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REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT

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DISCLAIMER

FOREWORD

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

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

The last six reports issued in this series are:

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I. WATER COOLED REACTORS (040101)

A. EBWR

1. 100-Mw Modifications - Reboilers

Struthers-Wells received the reboiler tubing from the tubing vendor in the early part of July. Bending of the tubing has been completed and the tube bends are being annealed. A new procedure for fabricating the reboilers has been reviewed by the Laboratory.

Specimens of the tubing for the reboilers are being prepared and the new fabrication techniques are to be tested on these specimens before proceeding with fabrication of the reboilers. This will probably cause some delay in the delivery schedule of these units.

2. Control Rods

a. Zircaloy-2 Follower Control Rods - A new set of control rods is being fabricated. They will have boron-steel poison sections and Zircaloy-2 followers, rather than fueled followers as in the set now in use. Development work is in progress to overcome cracking in boron-steel blades at the spotwelds, and deterioration of rivets.

One of the Zircaloy-2 crosses is being corrosion tested to ascertain the quality of the spot welds. After approximately 96 hours at 307°C the loop heater broke and the cross was removed for examination. Some indications of corrosion near the spot welds was noted. This unexpected corrosion behavior is attributed to the fact that the welding procedure used deviated from that recommended. The result was a hotter weld which expelled more metal than desired. The cross has been placed in another loop for future corrosion testing.

Attempts are being made to find a heat treating facility to heat treat one of the boron poison crosses to determine if a solution anneal will alleviate the spot weld cracking problem. To date no facility has been found which meets the heat treat and size requirements.

b. Control Rod Drive Parts - The present EBWR control rod drives were manufactured to ANL design and specifications by AMF Atomics Division. These nine drives are of a rack-and-pinion type. Three parts of the present type are made of 17-4 PH type steel: the rack, the pinion shaft, and the seal shaft.

This type steel has recently become suspect as a result of some failures in certain other applications in which cracks had developed. These

cracks are believed, however, to be related to the highly hardened condition (45 R_C) of the metal produced by an aging temperature of 480°C. It is also now believed that this treatment causes internal stresses of sufficient magnitude to initiate cracking.

By changing the aging temperature to 590°C a lower hardness (35 R_C) and lower internal stress levels are obtained. In addition, a more carefully controlled manufacturing procedure has been established for 17-4 PH parts to be made in the future.

Meanwhile, existing parts of the present drives are being checked. The above mentioned parts are being removed from the reactor control rod drives and individually checked for cracks or defects in material by the following steps:

1. Zyglo (post-emulsification type)
2. Magnaglo (direct current)
3. Sonic test

These checks are made with chromium plate on components. The chromium plate is then removed from all parts electrolytically and they are then rechecked.

The seal shaft and pinion shaft are again chromium plated and inspection steps 1 through 3 repeated. Immediately following plating the parts are heated to 230°C for 8 hours.

By removing only two assemblies at a time the reactor downtime can be held to a minimum. Two control rod drive assemblies, No. 6 and No. 8, were submitted to this testing on July 5th. These represent the first in a series of periodic inspections based on the number of hours of reactor operation.

The Zyglo and sonic methods gave no indications of defects. The Magnaglo technique gave light surface indications on one part only, rack No. 8. Microscopic examination of these indications proved them to be harmless surface scratches or gouges in both horizontal and vertical directions.

Metallographic examination of shafts sectioned previously with similar indications proved them to be very shallow indentations and not detrimental to the service life of the part.

The two drive assemblies were returned to the EBWR on July 7th for reassembly into the reactor.

3. Reactor Operation

a. Hydrodynamic Studies - A series of tests have been completed on EBWR, during which the in-reactor instrumentation for steam carry-under and liquid carryover was thoroughly checked out, and preliminary data obtained on the reactor performance characteristics with the new riser installed. Thirty-two tests were run covering the following reactor operating conditions: reactor power 5-10-15-20 Mw, saturated hot water level above the riser of 3 and 12 in. (7.62 and 30.48 cm), pressures of 300 and 600 psi (20.4 and 40.8 atm). The upper and lower feedwater rings were used to vary the location of feedwater injection.

