to the bearing housings of the turbine-driven feedwater pump and the turbine-driven condensate pump.

The feedwater pressure-control valve associated with the motor-driven feedwater pump was disassembled for inspection and repair. The lower seat had partially backed out, thus not allowing the valve to seat properly. In addition, the upper and lower guide bushings had broken loose. Temporary repair of the valve was made by installing new guide bushings in the bonnet and end plate of the valve.

4. Fuel Cycle Facility

a. Operations. Two core-type reactor subassemblies were transferred to the FCF for sodium removal after attaining 0.42 and 0.50 a/o burn-up, respectively. One of the cleaned subassemblies has been examined and dismantled.

Scrap irradiated alloy from fuel-fabrication operations was melt refined to obtain a 10.8-kg ingot (92% pouring yield). Two additional ingots of unirradiated alloy were prepared, and a charge of chopped rejected rods was also refined and consolidated.

Three injection-casting runs were made with normal yields, in one of which recycled irradiated fuel was charged. In the subsequent demolding and pin-processing operations, yields of acceptable pins ranged from 47 to 67%. Generally satisfactory results continued to be obtained in element assembly, welding, and bonding. The first subassembly fabricated

from irradiated fuel was completed and returned to the reactor. The radiation level of this assembly was about 1000 R/hr at one foot (decay time averaged about 5 mo). Two additional, unirradiated, core-type subassemblies were also completed remotely.

b. Maintenance

(i) Electromechanical Manipulators. Repair and rebuilding of the electromechanical manipulator, which had been damaged when dropped in the Air Cell last year, has been completed, and the unit is ready for test. The principal cause of delay was the repair and reassembly of the telescoping tubes at the vendor's plant.

A prototype electromechanical manipulator has been obtained and set up in the mockup area of the Fuel Cycle Facility (FCF) for testing and training. Examination of the brushes (Stackpole 660 carbon) of the grip-drive motor from a manipulator carriage removed from the Argon Cell after more than a year's service revealed negligible wear. This is excellent service compared to rapid wear obtained on previously used carbon brushes.

(ii) Fuel-decanning Machine. The fuel-decanning machine was modified by installing a double chain drive for heavier loads which have been encountered. It was reinstalled in the Argon Cell and successfully operated to prepare a charge for the melt-refining furnace.

c. Research and Development. An argon-atmosphere enclosure system¹ has been installed in the Chemical Engineering Division at Argonne, Illinois, in order to test the effect of a dry argon environment on the operation of equipment and to demonstrate processes for use in the EBR-II Argon Cell. The final phase in the construction of the enclosure system² has been completed with the installation of a large inert-atmosphere enclosure, 24 ft long by 12 ft wide by 14 ft high. Argon gas has been introduced into this large enclosure and purification of the enclosure atmosphere is in progress. The concentration of impurities in the enclosure atmosphere has been reduced to 1.5 v/o nitrogen, 56 ppm oxygen, and about 9 ppm water. The water concentration is being maintained at a low level to facilitate the drying out of the large enclosure. Following completion of the drying-out period, the concentrations of impurities in the enclosure atmosphere will be maintained at the same concentrations as those in the EBR-II

¹Chemical Engineering Division Semiannual Report, January-June 1964, ANL-6900, pp. 114-7.

²Two gloveboxes and an argon-purification system were previously installed and placed in operation (see Progress Report for January 1965, ANL-7003, p. 29). The purification system is similar to that installed in the EBR-II Argon Cell.

Argon Cell. These concentrations are as follows: nitrogen, ~5 v/o; oxygen or water, between 20 and 60 ppm each.³ The oxygen and water are being purposely maintained at these levels in the Argon Cell to reduce wear of graphite bearings and motor brushes and to improve operation of equipment having metal parts wetted by sodium.

The plant-scale equipment⁴ for the skull reclamation process is to be tested in the large enclosure. The status of this apparatus is as follows:

1. A skull-oxidation furnace has been installed and is operational.
2. A skull-reduction and purification furnace (M-2) and associated auxiliary equipment (see Progress Report for January 1965, ANL-7003, p. 30) have been installed and are being operationally tested without process reagents for checkout of equipment and controls and for training of personnel.
3. Retorting apparatus has been installed but not yet tested.

A vane-type rotary blower is to be used for recirculating and purging the argon atmosphere within the M-2 furnace. In order to determine if the performance of the blower vanes is affected by the low concentrations of oxygen and water vapor in the furnace atmosphere, testing was carried out with vanes fabricated of various materials. Blower vanes of graphite or graphite-impregnated Teflon were operationally tested in one of the gloveboxes of the argon-atmosphere enclosure system. The concentrations of oxygen and water in the argon atmosphere of the glovebox were each less than 20 ppm. Of the materials tested, the graphite vanes proved unsatisfactory because of excessive wear. The graphite-impregnated Teflon vanes showed only nominal wear; the weight losses after 16 and 104 hr of continuous operation were 2% and 2.4%, respectively. A substantial part of the initial weight loss was apparently due to the wearing-in of the vanes. The rotary blower with graphite-impregnated Teflon vanes has been installed in the large enclosure for use with the M-2 furnace.

The radiation stability of graphite-impregnated Teflon is being determined. Samples of this material are being irradiated in the EBR-II Argon Cell.

³The actual concentrations of impurities in the atmosphere of the EBR-II Argon Cell were recently reported as being: ~5 v/o nitrogen, 30-60 ppm oxygen and 25-50 ppm water (see Progress Report for February 1965, ANL-7017, p. 7).

⁴The equipment to be used in the EBR-II Fuel Cycle Facility is sized for operation on ~5 kg of skull oxide.

B. FARET

1. General

Progress on the Title II design packages beyond that reported previously (see Progress Report for March 1965, ANL-7028, pp. 6-7) is as follows:

Package I - Site Preparation - Drawings incorporating revisions in excavation contour and foundation pads for the heat exchanger have been signed by the A-E, AEC, and ANL. Reproducibles have been issued to the AEC which will enable the preparation of the construction bid package.

Package II - Reactor Vessel - The recommendation for the formal vendor selection previously submitted to the AEC for review has been considered favorable. Correspondingly, the AEC has now authorized ANL to proceed with contract negotiations with the selected vendor, Babcock and Wilcox Company.

Package III - Liquid Metal Heat Exchangers - Specifications and drawings have been signed by the Laboratory and the AEC. Reproducibles have been sent to the AEC for construction-procurement purposes.

Package IV - Liquid Metal Pumps - Specifications and drawings have been signed by the Laboratory and the AEC. Reproducibles have been sent to the AEC for construction-procurement purposes.

Package V - Control and Instrumentation - ANL comments on the initial submittal of drawings and specifications have been completed and returned to Bechtel. As soon as these comments have been incorporated into the drawings and specifications, it is expected that they will be signed at the Bechtel offices by the A-E, AEC and ANL.

Package VI - Special Piping Materials and Valves - Specifications and drawings have been signed by the Laboratory and the AEC. Reproducibles have been sent to the AEC for construction-procurement purposes.

Package VII - Special Early Procurement Items - Specifications and drawings have been signed by the Laboratory and the AEC. Reproducibles have been sent to the AEC for construction-procurement purposes.

Package IX - General Design - ANL comments on the initial submittal of drawings and specifications have been completed and returned to Bechtel. However, several late items (see Progress Report for January 1965, ANL-7003, p. 33) of design which were not released with the initial submittal have recently been received by the Laboratory for review.

Action on these items is not expected to delay scheduled completion of the design package. It is expected that tracings and drawings can be signed at the Architect-Engineer's offices late in May.

The Architect-Engineer has reported that Title II design is approximately 93% complete, as of March 31, and estimated 97%, as of April 30. The remaining Title II design effort is expected to taper off by the end of May. The character of this work is generally described as "cleanup items" as indicated by the itemized package analysis above.

2. Control Drives

To date the bellows seal has successfully completed 3300 cycles, each cycle consisting of moving the drive shaft up and down through a 30-in. stroke at a speed of 10 in./min (see Progress Report for January 1965, ANL-7003, pp. 34-37).

A gripper for the control rods is being fabricated. This gripper is essentially an EBR-II-type gripper with some slight modification to make it more adaptable to the FARET conditions, i.e., 1150°F in a sodium environment. The main parts of the gripper are fabricated of Haynes 25 alloy, which appears to be a promising material for functional parts or load-carrying members at elevated temperatures. A sodium facility to test this gripper at a temperature of 1150°F is under construction. In this facility the gripper will be subjected to the same static and shock loads to be encountered in the FARET reactor, and the facility will also permit the gripper to grip or release the simulated load at scheduled intervals.

3. Core Instrumentation

a. Electrical. Continued seal tests at elevated temperatures in a sodium environment were carried out (see Progress Report for March 1965, ANL-7028, p. 12). Results are summarized as follows:

(i) Sodium vapors in the argon atmosphere in which the specimens were heated showed no appreciable effect on the seal behavior.

(ii) Immersion of the test capsule in sodium did not cause changes in the tightness of the seal. Slow gas leaks tend to be closed off or blocked by liquid sodium.⁵

(iii) However, whenever the sodium bath was considerably colder than the immersed specimen, e.g., by several hundred degrees F, the seal failed, as revealed by the passage of several cc of sodium. This was ascribed to differential thermal expansion of the assembly, a condition not

⁵Grunwald, A. P., Leak Testing of EBR-II Fuel Rods, Nucl. Sci. Eng. 12, 419-423 (1962).

expected in operation of the reactor. The temporary moderate increase in leak rate which typically occurred in previous tests during the heating cycle were probably similarly caused. A gradual and uniform heating with immersion (more typical for reactor conditions) did not result in any leakage of sodium.

(iv) Since silver forms an alloy with sodium it is expected that the silver deposited on the gasket may be dissolved at high temperatures on immersion. Although no "leaching" of silver from the clamped seal surfaces was observed, the silver imprint of the seal on the cavity surface (see ANL-7028, p. 12) was "mottled" where sodium had broken through, indicating that the dissolution of silver may be fairly rapid. For this reason the silver-plated gaskets have been replaced with copper-plated gaskets. Although the copper-plated gaskets require higher clamping pressures, tests on them have been successful.

4. Fuel Assembly Sodium Flow Test Loop

a. Construction. Welding and radiographic inspection of the piping system continued. The 3-, 4-, and 5-in. lines are completed, and the 2-in. lines and valves are being field welded. Final installation of instrumentation into the panelboards is now in progress. Field installation of the pipe-line heaters will begin in May.

The vendor reports that the motor barrel case for the 800-gpm pump is due on April 29, with welding to begin May 3. Due to strike difficulties the pump case is due on May 15.

b. Experimental. The endurance test of two high-temperature strain gauges is still continuing. At this time the gauges have been subjected to about 1.36×10^6 hr-°F; for the most part the maximum exposure temperature was 900°F.

Construction of the frame required for determining the displacement of the end conditions of the pipe has been started. Some revisions of the original design were made to improve vibrational characteristics. As a further step, dashpots will be used in the attempt to reduce vibrations of the frame during the operation of the loop.

The boxes for sighting and reading the indicated pipe displacements are also being constructed. A transparent grid is fixed to the plastic sides of the box in such a way that an effective three-dimensional coordinate system is established.

5. Static Test Vessels

The final drawings of this facility have been completed, checked, and approved for construction, except for wiring diagrams and the insulation-installation drawings.

The large vessel simulates dimensionally, the upper end of the proposed reactor vessel including four instrument penetrations which are oriented similar to the proposed reactor vessel nozzles as shown in the February 1965 monthly report. Likewise, the grid plate spacings are identical to those proposed for the 301 fuel element core. Any penetration in the large flange for gas pressure gauges, supplemental level indicators and the sodium filler pipe will be done at field assembly. The penetrations in the 12 in. blind flanges for test instrumentation and those in the 4 in. blind flanges for heater and spark plug leads will also be done at field assembly. Dummy fuel assemblies 60 in. long have been designed for costing purposes.

Three smaller vessels, of 8-, 6-, and 4-in. pipe size, essentially pipes with 300-lb flanges at the open ends, are intended for testing full-length subassemblies. The internals are readily adapted to suit future test programs. Purchase inquiries have been placed with manufacturers for fabrication of these units.

6. Sodium-Air Reaction

The principal experimental investigation of sodium burning in the form of a spray was by Humphreys.⁶ From a comparison of experimental pressures with theoretical maximum pressures, it was concluded that measured pressures were no more than about $2/3$ of the theoretical maximum. Theoretical maximum temperatures as high as 3500°C were also reported. These high values have been suspect, since the theoretical temperature according to thermodynamic investigation of the sodium-oxygen system is much lower. Support for the idea that limiting temperatures are much lower than 3500°C is given in a paper by Grosse and Conway.⁷ They pointed out that metal burning temperatures are often limited by the decomposition temperature of the oxide and gave a limiting temperature for sodium of about 2000°K .

The most recent compilation of thermodynamic data is the JANAF (Joint Army Navy Air Force) tables compiled by the Dow Chemical Company. These data were used for additional calculations, wherein two steps were used. First, the establishment of the chemical species present, and second, the determination of the equilibrium composition at trial temperatures. The pressures were then obtained from the equilibrium equations.

⁶Humphreys, Jr., J. R., Sodium-Air Reactions as They Pertain to Reactor Safety and Containment, Geneva Conference Paper I893 (U.S.A.), 1959.

⁷Grosse, A. V., and Conway, J. B., Combustion of Metals in Oxygen, Ind. and Eng. Chem. 50, 663 (1958).

Plots were prepared of the energy available and the energy required as a function of trial temperature. The intersection of these curves defined the equilibrium temperature. The partial pressures and total pressure were interpolated from plots of pressure versus trial temperature. The results for temperature and pressure are given in Figure 1. The maximum temperature found was 1950°C for a mixture consisting of 5 atoms sodium to 2 atoms oxygen; the peak pressure was 102 psig for a mixture of 7 atoms sodium to 2 atoms oxygen.

Figure 1. Calculated Adiabatic Pressures and Temperatures for Sodium-Air Reaction

The location of the peak pressure at such a sodium-rich mixture was unexpected. It appears to be due to the effect of oxide dissociation which suppresses the temperature and pressure in sodium-lean mixtures. Beyond a sodium content of 6 atoms sodium to 2 atoms oxygen, the calculations were very simple since there are no dissociation products. The mixtures consist of 2 moles Na₂O (liquid), 4 moles N₂, and Z-4 atoms NaO(g) where Z is total moles. It is only necessary to calculate the energy

balance in order to find the adiabatic temperature. The nitrogen pressure is then $4/(V/RT)$ and the sodium pressure is $(Z-4)/(V/RT)$ while the total pressure is $Z/(V/RT)$.

Beyond a sodium content of about 9 atoms Na/mole O_2 , a new equilibrium phase will form, namely, Na (liquid). Sodium has a vapor pressure of 6 atoms at 1125°C and of 2.6 atoms at 1000°C . Thus liquid Na will be present when the temperature approaches 1000°C .

The calculated results are compared with Humphreys' experimental pressures in Figure 2. The experimental pressures exceed calculated pressures in some cases. However, the calculated curve refers to an initial pressure less than one atmosphere. A single point calculated for 14.7 psia initial pressure shows a considerably increased pressure which is high enough to encompass the experimental points. In general, it appears that Humphreys' experimental pressures were very nearly the theoretical maximum values. More study of the situation is required since the energy content of the hydrogen-oxygen mixture and the water vapor content of Humphreys' system were not considered. The calculated curve should be directly applicable to show maximum theoretical values for the FARET spray-type accident.

Figure 2
Comparison of Maximum Theoretical Pressures with Humphreys' Experimental Points

7. FARET Criticals

Experiments in ZPR-3 with Assembly 46B, the simulation of a 68-liter, carbide-fueled FARET core, were completed during the month. Radial traverses through the core with neutron detectors and reactivity samples were finished, and period measurements were made to provide absolute calibrations of the control rods for this core. Experiments to determine the neutron lifetime in the core were carried out with a pulsed neutron generator.

Analysis of the distributed reactivity coefficient measurements in Core 46B (see Progress Report for March 1965, ANL-7028, pp. 15-8) was completed. The experiments were done in two phases: first, the material substitutions were made in a distribution of nine drawers representing a quadrant of the core excluding the central drawers; then, the worths were found in the central drawers. Table I lists the results of the two series of measurements and also the average worths of the materials in the core derived from these.

Table I. Distributed Worths of Materials in Core of ZPR-3 Assembly 46B

Material	Reactivity Coefficient, $\Delta k/k$		
	9-drawer Distributed Quadrant	Central Drawers	Core Average
Sodium	55.9 ± 0.5	26.4 ± 0.9	55.1 ± 0.5
Stainless steel (Type 304)	9.9 ± 0.1	-1.2 ± 0.2	9.6 ± 0.1
Carbon	132.0 ± 0.7	113.9 ± 1.0	131.5 ± 0.7
Depleted Uranium	7.4 ± 0.1	6.6 ± 0.1	7.4 ± 0.1
Oxygen	92.1 ± 1.5	69.7 ± 1.5	91.5 ± 1.5

The reactivity samples traversed radially through the 46B core were of the fuel materials: plutonium, enriched uranium, and depleted uranium. Analysis of the data is still under way. In the radial traverses of neutron detectors, fission chambers containing U^{235} , U^{238} , U^{234} , and Pu^{239} were used, as well as a BF_3 proportional counter. The relative reaction rates as a function of radius are plotted in Figure 3 for the BF_3 , U^{234} , and Pu^{239} counters.

Figure 3. Radial Traverses of Neutron Detectors through ZPR-3 Assembly 46B

A neutron generator, utilizing accelerated deuteron bombardment of a tritium-absorbed target, was installed behind the ZPR-3 machine, with the accelerator tube running axially through the assembly at the radial edge of the core. The target was thereby positioned about one inch from the reactor midplane at the radial core-reflector boundary. During operation at subcritical levels ranging from about -0.04 to -2.2% $\Delta k/k$, the generator was pulsed at about 1000 cps. The power-decay characteristics at the different reactivity levels were then obtained from neutron detectors at the center of the core. Preliminary estimates from the data, not yet analyzed, indicate a prompt-neutron lifetime of 25×10^{-8} sec.

C. General Fast Reactor Physics

1. ZPR-3

a. Doppler Experiment with Assembly No. 45. The analysis of the experimental data of the Doppler-effect measurements with ZPR-3 Assemblies Nos. 45 and 45A, zone-loaded criticals, has been completed. Assembly No. 45A had a central zone which mocked up a large plutonium-uranium fast power breeder. Assembly No. 45 differed from Assembly No. 45A by having 40% of the sodium replaced with graphite.

The results of the U^{238} measurements are shown in Table II. These measurements agree relatively well with preliminary theoretical calculations. The measurement labeled "Dummy" represents a test for spurious instrumental non-Doppler sources of reactivity. This was done by performing a Doppler measurement which was identical with the first Doppler measurement except that the U^{238} was absent. The experimental uncertainty shown is considered to be relatively generous and will be readily reduced by changes now in progress.

Table II. Uranium-238 Doppler Measurements,
 $\Delta T = 500$ to $1100^\circ K$

Material	Assembly	$\Delta\rho_{lh}$ Exp, (± 0.020)	$\Delta k/k$ Exp ($\times 10^{-6}$)	$\Delta k/k$ per kg ($\times 10^{-6}$)
U^{238}	45	-0.327	-6.39	-27.7
U^{238} , 1.0-in. dia }	45	-1.38	-26.8	-23.3
U^{238}	45A	-0.184	-3.68	-16.0
Dummy (no U^{238})	45	-0.006	-0.117	-

Table III contains the results of measurements with plutonium-239 and mixed plutonium-239 and U^{238} in the same ratio in which they might occur in a fuel element of a large fast power breeder. Because these reactivity signals are small, it is important to make known corrections to the data. One of the most important of these is due to small expansions which occur as the Doppler elements are heated from 500 to $1100^\circ K$. Preliminary theoretical analysis has shown that for Assembly No. 45 the correction seems to be close to the magnitude of the signal observed. Preparations for a more careful evaluation are now complete and the computational results should be available soon.

Table III. Doppler Measurements with Plutonium,
 $\Delta T = 500$ to 1100°K

Material	$\Delta\rho$ Exp, lh (± 0.020)	$\Delta k/k$ Exp ($\times 10^{-6}$)	$\Delta k/k$ per kg ($\times 10^{-6}$)
Pu ²³⁹	-0.060	-1.21	-5.36
Pu ²³⁹	-0.049	-0.99	-4.39
Average	-0.055	-1.10	-4.88
Pu ²³⁹ , Assembly 45	-0.029	-0.044	-0.56
Pu ²³⁹ ; 1 Element	-0.044	-0.028	-0.88
Pu ²³⁹ ; Homogenized	-0.034	-0.016	-0.68
Pu ²³⁹ ; B ₄ C	+0.042	+0.83	+3.70
Pu ²³⁹ /U ²³⁸ ; 1/7	-1.56 (3.4%)	-3.1	-13.6

2. ZPR-6

The installation of the equipment for measurement of the Doppler effect has essentially been completed and the mechanical apparatus tested. Even though several minor items must be corrected, the equipment is ready to be used for starting the measurement program.

It was necessary to align the central tubes of the two halves of the matrix to within ± 0.010 in. to assure free passage of the Doppler drawer through both halves of the reactor.

The rate of leakage of air of the reactor cell was checked and found to exceed slightly the allowable limit of 12% in a 24-hr period at 10 psig. The leaks, resulting primarily from installation of new equipment, etc., during the shutdown period, were reduced to acceptable levels by sealing.

Loading of Assembly No. 4 has begun. Assembly No. 4 is a zoned system with the central region having the composition of a 2600-liter carbide system. The driver region is separated from the central region by a uranium and carbon buffer zone. The critical mass for this assembly is expected to be approximately 1075 kg U²³⁵.

3. ZPR-9

Assembly No. 6 was reloaded into the ZPR-9 facility after the reactor had been shut down for approximately one month for fuel inventory,

matrix alignment, and rod-drive modifications. Unlike the previous loading, opposite quadrants (upper left in the stationary half, and lower right of the movable half) were fueled with pairs of 1/32-in.-thick U^{235} fuel plates rather than the normal 1/16-in. plates, in order to extend the bunching experiments to a more dilute case than was possible with Assembly No. 4. Critical properties in this geometry are given in Table IV.

Table IV. Critical Properties of Assembly No. 6,
First Reloading

Composition: U^{235}	477.5 kg
U^{238}	35.1 kg
W	1803.8 kg
Al	198.2 kg
Fe	4.6 kg
Volume	275.3 liters
Length	61.18 cm
Effective radius	37.92 cm
Excess reactivity	95.56 lh
Edge worth	16.75 lh/kg U^{235}
Critical mass corrected for excess reactivity only	471.83 kg U^{235}

It should be noted that the critical mass of Table IV (corrected for excess reactivity only) is almost 20 kg greater than was previously reported for this assembly. This is probably because the pairs of 1/32-in. fuel plates are 9% lighter than the equivalent 1/16-in. fuel plates, with an accompanying decrease of average fuel worth in the two octants. To check this, the octants in question were reloaded with 1/16-in. fuel plates. The critical properties of this loading are listed in Table V.

Table V. Critical Properties of Assembly No. 6,
Second Reloading

Composition: U^{235}	458.6 kg
U^{238}	33.7 kg
W	1748.6 kg
Al	194.7 kg
Fe	4.6 kg
Volume	267.87 liters
Length	61.18 cm
Effective radius	37.33 cm
Excess reactivity	120.51 lh
Edge worth	16.75 lh/kg U^{235}
Critical mass corrected for excess reactivity only	451.43 kg U^{235}

The last figure in Table V is to be compared with the previously reported critical mass of 453.51 kg U^{235} . The 2-kg difference (0.44% in critical mass) probably is the result of the smaller central gap and the complete change in fuel, as well as uncertainties in the reported values of fuel edge worth.

Bunching of this fuel in the stationary quadrant to 1/8-in.-thick columns added 35.27 lh, and in the movable quadrant, another 37.29 lh. Following attainment of criticality a series of fuel bunching and unbunching experiments were performed. These experiments are currently being analyzed.

4. ZPPR

At the present time, the Title II effort by the architect-engineer is 55% complete.

The Preliminary Safety Analysis Report (Approval Copy, revised January 1965) was transmitted to the AEC in January 1965 and awaits action. Additional analyses of the consequential effects of maximum conditions under Zone-2 earthquake design have been made and indicate that there would be no release of activity under these conditions.

As a part of the Title II effort, a method of testing the filtering efficiency of the gravel-sand roof of the cell was developed and is being incorporated into the design of the facility.

Minor revisions have been made during review of Title II design specifications. The crane in the reactor cell was raised 6 in. to allow installation of at least a 14 x 14 x 8-ft matrix at some future date. Overall mechanical refrigeration instead of a combination of evaporative cooling and mechanical refrigeration will be used for the Support Wing. A major portion of the air-handling equipment of the experimental area under the mound was moved to a less expensive building outside of the mound. An additional inspection hood has been provided in the workroom. The pit for the reactor bed and tables was enlarged. However, these changes have not increased the estimated cost of the facility, which now includes the testing of the gravel-sand roof of the reactor cell.

The status of design specifications and bid solicitations on various components is as follows:

1. Bed and tables: bid package has been received and is being evaluated.
2. Matrix tubes: specifications complete, bids received and being reviewed.

3. Fuel rod drives: bids received and evaluated, order awaiting release of funds by AEC.
4. Matrix drawers: bids solicited, replies due at month's end.
5. Poison rod drives: engineering design about 80% complete.

The present construction schedule of ZPPR includes a bid package for outside steam and electrical distribution in early summer. This will assure a steam supply for curing of concrete during early parts of the construction of ZPPR. To date, schedules have been maintained as originally planned.

D. General Fast Reactor Fuel Development

1. Development of Jacket Materials

a. Development of Refractory Metal Alloys for Service in Oxygen-contaminated Sodium. The internal corrosion behavior of a series of vanadium alloys has been evaluated by means of microhardness traverses of metallographically polished transverse sections of the corroded specimens. The data for the depth of oxygen penetration are coupled with external measurements in Tables VI and VII.

It appears that the total penetration by oxygen in a one-week exposure is inversely related to the titanium content of the alloy. Chromium slightly reduces penetration.

Table VI. Effect of Chromium Content on Corrosion of V-20 w/o Ti in 650°C Sodium Containing about 45 ppm Oxygen

Alloy ^(b)	External Loss per Side, μ	$\Delta W/cm^2$, mg/cm^2	Depth Hardened, ^(a) μ	Total Metal Penetration, μ
TV Control V-20 Ti	45	-16.8	30-35	75-80
V-15 Ti-5 Cr	24	-14.7	40-45	64-69
V-15 Ti-7.5 Cr	21	-15.6	34-38	55-59
V-20 Ti-5 Cr	23	-19.5	32-35	55-58
V-30 Ti-5 Cr	14	-13.1	20-22	34-36
V-30 Ti-10 Cr	18	-14.1	15-18	33-36

Oxygen analyzed 42 ppm at start and 50 ppm at finish; 7.0 days of exposure; samples exposed about 50% cold reduced.

(a) Including adherent oxide.

(b) Numbers indicate weight percent.

Table VII. Effect of Alloy Composition on Corrosion in 650°C Sodium Containing about 45 ppm Oxygen

Alloy ^(c)	External Loss per Side, μ	$\Delta W/cm^2$, mg/cm ²	Depth Hardened, ^(a) μ	Total Metal Penetration, μ
TV Control V-20 Ti	42	-16.3	30-35	72-77
V-5 Ti-15 Cr	9	-9.2	60-70	69-79
V-20 Ti-2 Mo	17	-8.5	40-45 ^(b)	57-62
V-20 Ti-2 Ta	34	-11.2	35-40 ^(b)	69-74
V-20 Ti-2 Nb	34	-14.7	40-50 ^(b)	74-84
V-40 Ti	12	-4.9	15-20	27-32

Oxygen analyzed 42 ppm at start and 43 ppm at finish; 7.0 days of exposure; samples exposed about 50% cold reduced.

(a) Including adherent oxide.

(b) Longitudinal surface cracks in hardened layer, 10-20 μ deep.

(c) Numbers indicate weight percent.

Caution must be exercised in extrapolating these data for a single exposure to longer corrosion periods, since the shape of the corrosion curves has not been determined. Earlier shorter tests had indicated linear corrosion with time for some vanadium alloys in oxygenated sodium.

Sizable excesses of Na_2O_2 added to the system and digested prior to sample exposures were required for maintaining the oxygen content of the system at the specified level throughout a one-week exposure. A test with all samples of the same material (TV-20) showed a 10% increase in corrosion on going from the sample nearest the exit to the one nearest the entrance of the oxygen-replenished sodium stream. Samples with smaller exposed areas are now being used to reduce the depletion of oxygen.

A control sample from a single sheet of V-20 w/o Ti was used in each test run as a monitor of the oxygen content. If the control sample failed to show normal corrosion, the results of the test were discarded.

E. General Fast Reactor Fuel Reprocessing Development

1. Skull Reclamation Process

The skull reclamation process is being developed as an auxiliary process in the EBR-II fuel cycle to recover and purify uranium from residues remaining in melt-refining crucibles. In this process the uranium is recovered and purified by a series of selective oxidations and reductions and associated transfers between magnesium-zinc solvents and a

halide salt flux, and by fractional crystallization of uranium from the solvent metals. The product uranium is recovered as a metal by evaporation of the solvent metals.