The two make-up water points (upper and lower feedwater rings) were used to study the effect of steam carryunder on the behavior of the system. A pressure of 300 psi (20.4 atm) was used to simulate high power density operation. The initial hot water level (as indicated by the water column) was varied to study the effect on steam carryunder and to obtain data on the difference between the water column indicated level and the true interface (void level within the reactor vessel).

The tests demonstrated the success of the in-reactor instrumentation and provided preliminary data on reactor performance. The following data were obtained:

(1) a series of differential pressure drops in the riser and downcomer from which steam volume fractions were computed,

(2) velocities in the downcomer from which the total recirculation flow rates were computed,

(3) amount of subcooling,

(4) interface height.

The quantity of steam carryunder was computed from a heat balance on the downcomer using the measured recirculation flow rates obtained from the impact meter and the subcooling.

Further tests will be necessary to verify the accuracy of the temperature measurements.

The measured recirculation flow rates are of the magnitude expected. Previous experiments had indicated that the recirculation rate increased with increasing power and decreasing pressure. In the present experiments, this rate of change of recirculation rate was lower than estimated. The amount of steam carried into the downcomer became an appreciable fraction of the total generated. At 600 psi (40.4 atm), with the make-up water injected through the upper feedwater ring, this fraction was about 25%. The limiting parameter on power generation may well be steam entrainment in the downcomer.

b. Water Level Determination - An unexplained phenomenon in reactor water level measurement has been observed during low reactor pressure operation at the high water levels required with the full riser. While attempting to reach 20 Mw at 300 psi (20.2 atm) the feedwater control became erratic. The feedwater was taken off automatic and placed on manual control; the water level in the sight gauge remained erratic. Shortly thereafter the apparent water level started to rise quickly and though the feedwater was cut back, the level continued to rise until the reactor scrammed due to "High Reactor Water Level."

Tests were run at 400 psia and 500 psia which showed that if the level was raised slowly, the apparent level started to rise quickly after a certain level had been reached. This rise continued until the "High Reactor Water Level" interlock scrammed the reactor. Although the phenomenon is not yet fully understood, one possibility appears to be that there is a self-siphoning water trap built into the vent line. This line has been revised and work on the problem will continue.

c. Soluble Poison System - The piping has been slightly modified in the purification system to provide a positive pressure throughout the filter-ion exchange units-continuous monitor system under operating as well as shutdown conditions. Only the No. 1 pump-heat exchanger system was modified at this time. The No. 2 system will be modified when convenient.

Difficulty has been experienced with high temperature water effluent from the purification system subcoolers. The efficiency of the subcoolers had decreased to such an extent that the water flow rate had to be decreased to about 25% of its original value. The inefficiency of the subcoolers was traced to sludge and/or scale on the shell side. This is a typical condition throughout the cooling system of the EBWR plant. Chemical cleaning of two heat exchangers has been only partially successful.

It is desirable to increase the flow rate through the system for use with boric acid for Core I-A. It would take at least 110 hours at the present flow rate to reduce the boric acid concentration from 1.85 gm/liter to 0.025 gm/liter. At a normal flow rate of 32 liters/min only 31 hours would be required to accomplish the same removal. Other methods of cleaning the subcoolers will be attempted.

B. BORAX-V

1. Installation of Reactor and Components

Brackets to hold three differential-pressure probes, one Staucheibe tube and two reactor water sampling lines in the downcomer area have been welded inside the reactor vessel. A sampling line, a sample injection sparger and 8 core-structure guide dowels have also been installed in the

bottom of the vessel. This completes the installation of permanent components in the reactor vessel. Reassembly of the boiling core structure has been completed.

Fabrication and installation of the alternate auxiliary water system preheat piping to be used during natural circulation operation, and welding of the superheated steam line drains in the lower reactor pit were finished. The new reactor water demineralizer tank, filters, and associated piping were installed in the water storage pit. Anchoring of the boiling fuel rod and boiling fuel assembly storage racks completed the installation of permanent equipment in this pit.