Engineering-scale demonstrations of this process on a 1.5-kg-uranium charge basis (~2.0 kg of oxidized melt-refining residue) are being carried out to establish process procedures and to prove the mechanical reliability of prototype equipment. Emphasis has been placed on the development of engineering techniques, especially phase transfers of metal and salt solutions. In two experiments, uranium losses were 2.0 and 3.0% of the uranium charged. These losses are considered low in view of solubility and equilibria losses of 2% inherent in the process and indicate that the process goal of 95% recovery of the uranium will be realized.⁸

In recent runs, steps have been taken to reduce uranium entrainment in a supernatant metal solution during its removal following precipitation of the uranium as a uranium-zinc intermetallic. Uranium entrainment has been roughly proportional to the transfer rate, i.e., high at a high transfer rate and low at a low transfer rate. In order to permit a high rate of transfer of the supernatant solution without excessive entrainment of uranium, a strainer-weir, i.e., a short length of Mo-30W pipe containing small holes at the upper end, was attached to the inlet end of a transfer line in such a way that it extends above the solid cake into the supernatant solution. This has substantially reduced entrainment of the uranium precipitate while permitting rapid transfer of the supernatant solution.

In the second experiment in this series, the removals of cerium and ruthenium were good, but the removals of molybdenum and zirconium were marginal. The poor molybdenum removal has been traced to lack of sufficient agitation; this will be corrected in subsequent runs. The zirconium removal has been limited by incomplete reduction of zirconium oxide and subsequent extraction of the zirconium from the flux phase. Unfortunately, the unreduced zirconium oxide is reduced at a later point in the process, permitting its appearance as zirconium metal in the uranium product. Exploratory runs have indicated that this can be prevented by earlier removal and disposal of this flux. This procedure should greatly improve the zirconium removal and will be attempted in the next run.

2. Blanket Process

The blanket process⁹ is being developed for the isolation of plutonium from EBR-II blanket material. Separation of plutonium from uranium

⁸ In addition to the 2 to 3% uranium losses occurring in the skull reclamation process, a loss of ~2% occurs during the skull oxidation step required to remove the residual uranium from the melt refining crucible.

⁹ Hesson, J. C., Feldman, M. J., and Burris, L., Description and Proposed Operation of the Fuel Cycle Facility for EBR-II, ANL-6005 (April 1963), p. 122.

is based on the high solubility of plutonium in magnesium-rich solutions and the contrasting low solubility of uranium in such solutions. The process involves dissolution of the blanket metal in a 12 w/o magnesium-zinc solution to a uranium concentration of about 14 w/o, addition of magnesium to about a 50 w/o concentration to precipitate the uranium away from the plutonium, separation of phases, and retorting the individual phases for recovery of plutonium and uranium.

A solution-stability run has been made to investigate the suitability of Type 405 stainless steel as a container material for the retorting step in this process. The experiment involved maintaining the Mg-Zn solution containing some plutonium and a little uranium in a Type 405 stainless steel crucible for 100 hr at 730°C. The plutonium-bearing solution was found to be stable under these conditions; therefore, Type 405 stainless steel should be suitable as a material of construction for the blanket process.

3. Materials and Equipment Evaluation

a. Corrosion Resistance of Type 405 Stainless Steel. The corrosion resistance of Type 405 stainless steel to the Cd-Mg-Zn-U/MgCl₂-NaCl-KCl system employed in metal-salt extraction studies has been studied further. Good corrosion resistance had been demonstrated previously (see Progress Report for January 1965, ANL-7003, p. 21) under conditions in which the vapor phase was purged with argon to sweep away any hydrogen or HCl that might have evolved from the melt. A similar experiment without argon purging has been made; no corrosion (<0.5 mil) of Type 405 stainless steel was detected after a 200-hr exposure to liquid metal, liquid salt, and vapor phases at 650°C.

b. Corrosion Resistance of Vitreous Carbon. A new crucible material, "vitreous" carbon, is being tested for containment of liquid metal solutions containing uranium and zinc. In one previously reported test (see Progress Report for March 1965, ANL-7028, pp. 33-34), this material was not wetted by a Zn-Mg-10 w/o U/halide flux system in 24 hr at 800°C. However, in a much more severe test, in which molten uranium was contained at ~1400°C for one hour at a pressure of 10⁻⁴ to 10⁻⁶ Torr, a considerable portion of the uranium was converted to uranium carbide. It is therefore concluded that vitreous carbon is not suitable for use as a container material for solutions containing high concentrations of uranium.

II. GENERAL REACTOR TECHNOLOGY

A. Experimental Reactor and Nuclear Physics

1. Determination of U^{238} Capture Rate

In the investigation of the discrepancy between calculated and measured values of the ratio $\bar{\sigma}_c^{238}/\bar{\sigma}_f^{235}$, it was concluded that no substantial error exists in the fission rate determination. If any experimental error exists, it must lie in the determination of the U^{238} capture rate (see Progress Report for March 1965, ANL-7028, pp. 40-42). The capture measurement is determined from the Np^{239} activity produced by irradiating uranium foils in a thermal flux. The most logical source of error would lie in the counting efficiency.

Since a foil exchange program, as was done in the case of fission-rate determinations, could not be arranged, the beta-counting equipment was recalibrated and sample preparation re-evaluated.

The relative precision of the measurement was determined by irradiating a set of depleted uranium foils in a thermal-neutron flux. The foils were placed on the periphery of an aluminum disc and rotated during irradiation, so that each foil received the same integrated exposure. The irradiated foils were analyzed radiochemically for Np^{239} activity. The results showed a standard deviation of $\pm 3\%$, which was higher than anticipated. The most obvious source of inconsistency was poor preparation of the Np^{239} sample for counting. A new preparation technique was adopted, in which the neptunium activity is electroplated out of solution onto a platinum disc and flamed. This method provides a uniform and almost invisible sample. Foils were again irradiated on the rotating aluminum disc and counting samples prepared by the new technique. The new standard deviation was found to be $\pm 1.1\%$.

The beta-counting system was calibrated by averaging the results from three separate determinations of known Np^{239} samples. At the closest geometry, the value obtained was 0.1337 ± 0.0008 ,¹⁰ in contrast with the earlier value of 0.164. With the use of the new efficiency all measured values of U^{238} captures are increased by a factor of 1.227.

This result brings the experimental U^{238} capture measurements into closer agreement with calculations. Table VIII¹¹ shows a comparison between experimental data and calculated values. In the last column the experimental data has been multiplied by the factor 1.227; the results are to be compared with the values given in the second column.

¹⁰Smith, D., Private communications.

¹¹Avery, R., Hummel, H. H., Hwang, N., Meneghetti, D., Moldauer, P. A., Smith, A. B., Greebler, P., and Nims, J. B., "Physics of Fast Reactors," Third International Conf. on The Peaceful Uses of Atomic Energy, Geneva, Switzerland, Paper No. P/259, September, 1964.

Table VIII. $\bar{\sigma}_c^{238}/\bar{\sigma}_f^{235}$

Assembly No.	Calculated Data	Experimental Data	Corrected Data (x 1.227)
6F	0.101	0.084	0.103
9A	0.106	-	-
12	0.114	0.100	0.123
14	0.116	0.109	0.134
22	0.110	0.091	0.112
23	0.101	0.089	0.109
25	0.113	0.092	0.113
29	0.124	0.115	0.141
32	0.107	-	-
33	0.107	-	-
34	0.122	0.122	0.150
36	0.109	0.097	0.119
41	0.112	-	-

B. Theoretical Reactor Physics

1. Inelastic Scattering from Iron, Nickel, and Chromium

The inelastic scattering of neutrons by the major constituents in stainless steel (Fe, Ni, and Cr) plays a major role in the degradation of the energy of fission neutrons in a reactor system. A study of group-averaged and point-by-point data from five diverse sources¹²⁻¹⁶ showed only minor discrepancies in the total inelastic-scattering cross section. An evaluation therefore need concern only such internal details as level spacing and nuclear temperature.

The resolved level spacings and the effective nuclear temperature can be used to predict the distribution of final energies of neutrons suffering inelastic scattering. Although small differences in the value of these parameters cause negligible reactivity changes in hydrogenic media, they will cause significant changes in dilute fast reactor systems, especially under conditions where the Doppler coefficient may be affected.

The inelastic cross sections for the three constituents are shown in Figures 4-6. The dips at 4 MeV in the Mc^2 plot are due to an error in the code which reads the cross-section tape.

¹²Liikala, R. C., Updated RBU Basic Library, HW-75716 (May 20, 1963).

¹³Hanson, G. and Roach, W. H., Six and Sixteen Group Cross Sections, LAMS-2543 (Nov 1961).

¹⁴Jonau, G. D., Dudek, J. S., Goodjohn, A. J., and Wikner, N. F., Nuclear Data for GAM-I Data Tape, Vol. II, GA-2451 (Aug. 22, 1961).

¹⁵Bondarenko, I., Group Constants for Nuclear Reactor Calculations, Consultants Bureau, New York (Dec 1964).

¹⁶O'Shea, D. M., Argonne National Laboratory.

Figure 4. A Comparison of the Inelastic Cross Sections for Nickel at Various Neutron Energies

Figure 5. A Comparison of the Inelastic Cross Sections for Chromium at Various Neutron Energies

Figure 6. A Comparison of the Inelastic Cross Sections for Iron at Various Neutron Energies

2. Diffusion Theory Calculation

A diffusion theory code, MACH-1, has been prepared for the CDC-3600 computer. The code includes a modified version of AIM-6, regional reaction summaries, total power-flux normalization, a modified version of the DEL perturbation-theory program, beta effective, prompt-neutron lifetime, inhours versus period, flux-integral-weighted group collapsing, and a flux dump on cards in SNARG input format.

C. High-temperature Materials

1. Ceramics

a. Mechanical Properties of Uranium Dioxide. Specimens of UO_2 ground on the tensile face were tested in pure hydrogen, and the results confirmed the suggestion (see Progress Report for March 1965, ANL-7028, p. 47) that molecular hydrogen can dissociate at a porous surface, the atomic hydrogen diffusing through the lattice and increasing strength. The greatest fiber stress at 1500°C ever recorded was obtained under medium straining, with, as expected, a proportional loss of ductility.

A batch of specimens, of high density, was retested at 1000°C, to check past results. The stresses were higher, but not such as to affect the general dependence on temperature previously reported.

A final demonstration of the ductility of UO_2 at 1750°C was made to illustrate the relaxation properties of the material. A single specimen was first bent to a deflection of 0.76 mm under medium strain, cooled, inverted, and straightened by rebending at 1750°C . This procedure was carried out ten times, i.e., five complete cycles. Apart from indentations due to the knife edges, no change in the specimen was observed on completion of the test. Examination of the microstructure showed very considerable increases in the number of etch pits in all grains and in the number of slip systems involved. The dislocation density appeared to have increased by an order of magnitude. These results are shown in Table IX.

Table IX. Rupture Tests on UO_2

Temp ($^\circ\text{C}$)	Number of Samples	Density (g/cc)		Maximum Stress (kg/cm 2)		Mean Strain Rate (mm/sec x 10^4)	Mean Total Strain (mm)	Mean Load Rate (g/sec)	Atmosphere or Conditions
		Mean	Standard Deviation	Mean	Standard Deviation				
1500	4	10.61	0.0001	1970	450	9.1	0.36	176.9	Hydrogen. Ground on tensile face.
1000	4	10.60	0.01	1306	352	5.8	0.18	204.1	Vacuo
1750	1	10.61	-	919	79	13.0	10 x 0.8	86.2	5 cycles of bending and straightening.

Creep Tests on UO_2

Temp ($^\circ\text{C}$)	Density (g/cc)	Load (kg)	Total Strain (mm)	Time (hr)	Minimum Rate (mm/mm/hr) x 10^4	Atmosphere
1000	10.61	41	0.24	6	1.22	Vacuo
1250	10.62	41	0.81	54.5	2.69	Vacuo

Creep tests have been carried out at 1000°C and 1250°C on UO_2 . The ductility is considerably increased by testing at 60 to 75% of the maximum breaking load over periods up to 24 hr. Similar tests will continue at higher temperatures, principally to indicate the effect of relaxation over long times, applicable to reactor fuel conditions.

Preparation of bars of US and UP is now being investigated prior to commencing a program of mechanical tests similar to those conducted with UO_2 .

b. Thermal Stability

(i) US. Further experiments to study the oxidation of US were done. Sintered pellets were oxidized for one hour at 700°C in various atmospheres. The reaction appears to occur at an interface and results in the formation of an adherent product layer.

A specimen reacted in an atmosphere of helium containing 0.1% oxygen showed a weight gain of 1.1%. Microscopic examination of a section through the product layer showed several distinct sublayers.

A weight gain of 0.4% occurred in a specimen reacted in a helium atmosphere containing 0.01% oxygen. The product appeared to be a single phase similar in appearance to the inner product layer of the specimen reacted at the higher oxygen partial pressure. The results indicate that the reaction rate is dependent upon oxygen pressure.

Another specimen was reacted in an atmosphere of helium saturated with water vapor. The reaction proceeded to a much greater extent than in 0.1% oxygen, but the observed weight gain was less, amounting to 0.7%. The product again appeared to contain more than one phase.

The outer product layers of the specimens reacted in 0.1% oxygen, and in water vapor were identified as UO_2 by X-ray analysis. Since the molecular weights of US and UO_2 are virtually identical and no evidence for U_3O_8 was found, the observed weight gain must have resulted from a secondary process. It appears likely that the weight gain is a result of the formation of higher sulfides in the inner product layer. Similar behavior has been observed in the UP and UN systems, in which higher phosphides and nitrides respectively result from oxidation.¹⁷

It also seems significant that the sample reacted in oxygen gained more weight than that reacted in water vapor, even though the extent of reaction was much greater on the latter sample. The evidence may indicate a difference in the oxidation mechanism in the two atmospheres.

Work is continuing on the identification of the product phases and on the determination of reaction rates.

c. Anelasticity of Uranium Compounds. The elastic modulus and internal friction of several uranium oxide specimens were determined. The variables investigated were grain size and stoichiometry. The results are shown in Table X. The grain size was determined by metallographic studies made in specimens identical with the ones used for the measurement of elastic modulus and internal friction. The oxidation was performed by placing the sample and green pellets of U_3O_8 in a sealed quartz tube containing helium at a pressure of 500 mm Hg. The capsule was heated at 1100°C for 42 hr, cooled in 6 hr to 500°C, and then removed from the furnace. The oxygen/uranium ratio was determined by the weight gain. Sample 13 belongs to a batch in which the binder was not properly burned out during sintering. It is not possible yet to make any positive statement with respect to the effects of grain size, owing to the small number of samples tested and the scattering of the results. From the stoichiometric point of view the results were consistent and very interesting. An increase of the oxygen-uranium ratio from 2.005 to about 2.025 produced a decrease of roughly 10% in the elastic modulus.

¹⁷Baskin, Y., Oxidation Behavior of Uranium Monophosphide, J. Am. Cer. Soc. 48 (3) 153-156 (1965).

Table X. Effect of Stoichiometry and Grain Size on Young's Modulus and Internal Friction

Sample No.	Density (g/cc)	Grain Size (μ)	O/U Ratio	E_f (kbar)	E_e (kbar)	$Q^{-1} \times 10^4$
3	10.45	12	2.005	1884	-	8.4
3	-	-	2.027	1650	-	-
13	10.63	-	2.005	2148	2051	4.9
24	10.53	12	2.005	1920	1917	2.8
30	10.41	46	2.005	1888	1954	0.9
30	-	-	2.025	1709	1819	-
32	10.52	-	2.005	2069	2136	1.2
32	-	-	2.02	1898	1971	-

2. Liquid-Metal Corrosion

a. Polarization Studies. An inert-atmosphere glovebox suitable for handling of alkali metals is in operation.

During testing of a NaK gettering system, Teflon contacted the NaK in the dry helium atmosphere. The result was a very rapid reaction with the emission of much light and heat and the deposition of carbon (in the form of block) on the box surfaces. All the Teflon had reacted. After cleanup, the performance of the box was not adversely affected.

b. Lithium Corrosion at Elevated Temperatures. A horizontal high-temperature furnace has been assembled. The furnace is equipped with gas-purification and pressure-controlling systems, and is capable of maintaining a maximum temperature of 1250°C for prolonged operation. The sample section can be either evacuated to 10^{-5} Torr or provided with a positive inert gas pressure of about 38 Torr above atmospheric pressure.

Two capsules, one of molybdenum and one of tantalum (both high-purity, arc-melted products), each containing lithium, were tested at $1200 \pm 15^\circ\text{C}$ in the horizontal position under a slight helium positive pressure. After 100 hr of exposure, there was no evidence of external surface oxidation.

The tantalum capsule suffered both intergranular and trans-granular attack to a maximum depth of 120 μ . Above the liquid level, corrosion attack was less severe. An insert tantalum specimen, which was added for weight determination, showed a weight increase of 38 mg/cm². This weight increase is attributable to corrosion products in the penetrated regions.

The molybdenum capsule exhibited a slight general dissolution type of corrosion in liquid lithium. However, there was no evidence of corrosion above the liquid level. No weight change or evidence of attack was observed on the molybdenum insert specimen.

Microhardness measurements showed no disturbed layer on either of the materials. Effort is being made to evaluate the corrosion behavior of molybdenum and its alloys at higher temperatures and for longer periods.

D. Other Reactor Fuels and Materials Development

1. Nondestructive Testing

a. Determination of Elastic Moduli of High-temperature Materials by Ultrasonics. The object of this program is to measure the temperature dependence of the elastic moduli of materials which show promise in high-temperature reactor applications. To obtain these data a vacuum furnace has been constructed in which solid bar-shaped samples will be mounted. Pulsed ultrasonic waves will be propagated along the length of these samples, and the transit times of pulses passing a heated, indexed region in each bar will be measured. The indexing is done by machining a portion of the length at one end to a smaller diameter than the rest of the bar.

The velocity of sound in the indexed region can then be calculated since one knows the length of this region and the transit time in it. Both shear wave velocities and longitudinal velocities are determined. Any of the elastic moduli, e.g., Young's modulus, Poisson's ratio, shear modulus, can be determined if the sound velocities are known.

Calculations are proceeding for a recent run on TV-20 to 1100°C. A re-evaluation of the error analysis has meant that some previous results had to be recalculated. Final calculations are now under way.

A mirror to view specimens from the bottom (see Progress Report for February 1965, ANL-7007, pp. 56-57) has been made and leak tested. The temperature-measurement system will be checked. We have not been able to use the high-temperature furnace yet, but measurements on the tungsten bar will soon be made.

Room-temperature measurements have been made with several samples of diverse materials. The elastic properties of materials measured recently are reported in Table XI.

Table XI. Elastic Properties

Material	Longitudinal Wave Velocity (10 ⁵ cm/sec)	Shear Wave Velocity (10 ⁵ cm/sec)	Density (g/cm ³)	Young's Modulus (10 ¹² dyne/cm ²)	Shear Modulus (10 ¹² dyne/cm ²)	Poisson's Ratio
W-25 w/o Re (vacuum arc cast; rolled)	5.30	2.93	19.6	4.31	1.68	0.280
W-25 w/o Re (powder metallurgy; pressed, sintered, rolled)	5.24	2.95	19.6	4.33	1.71	0.269
Re (rolled sheet)	5.63	2.90	-----	Not yet completed	-----	-----
UO ₂	5.54	-----	-----	Not yet completed	-----	-----
				Acoustic Impedance (g/cm ² -sec)	Attenuation (db/μsec)	
Araldite 502 Epoxy	2.58	-	1.18	3.05 x 10 ⁵	-	
Araldite 502 Epoxy, fiberglass loaded	2.60	-	1.19	3.09 x 10 ⁵	14	
Araldite 502 Epoxy, silver loaded	1.67	-	2.70	4.50 x 10 ⁵	14	

Measurements of the elastic properties of a rhenium sample have been made, but calculations remain to be completed. The original uranium oxide sample broke during measurements of shearwave velocity, so a new sample must be prepared. The epoxy resin products mentioned in Table XI are being studied for use in the transducer development program, and we have been determining their acoustic impedance and attenuation for longitudinal waves so that these materials may be evaluated in that program.

b. Ultrasonic Instrument and Transducer Development. An important design consideration in the manufacture of ultrasonic transducer probes is the matching of the backing layer and piezoelectric material. The best match is obtained when the acoustic impedance, which is equal to the product of the velocity v and the density ρ of the backing member is equal to the acoustic impedance of the transducer material. This condition gives maximum damping, fast rise and fall times of the vibrations, and short pulse length as all of the energy incident on the rear face of the transducer is passed into the backing member and absorbed. Such features are highly desirable for testing purposes.

In the past month, velocity and density measurements were made on backing layer specimens of the types utilized in ANL-fabricated transducers. It was found that such backing layers do not provide optimum matching for such transducer materials as lead zirconate-titanate and lead metaniobate. In fact, considerable improvement in probe characteristics would be expected if transducers and backing members were more closely matched. Improved backing layers will be investigated in the months to come.

c. Development of a Neutron-image Intensification System. A second neutron-beam facility has been set up at the Juggernaut reactor in order that studies of the intensifier and of neutron-inspection applications can be handled more efficiently. The second beam has characteristics

similar to those of the initial beam, except that the cadmium ratio (about 3.0), chosen for the new beam is somewhat higher than that now used on the initial beam. The neutron-image intensifier system is being set up at the new beam facility.

The delivery of a new neutron-image intensifier tube, containing a window more transparent to neutrons than that of the first tube, has been delayed because of technical problems.

The contrast-sensitivity performance of the neutron-image intensifier system has been analyzed. The predicted contrast, based upon anticipated change in neutron intensity in the lower inspection range of object thickness and upon quantum-fluctuation considerations in the higher thickness ranges, very closely approximates the experimental data. The performance of the inspection system can, therefore, be predicted with a reasonable degree of confidence.

d. Infrared Systems for Nondestructive Testing. The indium antimonide detector of infrared chopped at 2400 cps has been assembled and subjected to initial tests. At object temperatures as low as 95°C, temperature differences of approximately 0.5°C appear to be reliably detectable. Stability problems in the preamplifier have interfered with a more thorough evaluation of the detection system. The preamplifier is now under study.

e. Thermal Conductivity of Irradiated Metallic Fuel as a Function of Burnup. The thermal-pulse method of measuring thermal diffusivity and conductivity is being utilized to determine the thermal properties of irradiated fuel pins as a function of burnup. This method will enable the diffusivity and conductivity to be measured in both the axial and radial directions at many points along the fuel pin, and thus will indicate any anisotropy in the thermal properties, as well as any variation of these properties in the axial and radial directions along the length of the fuel pin.

To check the system, measurements were made with several metals of known conductivity. Armco iron has been used as the standard to calibrate the system. All values as shown in Table XII are accurate within $\pm 10\%$.

One run has been made with uranium-fissium alloys. These results (shown in Table XIII) will be corrected when additional measurements are made. There are differences in the composition of the alloys studied here, and those previously investigated.¹⁸ These differences will not be considered now, but it should be noted that the presence of additional zirconium in the alloy should increase its conductivity.

¹⁸Zegler, S. T., and Nevitt, M. V., Structures and Properties of Uranium Fissium Alloys, ANL-6116 (July 1961).

Table XII. Metals of Known Conductivity

	Run	Check Measurements		Literature Values, $K\left(\frac{\text{cal-cm}}{\text{cm}^2\text{-sec-}^\circ\text{C}}\right)$
		$\alpha(\text{cm}^2/\text{sec})$ at 30°C	$K\left(\frac{\text{cal-cm}}{\text{cm}^2\text{-sec-}^\circ\text{C}}\right)$ at 30°C	
Steel 1020	1	0.127	0.110	0.116(a)
	2	0.117	0.106	0.12(b)
SS304	1	0.350	0.0379	0.038(b)
	2	0.350	0.0378	
Inconel	1	0.0311	0.0363	0.036(b)
	2	0.0324	0.0381	
Uranium	1	0.116	0.0679	0.0680(c)
				0.071(d)
				0.0645(e)

(a)Materials in Design Engineering, Vol. 48, No. 5 (1959).

(b)Metals Reference Book, Vol. 2 (1963).

(c)Zegler, S. T., and Nevitt, M. V., Structures and Properties of Uranium Fissium Alloys, ANL-6116 (July 1961).(d)Weeks, J. L., Thermal Conductivity of Uranium and Several Uranium Alloys, J. Metals 7, AIME Trans. 203, 192 (1955).

(e)Rare Metals Handbook, 2nd Edition (1961).

Table XIII. Thermal Diffusivity and Conductivity of Uranium-Fissium Alloys

Alloy	Experimental Values		Literature Values, $K\left(\frac{\text{cal-cm}}{\text{cm}^2\text{-sec-}^\circ\text{C}}\right)$
	$\alpha(\text{cm}^2/\text{sec})$	$K\left(\frac{\text{cal-cm}}{\text{cm}^2\text{-sec-}^\circ\text{C}}\right)$	
U + 3.3 w/o Fs + 1.7 w/o Zr; 10% enriched	0.084	0.058	
U + 3.0 w/o Fs			0.0541
U + 5 w/o Fs; 20% enriched	0.0600	0.0454	
U + 5 w/o Fs			0.0450
U + 5 w/o Fs + 2.5 w/o Zr; 20% enriched	0.072	0.0519	
U + 5 w/o Fs + 2.25 w/o Zr			0.0474
U + 10 w/o Fs + 5.0 w/o Zr; 10% enriched	0.0414	0.0357	
U + 10 w/o Fs			0.0345

E. Engineering Development

1. Two-phase Flow

a. Void Fraction -- Pressure-drop Facility. The loop test section has been repaired (see Progress Report for March 1965, ANL-7028, p. 54). Four separate flushing operations with clean sodium have been carried out to remove as much oxide as possible from the system. Some plugging difficulties still exist at low temperature; however, the sodium is now clean enough to proceed with the next series of boiling runs in an attempt to extend the quality range to yield larger values of two-phase friction loss.

Quantitative data evaluation is still underway. Some of the void-fraction measurements will be repeated in order to obtain better definition of the variation in two-phase conductivity in an electromagnetic flowmeter.

2. Boiling Liquid Metal Technology

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

The Nb-1% Zr loop construction is underway at Pratt and Whitney-CANEL. The boiler fabrication will be undertaken by Zee Tool and Die Corp., Cincinnati, Ohio, in two steps: gun drilling of the center hole, and electro drilling of the thermocouple holes. The latter holes have been increased in diameter to prevent galling during installation of the sheathed thermocouple assemblies. Preparations for the machining and welding of the 1/2-in. loop tubing are in progress.

Components for support of the sodium loop have been fabricated jointly by the Laboratory and by outside vendors, and delivered to Building D-308 for storage. Heater and preheater units, as well as the required tungsten heater wire, have been received. Receipt of aluminum oxide insulators, bushings and spacers is expected shortly.

Tantalum sheet required for the construction of both the "Reflector and Balancing Assembly" and the "Preheater Reflector Assembly" has been received. These units are about 40% completed. The experimental shutter, which will determine the feasibility of using 0.002-in.-thick corrugated tantalum as a material for construction of the remaining shutters, is about 75% completed.

The "Sodium Loop Support Lower Ring Assembly" design has been finalized. Construction drawings are being checked. This unit will permit adjustment in the vertical alignment of the entire loop by the use of four adjustable legs, thus eliminating extensive use of shimming. Lock nuts will insure permanent fixing once the alignment is made, after which the four leg pads will be spot welded to the vacuum chamber bottom.

In order to insure a dust- and grease-free assembly and storage room, a "clean area" of simple construction is being planned to be located near the vacuum chamber installation. The structural framework will consist mainly of unistrut members. "Filon" flat paneling will be used for walls and ceiling. Provisions for ANL stock air filters and a ceiling exhaust are being incorporated in the room design.

The pressure transducers are approximately 75% complete. The transducer oven is being wired. Calibration will begin in about two weeks.

The specifications for the C-A (chromel-alumel) and (W-3% Re)-(W-25% Re) thermocouple assemblies have been modified. The vacuum chamber flanges have been modified to accept 80 sheathed thermocouple leads and two flowmeter coolant lines. The electromagnetic flowmeter coolant cans and flexible line connections are complete.

Bids have been received on the components for the automatic data-acquisition system. Bid approval and purchasing will be completed in 8 weeks. Work has started on the data-analysis program and the required subroutines.

The vacuum chamber bell jar and upper oven have been received, cleaned, and assembled. The cooling-water circuits were leak checked; two small leaks were located and repaired. The vessel thermocouple leads were routed through an installed junction box. Some lifting hardware was modified for handling the equipment and a work area cleared for a small stand constructed to support the bell jar during final cleaning. After cleaning, the bell jar was placed on the chamber base, where it was observed that the main flanges did not mate properly. Work is presently underway to determine how to correct this defect. All ports on the lower chamber have been leak checked and are believed to be tight. A liquid-nitrogen-supply manifold was constructed and installed, and piping associated with the supply tanks was modified for the purpose of installing rupture discs. One tank was filled to gain handling experience.

b. Heater Experiments

(i) Thermal Radiation Heater Experiments. This experiment has been terminated and a topical report is being prepared.