A preliminary test installation was made of the control rod drives, seal housings, and motor drive assemblies to fix orientation. Then the drives and motor assemblies were removed to facilitate installation of seal water supply and leakoff tubing and blowdown piping. Based on calculated control rod worths, the motor drive assemblies are being modified to accept new change gears required to give much slower drive speeds.

Installation of large steel storage boxes with drains was completed for the storage of radioactive control rods, hold-down boxes and chimneys in the old reactor pit. The pit was back-filled with sand and capped with a poured concrete slab to reduce radiation from the old reactor vessel. This completes the installation of permanent components in the old reactor pit.

The interior of the existing makeup-water storage tank was sandblasted and coated with Phenoline paint. The polishing demineralizer was connected to the storage tank and charged with resin. The complete makeup-water system is now operable.

The prefabricated water sampling panel and hood was set in position and sampling lines, cooling water, drains, instrument cable, etc. are being connected.

The reactor pit working platform and access ladder have been fabricated and installed. Permanent bolting has been installed on the reactor vessel main flange. Overhauling of the turbogenerator and associated equipment was completed.

All detectors, amplifiers and preamplifiers in the nuclear instrument system are in place. Cables must be installed between the instrument room and reactor pit (for loading detectors), and from the instrument room to terminal cabinet No. 5, before checkout begins.

Wiring between rod drive brake relays and the rod drive control cabinet is now completed. Cables for the radiation monitoring system are now being installed between the reactor and turbine buildings, and cable

for test thermocouples in the access shaft is also being installed. Replacement wiring (chromel-alumel) has been installed for the main steam temperature thermocouple. Wiring for the forced-convection system reactor water temperature differential interlock was completed between the temperature control cabinet and the instrument room.

All transmitters in the process control and instrumentation system requiring relocation and repiping have been remounted, and are now awaiting final welding of the connecting lines. A new level transmitter is being sent to replace a faulty unit. The superheater-fuel-temperature-indicator scanning alarm and reactor vessel temperature recorder systems are now operating. The two main steam back pressure valves and the main feedwater control valve have been modified to fail-close on loss of air and electrical signal. All control valve limit switches have been set and the indicator lights now operate correctly. Checkout is proceeding on the control-rod-position light circuits and the annunciators. All alarms and scrams originating in the Control Building have been set. Checkout of circuits on busses CP-2, 3 and 4 is now about 95% complete.

2. Procurement and Fabrication

a. Boiling Fuel Rods - The entire lot of 3,543 boiling fuel rods has been inspected. Ten rods were rejected and the remainder are acceptable for use.

b. Superheat Fuel Elements - In accordance with contractual agreement, AI has fabricated a second group of 24 stainless steel-uranium oxide dispersion plates for ANL evaluation. Six each of types HCD and FPD plates were evaluated as reported in June. After discussion of evaluation results with personnel from AI, it was agreed that ANL specifications would be revised to increase allowable variation in core homogeneity and maximum length of UO_2 stringers. Based upon the revised specifications, ANL has accepted these 12 plates. It is understood that other discrepancies noted in these 12 plates will be corrected in subsequent plates. Six each of types HPD and FCD plates are now en route to the Laboratory and will be likewise evaluated upon arrival. Shipment of 80 FCD and HCD development plates from the supplier is promised by August 3.

Fabrication of brazing and assembly fixtures and nonfueled components for the superheater fuel assemblies is proceeding.

3. Reactor Components

All 35 of the high-temperature, tantalum-sheathed W/W-26% Re thermocouples, plus 8 calibration samples, for the boiling fuel thermocouple rods have been received from the vendor. Preliminary work on assembling these thermocouples into the rods has been started.

The remainder of the thermocouples for the instrumented superheater fuel assembly, and for measuring the in-vessel steam and water temperatures have been delivered. Fabrication has started on the coolant thermocouple "rake" for the instrumented boiling fuel assembly and the steam, water and reactor vessel specimen thermocouple assembly. The remaining two exit, and two entrance turbine-type flow meters for the instrumented boiling fuel assemblies are still being fabricated by the vendor.