(ii) Electron Bombardment Heater Experiment. The electron-bombardment sodium pool boiling experiment (see Progress Report for August 1964, ANL-6936, p. 59) began operation on March 30. Accelerating voltage to 20,000 V and emission current to 1 Amp have been obtained. Unsteady current flow was traced to sputtering of silvered joints, which are being replaced with stainless steel. Several other modifications of the system are now underway: additional thermocouples and an argon-purification system are being installed, the vacuum system is being cleaned, and minor instrumentation changes are being made. Upon completion of this work, the system will be degassed and operation continued to the maximum test power of approximately 10 kW.

3. General Heat Transfer

a. Heat Transfer in Double-pipe Heat Exchanger

(i) Liquid Metal Cocurrent Turbulent Flow. The experimental portion of the present investigation has been completed, and detailed analysis of the data has begun.

As mentioned in previous progress reports, distortions of temperature distributions along the outer wall of the heat exchanger were noted, and were thought to be a result of extraneous heat conduction through supporting "flange-type" plates at both ends of the exchanger. These distortions influence the accuracy of the new method being used to determine fully developed heat transfer coefficients, especially at the higher flow rates. Since, ideally, the heat transfer coefficients as well as heat exchanger efficiencies should be independent of the temperature difference between the fluids at the exchanger inlet, whereas heat conduction through the end plates will depend directly on temperature differences, it was decided to perform experiments in which the inlet temperature difference was varied over wide ranges while holding all other operating conditions fixed. At the higher flow rates computed values of heat transfer coefficients and efficiencies were found to depend on inlet temperature difference in a consistent manner. This consistency made it possible to extrapolate computed values of heat transfer coefficients and efficiencies to "zero"-temperature difference, thereby minimizing effects--such as conduction through the end plates--which depend directly on temperature difference. All experiments to determine both heat transfer coefficients and efficiencies have been based on this technique. Typical results are shown in Figures 7 and 8.

In Figure 7 measured heat exchanger efficiencies are plotted as a function of tube-side Peclet numbers, with a fixed flow rate through the annular space. As shown, the measurements compare very favorably with predictions of the analysis. Also shown are predictions based on the customary equation for heat exchanger design with heat transfer coefficients

appropriate to the boundary condition of uniform wall temperature. Although these predictions are significantly lower than the measurements, it is found that if the customary design equation is used with heat transfer coefficients appropriate to the boundary condition of uniform heat flux, the predictions will be very close to the results of the analysis.

Figure 7. Heat Exchanger Efficiency vs. Tube-side Peclet Number in a Cocurrent-flow Heat Exchanger ($L/D_1 = 20$; Annulus Peclet Number = 100; Fluid--Mercury)

Figure 8. Fully Developed Tube-side Nusselt Number vs. Tube-side Peclet Number in a Cocurrent-flow Heat Exchanger ($L/D_1 = 20$; Annulus Peclet Number = 100; Fluid--Mercury)

Figure 8 is a graph of experimentally determined tube-side fully developed Nusselt numbers as a function of tube-side Peclet number, with a fixed flow rate in the annular space. Although the data scatter significantly, the general trend is clearly in agreement with the analysis.

In order to reduce effects of heat conduction through the end plates, a short ($L/D = 8$) test section was built with stainless steel rather than nickel end plates and without a hydraulic entrance region. A special series of experiments was then performed. Since the length of the test section was such that the heat transfer was not fully developed for most of the tests, only heat exchanger efficiencies were determined.

The heat exchanger efficiencies were found to be independent of the inlet fluid temperature difference. In general, the values of the efficiencies were about 8% higher than those predicted analytically, but were about 50% higher than those predicted by the customary design equation.

4. Electric Master-Slave Manipulator Mark E4

Detailed design and fabrication are progressing on a considerable number of parts for both the master and slave arms of this manipulator with 50-lb capacity. About 80 percent of the detail drawings for the slave arm and about 65 percent of the detail drawings for the master arm are completed. About 50 percent of this group of parts is in process of being contracted for or is being fabricated. The servo drive units, except for the X-motion drive and the brakes, have been previously completed. The remainder of the manipulator is about 20 percent completed in detailed design.

An alternative design for the servo control system is being investigated which would permit the consoles to operate at an internal temperature of 60°F. Silicon transistors are used instead of germanium transistors in the power stages.

5. Viewing Systems

The intrinsic absorption of different glass compositions is being studied in order to determine the valence state of certain elements present prior to irradiation as an aid in understanding the radiation-induced changes. The ultraviolet absorption of phosphate glass in the region of 200-250 mμ was found to be mainly due to Fe^{3+} . This is similar to the results previously obtained for aluminoborate and silicate glasses.

The wavelength position of absorption bands due to Pb^{2+} in alkali halide crystals can be expressed by an empirical equation involving the cation and anion radii, such as that for Tl^+ given in the Progress Report for November 1964 (ANL-6977, pp. 62). Although the Pb^{2+} has the same electronic structure as the Tl^+ , the absorption bands due to Pb^{2+} in glasses do not fit the same type of equation as for Tl^+ . This is presently being investigated.

F. Chemical Separations

1. Fluidization and Volatility Processes

a. Recovery of Uranium from Low-enrichment Fuels

(i) Laboratory Support Work. Experimental work has continued in the program to evaluate the use of bromine pentafluoride (BrF_5) as a selective fluorinating agent for uranium in uranium-plutonium mixtures (see Progress Report for March 1965, ANL-7028, pp. 60-61). It is expected that fluorination of uranium-plutonium mixtures with BrF_5 will result in the conversion of uranium to volatile UF_6 while plutonium is converted to nonvolatile PuF_4 . In a subsequent step, plutonium would be recovered as PuF_6 by fluorination of PuF_4 with fluorine.

A series of experiments were performed in a horizontal tubular reactor to determine the extent of uranium removal from mixtures of alumina, uranium tetrafluoride, uranyl fluoride (UO_2F_2), and selected fission product oxides by fluorination with BrF_5 . The fluorinations were carried out for 30 min over the temperature range from 175 to 250°C, with a gas phase containing 20 v/o BrF_5 in nitrogen. At 175°C approximately 30% of the uranium was converted to UF_6 in 30 min, whereas at 250°C, 88% of the uranium reacted to form UF_6 in 30 min. In a similar experiment wherein the mixture was contacted with BrF_5 at 250°C for 1 hr, approximately 98% of the uranium was converted to UF_6 . These data indicate that the use of BrF_5 results in high uranium recoveries in relatively short exposure periods at modest reaction temperatures.

Two experiments were performed in which Zircaloy or stainless steel decladding product formed by reaction of the metal with HF-O_2 mixture was added to the solid reaction mixture containing alumina, UF_4 , UO_2F_2 , and fission products. The fluorination reactions were carried out at 300°C for 1 hr. Over 99% of the uranium was converted to UF_6 in both tests, indicating that the presence of the decladding products does not affect removal of uranium. Approximately 10% of the tin present in the Zircaloy decladding product was lost, apparently by volatilization, during the fluorination step; zirconium was essentially unaffected by the fluorination treatment. Neither iron nor chromium in the stainless steel decladding product reacted to form volatile compounds during fluorination with BrF_5 .

In other Laboratory work a study of the solid-liquid equilibria in the $\text{MoF}_6\text{-UF}_6$ system is being conducted to evaluate the apparatus and measurement techniques. These studies will be followed by studies with the $\text{UF}_6\text{-PuF}_6$ system. The phase diagram appears to be that of a simple eutectic system in which one of the components (MoF_6) undergoes a solid transition at a temperature below the eutectic temperature. The solid transition which occurs at increasingly higher temperatures as the concentration of UF_6 is increased is not completely understood.

(ii) Decladding and Fluorination. Development studies of the decladding and fluorination of UO_2 fuels have continued in the 3-in.-dia pilot-scale fluid-bed reactor. Zircaloy-clad UO_2 fuel elements may be chemically declad by the high-temperature reaction of the cladding with gaseous HCl . Uranium and plutonium are recovered as hexafluorides from the declad fuel by a two-zone oxidation-fluorination technique (see Progress Report for January 1965, ANL-7003, pp. 58-59). In this method, the UO_2 fuel pellets are oxidized in the lower part of the fluidized bed, forming U_3O_8 fines which are elutriated to the upper portion of the bed and fluorinated to the hexafluorides.

In order to provide versatility in the fluid-bed fluoride volatility process, an alternative scheme for processing UO_2 fuels is being studied in the reactor. This scheme comprises three steps: (1) the oxidation of the UO_2 pellets to U_3O_8 fines, (2) the reaction of the U_3O_8 fines with hydrogen fluoride to form UF_4 and UO_2F_2 fines, and (3) the fluorination of the mixture of UF_4 and UO_2F_2 fines with fluorine (or perhaps later with BrF_5) to produce UF_6 . This alternative process may offer several advantages over the two-zone oxidation-fluorination process: (1) sampling of the input fuel material is facilitated, (2) in the two-zone process, the rate of fluorination is governed primarily by the rate of transport of U_3O_8 fines from the lower zone of the bed, which may be widely variable especially in the processing of irradiated fuels, (3) the three-step technique appears readily adaptable to the processing of fuel materials which are already in the form of powders.

The alternative 3-step process was successfully demonstrated in a test with a 2.2-kg charge of UO_2 pellets. The pellets were immersed in a 3.4-kg bed of alumina particles. The processing conditions were:

	<u>Oxidation</u>	<u>Hydrofluorination</u>	<u>Fluorination</u>
Temperature ($^{\circ}\text{C}$)	450	450	450 to 500
Time (hr)	4	3.5	4
Gas Velocity (ft/sec)	1.5 to 1.8	0.9	0.8 to 1.1
Reactant Concentration in Feed (v/o)	5 to 25	4 to 44	5 to ~90

The progress of the oxidation reaction was monitored by continuous oxygen analysis of the off-gas stream and by periodic γ -ray photographs of the pellet bed. The course of the hydrofluorination step was followed by continuous analysis of the moisture content in the off-gas stream. High UF_6 production rates of 55 to 65 lb/(hr)(sq ft of reactor cross section) were achieved during the initial 1.5 hr of the fluorination step. The production rate data indicated that the fluorination of uranium fines proceeds at an initial constant rate, followed by a period in which the reaction rate slowly diminishes.

(iii) Supporting Studies on Fluidization of Fine Particles. In the fluid-bed fluoride volatility processing of UO_2 fuels, several operations are performed in fluid-bed equipment wherein fine particles (<200 mesh) are produced. The production of a large quantity of this fine material greatly affects the behavior of the fluidized bed. During the processing steps a fraction of this fine material is elutriated from the fluid bed and collected on filters located in a cooler zone above the fluid bed. During fluorination it is highly desirable to have a maximum amount of the uranium-plutonium fines in the fluid bed. This may be achieved by frequent blowback of the filters by a reverse flow of gas (nitrogen) through the filter.

A mathematical model was derived for determining the average concentration of fines in the fluid bed as a function of the frequency of blowback interval. This model predicts that blowback frequencies of one every 4 and 8 min result in average concentrations of 96 and 87% of the original amount of fines in the bed, respectively. More frequent blowback of the filters would assure higher fines concentration in the bed, but this operation would significantly dilute the process gas with nitrogen during periods of recycle fluorination.

(iv) Engineering-scale Alpha Facility. An engineering-scale alpha facility is being installed to permit demonstration of the major steps of the fluid-bed fluoride volatility process for the recovery of uranium and plutonium from ceramic fuels. Equipment units for fluorinating batches of UO_2 - PuO_2 pellets to hexafluorides and for demonstrating the conversion of hexafluorides to oxides have been installed, and shakedown testing of the units with uranium materials is in progress. A distillation unit is now being fabricated. A portion of the large glovebox that houses all equipment which will eventually be used for processing plutonium-containing material is shown in Figure 9. All manual operations on processing equipment are carried out through glove ports.

A second shakedown test in the fluorinator was completed with the processing of a 12-in.-deep bed of UO_2 pellets (8.8 kg) using the two-zone oxidation-fluorination technique (see Progress Report for January 1965, ANL-7003, pp. 58-59). The experiment was made in a 19.5-hr continuous period and provided the operating personnel with an opportunity for training and familiarization with operating procedures.

For the first part of the run (11.6 hr), in which the bulk of the UF_6 was produced, a temperature gradient of 150°C (from 300°C at the bottom to 450°C at the top) was maintained across the pellet bed (oxidation zone), while the upper (fluorination) zone was at 450°C . About 95% of the UF_6 was collected in this period at an average rate of about 1.0 kg/hr [$43 \text{ lb UF}_6/(\text{hr})(\text{sq ft reactor cross section})$]. During the final 8 hr of the run, the fluorinator was operated with nearly 100% recycle of the fluorination process gas stream. During this period, the bed was maintained at

490°C and recycle gas containing 50 v/o fluorine was introduced at the bottom of the reactor. Overall, 99.9% of the uranium was collected as UF_6 . The average fluorine utilization was about 50%.

Figure 9

Large Alpha Facility for Study of Fluid-bed Fluoride Volatility Recovery Processes for Uranium Oxide-Plutonium Oxide Fuels

Equipment items such as the fluorine-recycle pump, the refrigeration and control systems, and the UF_6 recording scale all operated satisfactorily. Alternating periods of high and low UF_6 production rates (maximum and minimum were 2200 and 300 g UF_6 /hr, respectively) were experienced in this test; it will be an objective of subsequent work to operate at periods of high, uniform UF_6 production rates.

2. Chemical-Metallurgical Process Studies

a. Sodium Purification. A preliminary study of the nature of carbon impurities in sodium by mass-spectrometric examination of the vapor above liquid sodium indicated the presence of substances with masses above 200 (see Progress Report for March 1965, ANL-7028, p. 65). It is believed that the results discussed below together with those previously reported give evidence for the existence of volatile and nonvolatile carbon species of high molecular weight in relatively pure sodium. The persistence of high-molecular-weight organic compounds in liquid sodium at moderately high

temperatures (300°C or 572°F) was also apparent. The presence of such material in sodium which had been distilled using an oil diffusion pump was also established during this reporting period. This suggests that the failure to effect high carbon removals by distillation may be due in part to contamination during and after distillation.

The volatilization of sodium from a Knudsen effusion cell into a Bendix time-of-flight mass spectrometer has been studied. Equipment, which permitted scans to be taken with and without a shutter in the path between the orifice of the Knudsen effusion cell and the ionization region of the mass spectrometer, was used. Thus the mass peaks which were obtained could be attributed with greater certainty to vapors from within the effusion cell or to residual gases in the instrument (or, possibly, to both sources). Four different samples of sodium, identified below, were examined at temperatures up to 300°C.

Sample No. 3. United States Industrial Chemicals Co. Reactor Grade; average of six analyses showed 25 ppm carbon.

Sample No. 4. Baker and Adamson Reagent Grade; same lot as used in the previous study (see ANL-7028), but melted under an argon atmosphere; analyses showed 69 and 140 ppm carbon. (This spread in analyses is probably due to insoluble material.)

Sample No. 5. Filtered through 5- μ -porosity stainless steel frit after about four months of settling (following one hour of stirring) in a graphite crucible containing graphite powder at 250°C under an argon atmosphere; one analysis showed 79 ppm carbon.

Sample No. 6. Baker and Adamson Reagent Grade; distilled in an all-metal vacuum still, and melted and bottom-poured under argon atmosphere. (This sample is the same as one of the samples analyzed last month by mass spectrometry and found to contain hydrocarbons of high molecular weight.) Total carbon analyses showed about 64 ppm.

The mass spectra of Samples 3, 4, and 5 were similar. With the exception of peaks at masses 23 and 46, presumably due to Na^+ and Na_2^+ , the intensity of peaks obtained with the shutter open was not significantly different from that obtained with the shutter closed. Mass spectra obtained with Sample 6 contained numerous peaks which were significantly more intense than found in the instrument background, particularly at masses 282, 207, 139, and 78. After holding the sample at 300°C for $2\frac{3}{4}$ hr, the intensity of these peaks decreased but remained significantly above background. The same was true of low-mass (40 to 80) peaks, which suggests that they may have originated from fragmentation products of species responsible for the high-mass peaks.

Although these results do not shed light on the origin or nature of carbon impurities in sodium, they indicate that organic compounds of high molecular weight persist in sodium at relatively high temperatures. The results, taken together with the carbon analyses, also show that both volatile and nonvolatile carbon compounds are present in relatively pure sodium. A portion of the type of graphite electrode used in sodium manufacture will be examined mass spectrometrically to determine whether a spectrum is obtained which is similar to that found for sodium containing volatilizable carbon.

G. Plutonium Recycle Reactors

1. EBWR Facility

a. Primary System Piping. Approximately 10% of the faulty 6-in. steam and safety valve piping welds were re-radiographed following repairs. A minor change in repair procedures, which consisted of internal cleaning of pipe bores adjacent to the weld zone, was required to eliminate weld-metal contamination by melted corrosion products.

All radiographs of Type 304 stainless steel piping welds between the reactor and first isolation valve were judged for further service utilizing radiographs of simulated piping welds. The standards of reference, one for each size of pipe, contain known circumferential artificial defects and represent the minimum piping wall thickness for 650-psig service. Numerous repairable defects in the following piping systems (in addition to those described on p. 73 of ANL-7017) are the following:

- (i) 6-in. Feed Water Line. Three of four welds were acceptable.
- (ii) 3-in. Feed Water Line. Seventeen of 25 welds were acceptable.
- (iii) 2-in. Purification Line. Twelve of 13 welds were acceptable.
- (iv) Start-up Heater Piping
 - (a) 3-in. Reactor to Heater. Four of 14 welds were acceptable.
 - (b) 4-in. Start-up Heater Return Line. Seven of 18 welds were acceptable.
 - (c) Welds between Isolation Valves and Start-up Heater. Three of 8 welds were acceptable.

b. Feed Water Filters. The design and the fabrication of the four high pressure filters are still under review.

(i) Design. Preliminary stress analyses reveal that Filters No. 1 and 2 met the original 1952 (and 1956) ASME Section VIII Code "Unfired Pressure Vessel" design criteria with two exceptions: (1) excessive average closure head flange stresses, and (2) overstressed alloy steel bolting. Under the newer ASME Section III "Nuclear Vessel Code," the vessel heads are overstressed (by approximately 6%) because cladding thickness is excluded from strength calculations. Since all seams have now been 100% radiographed, the shell (in Filter No. 1, only) is acceptable under Section III. Flange stresses can be reduced to meet the original design criteria by building up the fillet weld which joins the head flange to the vessel closure dome. Bolt stresses in excess of that permitted by the 1956 Unfired Pressure Vessel Code for alloy steel bolting materials were permissible under a later Code Case ruling No. 1234 and are below the maximum stress permitted by the more recent "Nuclear Vessel" Code.

Filters No. 3 and 4 were designed to meet the requirements of the 1956 Section VIII Code for "Unfired Pressure Vessels." The design met both the 1956 and 1962 criteria (of Section VIII) except for the thickness of the shell flange.

(ii) Nondestructive Examinations. Although all filters were examined by dye penetrant, sonic and magnetic particle (when applicable) methods, the principal inspection tool was radiography. The results are summarized as follows:

Filter No. 1

Dome cover flange weld--acceptable
 Shell closure flange weld--acceptable
 Longitudinal seam--acceptable
 Lower head seam--small repairable crack
 Inlet and outlet nozzle welds--excess porosity and
 repairable cracks
 Repairs required prior to further service

Filter No. 2

Dome cover flange weld--much fine porosity but
 acceptable (undersize)
 Shell closure flange weld--excessive repairable
 porosity in 30% of weld, cracks in 20% of weld
 Longitudinal seam--entire seam cracked
 Lower head seam--excessive repairable porosity in
 25% of weld, cracks in 20% of weld
 Filter unsuitable for further service

Filter No. 3

Dome cover lap stub weld--acceptable
 Shell closure lap stub weld--acceptable
 Longitudinal seam--small repairable crack
 Lower head seam--small repairable crack
 Repairs required prior to further service

Filter No. 4

Dome cover lap stub weld--acceptable
 Shell closure lap stub weld--acceptable
 Longitudinal seam--acceptable
 Lower head seam--acceptable

Surface lamination defect explored by grinding found to be shallow (less than 0.010 in.) and insignificant for strength of shell.

c. Air-lock Doors. The reinforcing members were completed and welded in place on both doors. Extreme care was exercised during welding on the doors as regards weld rod size, welding sequence, and heat to avoid harmful warpage during the repairs. Pressure tests on the doors after the repairs indicated that virtually no warpage had occurred.

2. Fuel Development and Fabrication.

a. Loaded Zircaloy-2 Tubing. A transverse defect was detected in a fuel rod (B-2668) for the Plutonium Recycle loading (see Progress Report for March 1965, ANL-7008, p. 68). The response from this defect was equivalent to the response from a standard inner-surface notch, 0.076 mm deep by 3.18 mm long. After the fuel rod was disassembled (end plug, spring, and slugs removed), the transverse test was repeated. An identical indication was obtained in the suspect area. This confirmed the belief that the indication was not caused by a slug in contact with the inner surface.

Subsequently, the defect area was analyzed by boroscopic, X-ray, and metallographic examination. A diametral void that did not extend to the surface of the slug was disclosed by the boroscope and substantiated by X-ray radiography. This defect is shown in Figure 10. Metallography showed that the void was about 0.025 mm deep and 6.35 mm long.

Fourteen fuel rods were received, inspected, and accepted this month.

50X

Figure 10. Transverse Internal Void Detected by Ultrasonics
in a Plutonium Recycle Fuel Rod

III. ADVANCED SYSTEMS RESEARCH AND DEVELOPMENT

A. Argonne Advanced Research Reactor (AARR)

1. Critical Experiment

a. Theoretical Analysis. Calculations were made to estimate a loading of boron-stainless steel poison strips for the two-thirds loading of 810 fuel foils (identified as Case I). The result was, that with 400 g of natural boron in the core, the available excess reactivity controlled by the control blades will be approximately 4%. With 0.525 g boron per poison strip, this corresponds to a loading of 762 strips. For convenience of loading and for reasons of geometric symmetry, a total of 810 boron-stainless steel strips were loaded, one per fuel foil. The calculations indicate that the effective multiplication factor k_{eff} varies almost linearly with the boron loading over the range from zero to 500 g boron. Therefore, the additional 48 poison strips are estimated to reduce the reactivity of the system by 1.1%. This should leave a reactivity margin sufficient for purposes of the experimental program, e.g., for calibrations of control rods.

To determine the sensitivity of k_{eff} to the choice of resonance-energy-group cross sections for U^{235} in the heterogeneous loading of 810 fuel foils, two extreme cases were studied: (i) the fuel foil is an isolated slab; (ii) the fuel foil is black to neutrons of all energies within the given group. For each of these two cases, group-averaged cross sections were chosen for capture and for fission events, and a 16-group diffusion-theory (REX)¹⁹ calculation was made. The result was that for case (i) the calculated reactivity was lower by 0.4% than for case (ii). The corresponding values of $\Sigma_s \text{ pot}/\text{U}^{235}$ atom were: (i) 1015 b/atom; and (ii) 459 b/atom. Similar computations for the minimum critical loading of 315 fuel foils yielded a negligible reactivity difference.

Six-group cross sections were determined for the minimum critical loading (3.5 fuel foils), using GAM²⁰ and SLOP-1.²¹ In the GAM calculation, the fuel foil was treated as an isolated slab. For an infinite medium, consisting of a slab lattice of the fuel foils and the detailed structure of layers of steel and water, the reactivity effect of a uniform 10% void in the water was -0.05%. Similar computations for the homogeneous volume-averaged composition yielded a reactivity change of -0.20% for a 10% void in the water. Note that these are changes in the infinite multiplication factor, k_{∞} . The overwhelmingly dominant contribution from spatial leakage was not included in this calculation.

¹⁹Butler, M. K., and Cook, J. M., UNIVAC Programs for the Solution of One-dimensional Multigroup Reactor Equations, ANL-5437 (Oct 1956).

²⁰Joanou, G. D., and Dudek, J. S., GAM-1: A Consistent P₁ Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants, GA-1850 (June 1961).

²¹Bohn, H., et al., SLOP-1 A Thermal Multigroup Program for the IBM-704, WAPD-TM-188 (Oct 1960).

b. Measurements. Experiments with the 615-fuel-foil core have been completed. In order to have a reference point for future experiments with core loadings of increased fuel density, the concluding series of experiments included reactivity measurements and "flux" mappings with a beryllium reflector from which the Plexiglas had been removed.

In Figure 11, a radial manganese-activation traverse is shown for the modified system. The foil holder in the beryllium at the corner of the core had been shifted from its position in the earlier traverse in the unmodified system (see Figure 12, and also Figure 19, Progress Report for March, ANL-7028, p. 76). The new position of the core-reflector interface is indicated in Figures 11 and 13. Comparison of Figure 11 with the earlier traverse shows that removal of the Plexiglas caused a rise in the activity in the outer portion of the core, relative to the manganese activity at the center of the internal thermal column (ITC). In the beryllium reflector, the peak relative activity was lowered by approximately 10%, the region of peak activity was broadened, and the activity dropped off much less rapidly with increasing radius.

Figure 11
Manganese Radial
Activation Traverses
in the 615-fuel-foil
System (No Plexiglas
in the Beryllium)

Figure 12
Foil Placement for Mn and
U-Al Radial Activation Traverses
in the 615-fuel-foil System
(Beryllium Reflector; No
Plexiglas in the Beryllium)

Figure 13
U-Al Radial Activation
(U^{235} Fission) Traverses
in the 615-fuel-foil
System (No Plexiglas
in the Beryllium)

Figure 13 shows a radial traverse of fission product activity of bare, highly enriched U-Al foils, normalized to the peak activity in the center of the ITC. In Figure 14, vertical traverses of fission-product activity are shown. The radial locations for these traverses are indicated in Figure 12. These traverses also indicate a possible increase in fuel-region activity as a result of the removal of the Plexiglas from the reflector.

Figure 14. U-Al Vertical Activation (U^{235} Fission) Traverses in the 615-fuel-foil System (No Plexiglas in the Beryllium)

The 615-fuel-foil core has been disassembled for inspection of core components and preparation for reloading to increased fuel density. The 615-fuel-foil core contained 31.4 kg U^{235} , and also included the equivalent of 2,610 0.010-in.-thick stainless steel strips, plus 420 poison strips of stainless steel with 1 w/o boron, containing a total mass of 220.5 g natural boron. The new core loading contains 810 fuel foils, with a total mass of 41.4 kg U^{235} ; 810 boron-stainless steel strips with a total mass of 425 g natural boron; and the equivalent of 2,025 stainless steel strips. There is no Plexiglas in the beryllium reflector.

2. Heating of Materials in the Internal Thermal Column

Heating rates at a reactor power level of 240 MWt have been calculated for aluminum, beryllium, and water in the ITC, excluding effects due

to the presence of target materials. At the midheight of the active core zone, at the radial center of the ITC, the heating rates are conservatively estimated to be: (a) 72 W/g for aluminum; (b) 78 W/g for beryllium; and (c) 138 W/g for water. At the interface of the ITC and the core, the corresponding heating rates are: (a) 79 W/g; (b) 95 W/g; and (c) 170 W/g. These rates were taken to be 20% higher than the corresponding axially averaged heating rates.

3. Heat Transfer

a. Analytical Studies. The third and final phase of a preliminary study of the effect of front-to-back ratio on transient response of heat fluxes from "hot" and "cold" sides of a fuel plate has been completed. Phases one and two of this study were reported previously (see Progress Report for October 1964, ANL-6965, p. 88, and Progress Report for March 1965, ANL-7028, p. 71).

In phase three the specific situation represented by addition of a \$1.00 step in reactivity to the core was investigated. The transient behavior of power and fluid temperature obtained from a preliminary safety analysis was used in the front-to-back study. The responses of the heat fluxes emanating from the "hot" and "cold" sides were monitored and limited by the value of critical heat flux as determined by the Zenkevich-Subbotin correlation. Insulation was assumed to occur when the heat flux from the fuel-plate surface equalled the critical heat flux.

The results of four cases studied are shown in Table XIV, accompanied by the preliminary safety analysis results for the same excursion (Case 5). Cases 1 and 5 show that insulation would be predicted ~4 msec sooner when the front-to-back ratio is explicitly used in the model. Case-5 results include a factor of 1.07 to account for the higher heat flux on the hot side. The effects of front-to-back ratio and clad thickness are also shown in Table XIV. The time when insulation is predicted differs somewhat, but over a small range.

Table XIV. Comparison of Front-to-back Transient Results with AARR Safety Analysis Results

Case	Front-to-back Ratio	Hot-side Cladding Thickness $\times 10^{-4}$ (ft)	Cold-side Cladding Thickness $\times 10^{-4}$ (ft)	Time ¹ q" (hot) Equals Crit Value (msec)	Time ¹ q" (cold) Equals Crit Value (msec)	Hot Side (Btu/hr-ft ² /10 ⁶)	Cold Side (Btu/hr-ft ² /10 ⁶)
1	1.15	3.33	5.00	16.5	22.0	6.63	5.90
2	1.00	3.33	5.00	17.5	21.4	6.53	5.96
3	1.00	4.17	4.17	18.8	18.8	6.25	6.25
4	1.15	4.17	4.17	18.0	19.5	6.36	6.16
5-Preliminary Safety Analysis ¹				20.8			6.275

¹Time with respect to start of \$1.00 step excursion.

The conclusions from this study are:

1. The transient behavior of heat flux cannot be exactly determined by the use of a constant factor to account for the effect of front-to-back ratio.