Final assembly of the boiling fuel assembly boxes has started. The second batch of nine stainless steel clad Boral control rod sections is now being autoclaved at 600 psig saturated conditions, and four additional control rod sections are being fabricated. Fabrication has started on the instrumented boiling fuel assembly, instrument hold-down boxes, and chimneys. Work continues on the dummy fuel assemblies, the boron-stainless steel poison rods, the void and water-filled rods, and the steam dryer. The stainless steel poison rods were finished and are ready for use in the critical experiments.

The neutron source handling tools and the long handled wrenches for the in-vessel coupling clamps have been fabricated. Fabrication was completed on the blow-down filter which will remove radioactive particulates from the reactor vessel and control rod drive seal housings blowdown effluent. Work on the control rod channel orifice plates continued.

The stainless steel reactor vessel extension spool and the additional depleted uranium ring for the fuel handling coffin have been received.

4. Design

Detailed design was completed on the instrumented superheater fuel assemblies and associated instrument leads and pressurized terminal boxes; the special reactor-vessel-nozzle flange for pressurized terminal boxes; brackets for in-vessel downcomer instrumentation; a long-handled right angle drive socket wrench for in-vessel couplings; and modifications on the motor drive assembly for the control rod drives. Detailed design was also completed on the apparatus for the subcritical BORAX-V-type assembly which will be installed in the thermal column of AFSR and used to check the Cd-ratio method of void distribution measurement. Detailed design work continued on the fuel handling coffin, layout of the upper reactor pit and the duplex flux-wire counting machine.

Detailed design was started on the differential pressure probes for the measurement of downcomer water-steam mixture density. Preliminary design work was started on a rotating oscillating rod and drive, blowdown-filter system, and a superheater fuel assembly storage rack. Preliminary design has also started on two types of flow transducers proposed for downcomer measurements. One unit utilizes the velocity head developed by flow

around a submerged cylinder, and will require and external D/P cell. The second unit utilizes a drag disk and magnetic system for direct flow transduction to an electrical system.

The study of the effect of heat transfer on reactor kinetics problems has been concluded. Calculations of control rod worths under hot operating conditions have been started. Methods of decreasing control rod worth to prevent achieving criticality with one rod are being considered and analyzed. Calculation of the critical mass of a BORAX-V fuel rod assembly in ZPR-VII have been completed and indicate a cold, clean, cylindrical loading of about 576-645 of the 5% enriched rods, depending on the self-shielding factor assumed. Comparison with experimental results will be made.

Results of the problem on temperature distribution in a superheated steam outlet nozzle indicate that no excessive temperatures are caused by thermal radiation heat transfer from 850°F steam. However, the results indicate a need for additional investigation for steam temperatures above 850°F.

The decision has been made to add another fission product monitor for simultaneous monitoring of superheated steam and saturated steam from the steam dome during operation with a superheating core.

5. Development and Testing

a. Control Rods - Nine stainless-steel clad Boral sections of the reference control rods were autoclaved for two weeks in 600 psig, 489°F water. Numerous blade thickness measurements were taken before and after the test. Seven of the rods were found to be satisfactory with only an occasional few mils decrease in thickness, presumably due to the external pressure. One cruciform section had swollen enough at one point to tear out a spot weld and had apparently exploded from internal pressure. Another cruciform rod had an increase in blade thickness of about 0.035 in., but had no visible leak. Examination of these defective rods to determine the cause of failure is proceeding.

b. In-Core Instrumentation - Development work is proceeding on the pressure seal portion of the pressurized terminal boxes for the instrumented fuel assembly leads and the installation of the high-temperature thermocouples in the boiling fuel thermocouple rods using NIORA brazine alloy. Microscopic examination of samples of tantalum sheathed thermocouples has revealed the presence of surface cracks apparently due to tantalum embrittlement by gas adsorption during high-temperature fabrication or testing. A method of non-destructively testing the tantalum-sheathed portions of the thermocouples is being sought. Bids have been requested for calibration of 8 sample tantalum sheathed W/W-26% Re thermocouples at temperatures up to 4200°F or higher in a vacuum or inert gas atmosphere which will not embrittle the tantalum.