2. The time when insulation is predicted for this transient study is 4 msec sooner than the time found in the preliminary safety analysis. This difference in time is not believed to have an important bearing on the analysis. Therefore, the original assumption of a constant factor to account for the effect of the front-to-back ratio in the analysis is believed adequate.

b. Experimental Program. The feasibility of measuring void fraction by the gamma-ray attenuation technique for the narrow AARR flow channel was studied, and it was concluded that void-fraction measurements can be obtained. As a result, work has been initiated to install the necessary equipment for void measurement in the AARR steady-state heat transfer loop.

The third test section, 22 in. long and with a heated length of 18 in., has been installed in the loop. The corners of this section have been rounded to avoid peak heat fluxes in the corners; the two sections previously tested had sharp corners.

Modifications have been made to the DC power supply used for the tests in order to achieve power shut-off to the test section when burnout conditions are reached. This feature of the equipment will be checked to determine if more than one true burnout point per test section can be obtained.

The burnout data obtained prior to physical destruction of Test Section No. 2, installed in the single-channel loop with positive-displacement pump, have been analyzed and found as follows:

Exit Pressure, psia	620
Inlet Velocity, ft/sec	42.9
Inlet Water Temperature, °F	137.5
Exit Bulk Water Temperature, °F	431
Average Heat Flux, Btu/hr-ft ²	3.28×10^6
Heat Flux at Burnout Point, Btu/hr-ft ²	4.69×10^6

Burnout conditions were approached by maintaining constant flow rate, inlet temperature, and exit pressure, while raising the power in small increments. Burnout occurred in the corner of the channel approximately 3/8 in. from the exit, as anticipated, due to the previously mentioned peak heat fluxes in the corners of the test section.

Calculation of the critical heat flux using the Zenkevich-Subbotin correlation based on the above testing conditions resulted in a value of 4.57×10^6 Btu/hr-ft², corresponding very closely with the experimentally obtained heat flux at the burnout point.

Corresponding power versus pressure drop measurements obtained prior to burnout of the test section are shown in Figure 15. The data in this figure are being analyzed further in an attempt to postulate the value of the burnout heat flux than can be obtained in parallel channel systems with the same initial conditions.

Figure 15. Pressure Drop vs. Power (Test Section II, 0.040 x 1.25 x 18-in. with Sharp Corners)

4. Reactor Pressure Vessel

The discontinuity stress at the junctures of the cylindrical shell with the closure head of the AARR pressure vessel has been investigated. Due to the differential radial growth of the cylinder and the head, a shearing force P and a moment M must be induced at the juncture to preserve the continuity of the vessel wall. These quantities are determined from the conditions that:

- a. the sum of the edge deflections of the cylinder and the head must equal the difference in radial growth of the cylinder and the head produced by the membrane stresses;
- b. the angle of rotation of the two edges must be equal.

Computations have been made based on the above assumptions at various junctures of the vessel. Stresses obtained from discontinuity have been combined with those due to internal pressure and thermal expansion, and tentative thicknesses of the vessel wall and heads have been selected. However, because of the presence of a heavy mechanical load (from core structure and pressure drop) at the juncture of the cylindrical shell and the ellipsoidal head at the lower portion of the pressure vessel, the shell is not in a momentless state. The results obtained from the membrane-theory analysis become unreliable.

Therefore, this portion of the pressure vessel will be analyzed on the bending theory of shells, even in the preliminary design. However, much more work will be involved because of the higher order of the governing differential equations and the series form of solution.

B. Magnetohydrodynamics (MHD)

1. Rotating Power-generating System

An electrical-power-generation system, for use in a zero-gravity field has been proposed. This system would use centrifugal forces to circulate the working fluid against the generator load and hydraulic pressure drops.

a. Description. One variation calls for rotating a natural-circulation, closed-flow circuit consisting of a nuclear reactor heat source in a peripheral position, connected by a riser and a downcomer to a separator-condenser heat sink near the axis (see Figure 16). The downcomer passes through the magnetic field of an MHD generator. The working fluid is a conducting metal having a mixed liquid-vapor phase in the riser

Figure 16
Diagram of a Rotating
Power-generating System

and a liquid phase in the downcomer. The centrifugal force field causes a greater pressure gradient in the dense fluid of the downcomer than in the two-phase fluid of the riser. The net pressure difference circulates the fluid against the hydraulic pressure drops and impedance load of the MHD generator. Pumps are therefore not required in the system.

The centrifugal force field promotes higher condensation rates of vapor than can occur in a zero gravitational field.

The cycle begins as heat is added by the nuclear reactor to the liquid metal before it enters the riser. The centrifugal force field diminishes in intensity toward the axis of rotation, so that at some point in the riser the coolant drops below saturation pressure, and a lower-density, two-phase flow region is created. The two-phase fluid enters the separator-condenser wherein the vapor rejects heat to the sink and condenses. The liquid leaving enters the downcomer or high-density leg.

Pressure builds up due to the centrifugal force field and increases as the periphery is approached. The fluid is at maximum velocity and pressure as it enters the magnetic field of the generator near the periphery. After conversion of the fluid work energy to electrical energy, the fluid returns along the periphery to the reactor heat source.

In another variation an annular core is placed at the periphery of the rotating system and is used with multiple fluid loops to realize compactness. In this system, it may be possible to use a turbogenerator converter.

A purely analytic standpoint analysis is very complex and involves a nonuniform force field superimposed upon the gravitational force field. Centrifugal and Coriolis forces would modify void formation, two-phase flow (as well as the mechanisms), the processes of separation and condensation, and heat transfer. The variation in position of the liquid-vapor interface or slug flow might lead to inertial instabilities as a result of mass imbalances.

Because of these complications, both experimental and analytical approaches are being undertaken simultaneously.

The requirements of the evaluation call for construction and operation of a rotating heat transfer loop in which the reactor heat source is simulated by an electric heater, the MHD generator by a variable impedance valve, and the metal working fluid by water. This test is being constructed at the University of Arizona to provide a check on the analysis and establish stability characteristics.

C. Regenerative Emf Cells

1. Bimetallic Cells

The sodium-bismuth cell (without thermal regenerator) previously described (see Reactor Development Program Progress Reports for January and February 1965, ANL-7003, p. 74, and ANL-7017, p. 77) has now been in operation as a battery for over 125 days at temperatures from 540 to 630°C. The cell has been discharged and electrically recharged approximately 70 times with no observable decrease in cell performance.

An experimental bimetallic regenerator for continuously regenerating sodium from cathode metals has been constructed of stainless steel. This regenerator is being used to study some of the engineering aspects associated with the separation of sodium from cathode metals by distillation. Among the parameters being investigated are distillation and circulation rates; temperatures of the still, column, condenser, and reservoir; mole fraction of sodium in the cathode metal; and distillation pressure.

The apparatus, shown in Figure 17, consists of a vertical cylindrical reservoir for containing the cathode metal (a 30 m/o sodium-70 m/o lead alloy for the initial studies) and a vertical cylindrical still with a sodium

Figure 17. Experimental Apparatus for the Regeneration of Sodium from Cathode Metals of Bimetallic Fuel Cells

condenser at its upper end. The reservoir is connected to the still by two tubes through which the bimetallic alloy is thermally circulated from the reservoir to the still and return. The sodium vapor distilled from the sodium-lead alloy in the still is condensed at the upper end of the still and returned to the reservoir, where it mixes with the sodium-lead alloy for continuous circulatory operation. During a test run a cup can be positioned under the sodium-return tube to catch the distilled sodium for a predetermined period of time. This allows measurement of the distillation rate and provides a sample of the distilled sodium for analyses. A test run must be terminated to remove the sample; for the subsequent run an amount of sodium equal to the sample removed is added to the reservoir to maintain the alloy composition.

To date, 27 test runs have been made for a total operating time of 88 hr for the regenerator. Still temperatures of about 750 to 800°C and distillation pressures of 3 to 7 mm Hg have been used. Total distillation rates of about 14 g of Na/hr or specific distillation rates of about 4.8 g of Na/(hr)(cm² of still evaporating surface) have been attained. Distilled sodium with less than 0.5 m/o lead has been obtained; this would be suitable for the anode in a sodium-lead cell. In a bimetallic cell, current densities of about 500 mA/(cm² of electrode area), which are high values, would require only about 0.5 g of Na/(hr)(cm² of electrode area). Thus, the distillation surface area of a regenerator such as the above could be considerably less than the electrode area of an associated cell.

IV. NUCLEAR SAFETY

A. Reactor Kinetics

1. Fast Reactor Safety

a. Coolant (Water) Expulsion Studies. The physical set-up for the water-expulsion tests has been completed. However, several instrumentation problems have been encountered. The most critical problem is obtaining a suitable earth ground for the equipment and instrumentation.

The electrical resistance of the test element has been measured as a function of temperature in the range from 70 to 1700°F. Based on the results, it was determined that the elements used as a resistance thermometer would not be sufficiently accurate. Optical pyrometry was also investigated, but was found to be unsuitable because of the water surrounding the test element. Therefore, thermocouples will be used for temperature measurement.

b. High-temperature Equation of State for Alkali-metal Vapors. The thermodynamic properties of metal vapors are of considerable interest to fast reactor safety. Since these substances are used or are being considered for advanced reactors, magnetohydrodynamic power generators, thermionic devices, ion propulsion systems, and current rectifiers, there is additional practical, as well as theoretical, interest in their properties. The alkali-metal vapors are of particular interest because of their relatively high vapor pressure and ease of ionization.

The equation of state and derived quantities, as well as transport properties of metal vapors, can be determined once the potential energies of the interactions of atoms and molecules in the vapor are known. As the alkali-metal vapors contain only one valence electron, two atoms in their ground state will interact along one of two possible potential curves, depending on whether their electron spins are parallel or antiparallel, corresponding to the lowest $^3\Sigma$ and $^1\Sigma$ states of the diatomic molecule.

In the present state of knowledge of such interactions, calculations which do not take into account the existence of two or more interactions are physically meaningless. Estimates of Lennard-Jones potential parameters from boiling points and densities for calculation of the properties of the vapor at elevated temperatures, although widely used,^{22,23} have no real physical bases and lead to incorrect results.

²²Robinson, L. B., Thermodynamic and Transport Properties of a Cesium Plasma, Adv. Energy Conv. 3, 19 (1963).

²³Gottlieb, M. and Zollweg, R., Thermal Conductivity of Cesium Vapor, Adv. Energy Conv. 3, 37 (1963).

Because band spectral data exist and have been analyzed for the alkali-metal molecules, the "true" potential energy curves for the $^1\Sigma_g^+$ molecular state can be determined over limited ranges of internuclear separation by means of the Rydberg-Klein-Rees technique.²⁴ The shapes of the curves were obtained by plotting the potentials in reduced coordinates:

$$V^*(r^*) = V(r)/\epsilon, \text{ the reduced potential energy,}$$

and

$$r^* = r/r_e, \text{ the reduced internuclear separation,}$$

where

V = potential energy

r = internuclear separation

ϵ = depth of potential well

r_e = equilibrium internuclear separation.

Comparison of these curves with the shapes of empirical potential showed that the Lennard-Jones (12-6) potential was inapplicable. The Rydberg potential in reduced form:

$$V_R^*(r^*) = -[1 + b(r^* - 1)] \exp\{-b(r^* - 1)\},$$

gave the best fit. This was also shown by Varshni²⁵ from other considerations. The values of b were calculated from the band spectral constants, and no attempt was made to adjust them for a better fit.

Calculations of the second virial coefficient $B(T)$ in the virial equation of state:

$$PV/RT = 1 + B(T)/V + C(T)/V^2 + \dots,$$

were made by direct numerical integration from the relationship

²⁴Vanderslice, J. T., Mason, E. A., and Maish, W. G., Ground State of Hydrogen by the Rydberg-Klein-Rees Method, J. Mol. Spectr. 3, 17 (1959); [Erratum: 5, 83 (1963)].

²⁵Varshni, Y. P., Comparative Study of Potential Energy Functions for Diatomic Molecules, Rev. Mod. Phys. 29, 664 (1957).

$$B(T) = 2\pi N_A \int_0^\infty \{1 - \exp[-V(r)/kT]\} r^2 dr,$$

where

N_A = Avogadro's number

T = temperature

k = Boltzmann's constant,

by defining the following reduced integrals:

$$B^*(b, T^*) = B(T)/2\pi N_A r_e^3 = \int_0^\infty \{1 - \exp[-T^*V^*(r^*)]\} r^{*2} dr^*,$$

where

$$T^* = \epsilon/kT.$$

For the $^1\Sigma$ curves the Rydberg potential was used and the results agreed well with the results of Sinanoğlu.²⁶ For the $^3\Sigma$ curves a purely repulsive potential was constructed from the Morse potential:

$$V_M^*(r^*) = \{\exp[-2c(r^* - 1)] - 2 \exp[-c(r^* - 1)]\},$$

according to the suggestion of Keyes.²⁷ This led to the "antibonding" Morse potential

$$V_{MA}^*(r^*) = \{\exp[-2c(r^* - 1)] + 2 \exp[-c(r^* - 1)]\},$$

where the constant c was chosen to give the same exponential slope as the Rydberg potential:

$$c = b - \log_e 2.$$

The results of the integrations are represented by B^1 and B^3 for the singlet and triplet, respectively. These must be combined, according to Sinanoğlu, to give

$$B(T) = \frac{1}{4} B^1 + \frac{3}{4} B^3.$$

²⁶Sinanoğlu, O., and Pitzer, K. S., Equations of State and Thermodynamic Properties of Gases at High Temperatures; 1. Diatomic Molecules, J. Chem. Phys. 31, 960 (1959).

²⁷Keyes, R. W., Morse Potentials for Excited States of Diatomic Molecules, J. Chem. Phys. 29, 523 (1953).

Over the range of temperature of interest, B^1 was negative and decreased with increasing temperature; B^3 was positive and decreased only slowly with temperature. At low temperatures the singlet potential contribution is dominant and

$$B(T) \approx \frac{1}{4} B^1.$$

This shows that, even if the Lennard-Jones potential had the correct shape, it would have to be weighted by $1/4$ to give the correct results. At some high temperature the contribution from the triplet potential becomes dominant, and $B(T)$ changes sign, becoming positive. In the approximation of the second virial coefficient, $B(T_B) = 0$ defines the Boyle temperature. Were it not for the positive contribution of the triplet, the Boyle temperature would be predicted to be extremely high.

In Figure 18 the virial coefficient of potassium is plotted against reciprocal temperature. The contribution from the singlet potential, the total corrected for the contribution of the triplet, and results of the Battelle²⁸ and the Naval Research Laboratory²⁹ equations are shown. The agreement is extremely good, considering that no adjustable constants have been used. The NRL equation, based on a wider range

Figure 18. Second Virial Coefficient of Potassium

²⁸Lemmon, A. W., Deem, H. W., Eldridge, E. A., Hall, E. H., Matolich, J., and Walling, J. F., Engineering Properties of Potassium, NASA-CR-54017 (BATT-4673-Final)(December 31, 1963).

²⁹Ewing, C. T., Stone, J. P., Spann, J. R., Steinkuller, E. W., Williams, D. D., and Miller, R. R., High Temperature Properties of Sodium and Potassium, 12th Progress Report, July 1-September 30, 1963, NRL Report 6094 (June 9, 1964).

of experimental data, shows incorrect curvature at high temperatures and would not yield the Boyle temperature; the Battelle equation predicts the correct curvature and agrees well with theory over an extremely wide temperature range in slope and magnitude.

In Figure 19 the theoretical results for sodium are compared with an equation based on NRL data.³⁰ The agreement is excellent, although again the curvature of the NRL equation is incorrect at high temperatures. A comparison was also made with recent results for cesium from NRL,³¹ and the agreement is satisfactory. Since no corrections were made in these calculations for diatomic and higher polymeric species or ions, it is believed that if taken into account some of the discrepancies might be reduced.

Figure 19. Second Virial Coefficient of Sodium

B. TREAT

1. Operations

Fast Ceramic Reactor fuel sample C-3-B received a second transient irradiation and was returned to the Vallecitos Laboratory for examination. Transient irradiations of FCR fuel are being conducted for the General Electric Company.

Four samples of High Flux Isotope Reactor fuel were irradiated in room-temperature water and returned to Argonne, Illinois, for examination.

³⁰Stone, J. P., Ewing, C. T., Spann, J. R., Steinkuller, E. W., Williams, D. D., and Miller, R. R., High Temperature Properties of Sodium, Potassium, and Cesium, 14th Progress Report, January 1.-March 31, 1964, NRL Report 6128 (June 4, 1964).

³¹Ewing, C. T., Naval Research Laboratory (private communication).

Examination of the fuel and cladding from a standard TREAT fuel element that had been in the reactor since the start of operations in 1959 did not reveal any evidence of deterioration. During the manufacture of these elements, the blocks of fuel were sealed in the Zircaloy cladding under vacuum. A good vacuum remained at the time the cladding was opened for examination of the fuel. Metallographic examination of the Zircaloy cladding showed a very thin oxide film, and there was no evidence of accelerated oxidation in the weld areas.

Experiments in the TREAT reactor are in progress to study accident conditions involving material movement in fast reactor core designs. Also in progress is the auxiliary out-of-pile work needed to obtain information for analysis of experimental results and instrument response.

Earlier experiments on the failure thresholds and modes of failure of EBR-II-size, gas-bonded UO_2 elements are to be extended. Cylinders of UO_2 fuel, of 0.384-cm diameter and in nominal lengths of 2.5 cm, of 13% enrichment are being produced by means of an extrusion technique. Destructive experiments are scheduled to study effects of high burnup, sodium bonding of oxide to cladding, and transient heating in a sodium environment.

a. MTR Irradiations. A batch of 10 capsules, each containing two UO_2 irradiation specimens, has been prepared for steady-state irradiation in MTR. Each specimen consists of a stack of oxide pellets, approximately 14 cm long, bonded by argon to standard EBR-II stainless steel cladding. Irradiation specifications call for burnup ranging from 1.3 a/o to 5.9 a/o for the samples. Nominal centerline fuel temperature calculated for the specimens is 2330°C , although this is expected to change appreciably over the irradiation time as the physical condition of the ceramic changes and the U^{235} is depleted.

b. Sodium-bonded Oxide Samples. A short survey has been made of techniques for sodium bonding the 0.384-cm-dia oxide cylinders to EBR-II-sized stainless steel tubing. Bonding techniques tested were:

- (i) 16-hr heat treatment ("soaking") at 700°C , water quench;
- (ii) 16-hr soaking at 550°C , air cooled to 200°C , and water quench;
- (iii) 2-hr heat treatment with vibration at 550°C , water quench.

Three specimens were given each treatment. In addition, argon-bonded pins were given identical heat treatment as controls with respect to thermal shock. Each treatment gave acceptable bonding, as determined by cyclograph eddy-current scans of the sodium distribution inside the cladding, and dissection after the cyclograph testing. Each technique also resulted in sodium-impregnation of the UO_2 (which was of approximately 93.8% theoretical density). A check of thermal shock effects on the samples is underway.

c. Calculations of Transient Temperature for UO₂ Samples. In the past, calculations of transient temperature of fuel samples have been performed through use of finite-difference equations based on the assumption that the thermal conductivity stays essentially constant over the calculational cell associated with each mesh point. In accord with this assumption, the basic heat transfer equation for radial heat conduction in cylindrical symmetry is

$$\frac{\partial T}{\partial \tau} = \frac{k}{\rho c_p} \left(\frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} \right) + q,$$

where T is temperature, τ is time, k is thermal conductivity, ρc_p is volumetric heat capacity, r is radius, and q is specific power. This has been adequate in past calculations for both metallic and ceramic fuels.

The metal and mixed uranium-plutonium oxides, however, do have thermal conductivities which vary considerably as a function of temperature. Transient experiments with sodium may produce large temperature gradients.

If the variation in k is taken into account, the heat transfer equation becomes

$$\frac{\partial T}{\partial \tau} = \frac{k}{\rho c_p} \left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} \right] + \frac{1}{\rho c_p} \frac{\partial k}{\partial T} \left(\frac{\partial T}{\partial r} \right)^2 + q.$$

Preliminary calculations indicate that the $\frac{\partial k}{\partial T} \left(\frac{\partial T}{\partial r} \right)^2$ term produces a significant effect for rod-type elements of sizes projected for fast power reactor cores at power densities of interest for fast reactors, and under at least some accident conditions. More detailed calculations are being set up.

d. High-temperature Out-of-pile Studies. The modification of the tungsten-filament high-temperature resistance furnace has been completed, and the facility is prepared for operation. A lateral view port into the hot zone of the 3-in.-dia tubular filament was added to the furnace to permit continual remote monitoring of temperature while leaving the access aperture at the top available for the insertion of experimental apparatus. Operational tests of the furnace indicate that the device has a temperature capability of $>3000^\circ\text{C}$, and can be operated at near-maximum temperatures for extended periods.

2. Large TREAT Loop

The construction of the large TREAT loop has been resumed with the shipment of the bellows assemblies to the Idaho site. During the process of welding, one of the single bellows was damaged; this unit will have to be replaced. The incorporation of the other assemblies into the loop is continuing.

A redesign of the test-section liner and insulation region has been initiated because of considerations of the nuclear cross sections. The liner will be constructed of Zircaloy rather than steel. Also, additional graphite will be supplied in the external void region to compensate for reactivity effects.

Keller Industries of Hanover, N. J., will construct the surge-suppressor portion of the test section. The shop work on the remainder of the test section is approximately 10% complete.

All pipe, tubing, and fittings required for the construction of the gas system have been obtained from stock or placed on order. The only items remaining to be purchased for the gas system are valves, pressure gauges, and flexible hose. Samples of the ball, gate, and pilot-operated solenoid valves that are specified for the gas system have been obtained and are being tested for seat and stem leak tightness.

Current production units of high-temperature electrical connectors capable of extended operation at 1000°F from several manufacturers appear suitable for service on the test section. Pending further information, the required units have not been ordered.

C. Chemical and Associated Energy-transfer Problems in Reactor Safety

1. Metal-Water Reactions

a. Extent of Metal-Water Reaction during a Loss-of-coolant Accident. Metal-water reaction data were applied to the analysis of a loss-of-coolant accident. The loss-of-coolant accident resulting from a sudden break in a primary system water pipe has been taken to be the "maximum credible accident" for most of the power reactors built to date. Water loss following the pipe break results in the core being left in contact with steam. The decay energy in the core is sufficient to cause the fuel cladding to reach temperatures in a short time at which a metal-steam reaction can occur. The heat generated by the metal-steam reaction may add considerably to the total energy which must be dissipated safely within the containment building. The hydrogen gas generated by the reaction causes an additional fire and explosion hazard. The decay energy augmented by chemical energy probably would result in core meltdown and collapse into the bottom of the reactor vessel. If water is still present in the vessel, additional metal-water reaction can occur.

Since Zircaloy-2 is likely to be the cladding material in many large power reactors, an analysis of the effect of the cladding reaction in a loss-of-coolant accident for such a reactor was performed. The time sequence of the core meltdown is found to be markedly affected by the zirconium-steam reaction because of the large energy release associated with the reaction.

The analysis makes use of the proposed LOFT (Loss of Fluid Test) reactor as a model, except that the calculations were made for a 25-mil Zircaloy-clad core rather than the 15-mil stainless steel-clad core proposed in the LOFT design. The core was divided into ten power sections. The reported power distribution in the LOFT reactor was used.

The rate of the zirconium-steam reaction was calculated on the basis of a parabolic rate law and the assumption that sufficient steam was available to react at the calculated rates. A rate based on the supply of steam was used to calculate the reaction when limited amounts of steam were assumed to be available. An energy balance equating the internal energy of each section to the chemical and decay energy developed in each section was used to calculate the temperature-time histories of the ten sections. A flat temperature profile through the fuel was assumed, and heat losses were ignored. The presence of support structures was also ignored. The extent of reaction of the entire core was calculated by summing the contributions from each core section.

Because of the uncertainties in the analysis of the blowdown period during which the water leaves the core, two alternative assumptions were made to establish the initial temperatures of the sections in the core. In one case it was assumed that sufficient cooling had occurred to reduce the temperature of each section to the moderator (water) temperature at power operation. In the second case it was assumed that no cooling had occurred during blowdown and that the initial section temperatures were those attained during power operation.

Extent of zirconium-water reaction at the times the core reached average temperatures equal to the melting point of (1) zirconium cladding (1852°C), and (2) the urania core (2800°C) is shown in Table XV. The results demonstrate a strong interdependence of the core temperature

Table XV. Core Collapse Time and Extent of Zirconium-Water Reaction
Based on the Average Nuclear Decay and Chemical Energy Content

(Analysis based on LOFT-type reactor utilizing 25-mil Zircaloy-clad core)

Steam Flow	Case of Average Core Energy Sufficient to Melt Cladding Completely (1852°C)				Case of Average Core Energy Sufficient to Begin to Melt Fuel (2800°C)			
	Collapse Time, sec		Percent Metal Reacted		Collapse Time, sec		Percent Metal Reacted	
	A ^a	B ^b	A	B	A	B	A	B
Unlimited	465	105	40	27	765	240	62	50
1000 lb/hr	740	390	21	16	1080	685	41	34
100 lb/hr	1200	700	6.0	4.0	2090	1500	13	10
No Metal-Water Reaction	1400	1100	-	-	2600	2300	-	-

^aCore temperature initially that of moderator: 544°F (285°C).

^bCore temperatures initially those at operating power (50 MW).

history and the extent of reaction. Since no satisfactory model for the collapse of the core exists, the times shown in the table were taken to represent those required for the core to collapse onto the lower support structure. In the case of little or no chemical reaction, the time required to bring about the collapse of the major portion of the core can be assumed to be the time required to melt the metal cladding.

In the case of the metal cladding being converted extensively and rapidly to oxide, a better estimate of the time required to bring about the collapse would be the time required for melting of the uranium fuel. The course of the accident subsequent to core collapse depends upon how much water is in the reactor vessel following blowdown and the time required to melt through lower support structures. If the core collapses into a nearly dry region of the reactor vessel, it is likely that the core would form a random pile of UO_2 , ZrO_2 , and Zr which might or might not melt through the vessel. In the case of such a collapse, further reaction would proceed slowly by diffusion of steam or air into the mass. However, if the hot core material were to collapse into a large quantity of water, then a rapid quenching reaction should occur. If a major portion of the core falls at one time, a violent steam explosion might occur. It seems unlikely, however, that a major portion of the reactor core debris would drop at one time. The extent of the quenching reaction can probably be estimated from results of condenser-discharge experiments and meltdown experiments in TREAT. Results of zirconium-water reaction studies by both methods indicated that the extent of reaction would be quite limited ($< 15\%$) unless the temperature of the debris was above 2600°C , the melting point of ZrO_2 . In this case, the ZrO_2 would be molten, and extensive particle subdivision and greater metal-water reaction could occur.

PART V. PUBLICATIONS

Papers

OXIDATION BEHAVIOR OF URANIUM MONOPHOSPHIDE

Yehuda Baskin

J. Am. Ceram. Soc. 48, 153-156 (March 1965)DISCUSSION OF "MAXIMUM FLOW RATE OF A SINGLE-COMPONENT,
TWO-PHASE MIXTURE," BY F. T. MOODY

H. K. Fauske

J. Heat Transfer 87(1) 141 (February 1965)DOPPLER EFFECT CALCULATIONS WITH INTERFERENCE
CORRECTIONS

R. N. Hwang

Nucl. Sci. Eng. 21, 523-525 (April 1965)

SCATTERING OF FAST NEUTRONS FROM Ag, In, AND Cd

W. G. Vonach and A. B. Smith

Bull. Am. Phys. Soc. 10, 498 (April 1965) Abstract

The following appeared in Proc. American Nuclear Society National Topical
Meeting on Fast Reactors, Detroit, April 26-28, 1965. ANL-100

EBR-I AND EBR-II OPERATING EXPERIENCE

Meyer Novick, F. D. McGinnis, and G. K. Whitham

pp. 25-40

METALLIC FUEL ELEMENTS FOR FAST REACTORS

J. H. Kittel, F. Sebilliau, and R. G. Rateick

pp. 157-169

CRITICAL ASSEMBLIES

F. W. Thalgott, A. R. Baker, and S. G. Carpenter

pp. 225-243

FAST REACTOR TEST FACILITY (FARET) INITIAL EXPERIMENTAL
PROGRAM

W. R. Simmons, P. J. Persiani, and T. R. Bump

pp. 273-274

URANIUM-PLUTONIUM CARBIDE FUELS FOR FAST BREEDER
REACTORS

A. Strasser, C. Wheelock, and L. A. Neimark

pp. 126-156

Also, United Nuclear Corporation Report UNC-MTL-4470

(April 1965)

FAST REACTOR FUEL HANDLING

L. J. Koch

pp. 333-347

ANL Reports

- ANL-6845 STUDIES OF FAST REACTOR FUEL ELEMENT BEHAVIOR
UNDER TRANSIENT HEATING TO FAILURE. II. In-pile
Experiments on UO_2 Samples in the Absence of Coolant
C. E. Dickerman, L. E. Robinson, E. S. Sowa, and
S. B. Skladzien
- ANL-7006 CORROSION OF EXPERIMENTAL THORIUM-BASE ALLOYS
James Y. N. Wang
- ANL-7010 REACTOR PHYSICS DIVISION ANNUAL REPORT, July 1,
1963 to June 30, 1964