6. BORAX-V Superheater Critical Experiment

Draft copies of hazards studies for a proposed critical experiment with BORAX-V superheater fuel subassmblies are being reviewed. The experiment will produce information on flux and power distributions; reactivity coefficients of void, fuel, poison, and temperature; and spectral effects. The experimental cores will include twelve BORAX-V superheater subassemblies plus sufficiently large UO_2 fuel zones to attain criticality. These cores may be visualized as being part of the High Conversion Critical Experiment but with part of the 3 wt-% or 5 wt-% enriched UO_2 fuel zone replaced by highly enriched superheater elements.

During voided runs, the superheater fuel will be insulated from coolant and moderator. This condition reduces the prompt negative reactivity coefficient of the cores involved; consequently, the 3 or 5 wt-% enrichment fuel zones must provide the shutdown coefficients. The uncooled fuel may melt during an accident but would not if it were in contact with coolant. For this reason a favorable shutdown coefficient is quite important.

The hazard of accidental flooding into a voided superheater is believed to be controllable by a system of compartmentation and double water barriers. The voided region will be divided into 12 regions which will not produce prompt critical conditions if any one should flood. The total void worth controlled by all 12 elements is estimated at 4 or 5% in reactivity.

7. Training Program

Editing and revising of the BORAX-V Operating Manual continued. All prospective Operations Supervisors and Operators have completed the BORAX-V Operators Training Course.

II. SODIUM COOLED REACTORS (040103)

A. General Research and Development

1. ZPR-III

a. Experimental - Work was started and completed on Assembly 37 during the month of July. This assembly was constructed to measure the gamma dose rate, neutron flux, and core temperature buildup as a function of time for a subcritical assembly fueled with plutonium.

Each core drawer contained two columns of plutonium, three columns of depleted uranium and eleven columns of aluminum. The eleven columns of aluminum were divided into six columns of 63% Al and five columns of 45% Al. An attempt was made to duplicate as nearly as possible the core volume fractions of Assembly 2, a U-fueled critical assembly of simple construction. The dimensional and volume fraction parameters are given for this assembly in Table I.

Table I. Physical Parameters of Assembly 37

Mass of Plutonium (kg)		20.05
Plutonium Analysis (%)	Pu^{239}	95
	Pu^{240}	4.5
	Pu^{241}	0.5
Core Length (in.)		8
(cm)		20.3
Core Radius (in.)		7.68
(cm)		19.5
Core Volume Fractions		
Plutonium		0.04
Depleted Uranium		0.15
Stainless Steel		0.12
Aluminum		0.31

Although this assembly had an effective k of approximately 0.4, it was loaded as if it were a critical assembly. After each small addition of core material, subcritical counts were taken and plotted as inverse count rate versus total mass of plutonium in the assembly. As loading proceeded, the subcritical multiplication curve was continuously observed to insure that the effective k was not far different from that anticipated.

After the assembly was completed, gamma dose rate measurements were made with both Juno-type survey meters and X-ray films. These measurements are given in Table II.

Table II. Gamma Dose Rate Measurements

Distance from Core (inches) (measured to front of detector)	Juno Shielding	Juno Reading (mr/hr)	X-ray Film	
			Radium Calibration	50 kvp Calibration
0	γ	66*	260	21
	$\beta\gamma$	69		
	$\alpha\beta\gamma$	71		
3	γ	42		
	$\beta\gamma$	43		
	$\alpha\beta\gamma$	44		
6	γ	32		
	$\beta\gamma$	33		
	$\alpha\beta\gamma$	34		
12	γ	16		
	$\beta\gamma$	18		
	$\alpha\beta\gamma$	21		
20	γ	9		
	$\beta\gamma$	11		
	$\alpha\beta\gamma$	12		
29	γ	6	7	0.5
	$\beta\gamma$	6		
	$\alpha\beta\gamma$	6		

*Surface measurements are not comparable.

The large discrepancies between readings could be due to the rapidly changing efficiency versus energy of these detection systems in the energy region of interest. The Juno-type survey meters are calibrated with a radium source. The X-ray films have been calibrated using both radium gamma rays and a 50 kvp X-ray machine. The average gamma energy for the gamma rays emitted from this core is probably somewhere between these two energies and may be in the region of 60-70 kev. The open-window Juno readings should not be in error by more than 30% if the effective gamma ray energy is of the order of 60-70 kev as anticipated.

The neutron flux at various points between the assembly halves was measured using a lithium iodide crystal surrounded by polyethylene spheres of varying diameters, a Hanson-Long counter, an RCL fast neutron survey meter, and Eastman Kodak neutron monitoring films. The results of these measurements are given in Table III.

Table III. Neutron Flux Measurements

Distance from Core Face	Instrument	Reading
29 in. (midway between halves)	Modified "Long Counter"	70 n_f/cm^2 - sec
29 in.	RCL Fast Neutron Survey Meter	14.7 n_f/cm^2 - sec
At contact	RCL Fast Neutron Survey Meter	42 n_f/cm^2 - sec
At contact	Neutron Films	<40 n_f/cm^2 - sec

The results of the lithium iodide measurements are not yet available. The wide variation in the results of the neutron measurements is felt to be indicative of calibration problems of the types of system. It is felt that a more rigorous evaluation of the data coupled with instrument recalibrations may shed some light on the large discrepancies.

Temperature measurements were made at 36 points within the core at various times with iron-constantan thermocouples to determine the temperature increase as a function of time.

Thermocouples were spaced to indicate a complete temperature profile through one plane of the cylindrical core. In one of these central drawers, the thermocouples were placed against the side of plutonium fuel plates to indicate the maximum temperature in the core. Two types of temperature buildup were studied. The first type was merely a measurement of the temperature increase in one half of the assembly followed to saturation temperature. The second measurement consisted of following the temperature buildup with 3 in. of Fiberglas insulation placed in front of the assembly. The presence of the Fiberglas insulation will cause heat flow from the front face of the assembly half to be at a minimum, duplicating to some extent the condition existing when both halves of the assembly are together. A typical temperature profile, and the temperature buildup data for the noninsulated condition and the insulated condition, are given in Tables IV, V, and VI.

Table IV. Radial Temperature Distribution at Core Face (with insulation)

Thermocouple No.	Temperature - (°C)	Radial Distance Along Core Face from Core Center (inches)
12	43.3	2.54
23	43.2	4.57
29	41.8	6.83
35	38.8	9.25

Table V. Temperature Measurement - Assembly 37

Temperature measurements for thermocouple No. 12 located under Pu fuel element, using complete Pu loading with no insulation against face. Thermocouple No. 12 located in center face of drawer No. 1Q-16.

Time of Measurement (hours)	Temp. (°C)	Room Temp. (°C)
0	33.0	22
29	36.0	26.5
99	36.9	29.5
100	38.1	30.0
121	39.0	25.5
122	39.0	26.0
124	39.0	27.0
126	39.2	27.0
127	39.4	28.3
144	37.7	24.0
146	37.6	25.3
149	38.1	27.0
151	37.8	27.0
168	37.8	25.0

Table VI. Temperature Measurement - Assembly 37

Temperature measurements for thermocouple No. 12 located under Pu fuel element in center drawer No. 1Q-16. Insulation against the matrix face.

Time of Measurement (hours)	Temp. (°C)	Room Temp. (°C)
0	38.7	28.0
1	39.3	28.0
2	39.5	28.0
19	40.9	25.7
24*	40.1	28.0
26	40.8	29.0
92	43.4	27.0
96	43.3	29.0
98	43.5	29.0
115	43.0	25.6

*Insulation removed for 3.5 hours to add additional thermocouples.

The gamma and neutron dose rates at several of the measured points were also computed by approximate methods derived from the literature on the fuel material. In the calculations, the core was considered as a homogeneous cylindrical source; the effects of the reflector and multiplication were not considered. The results are shown in Table VII.

Table VII. Calculated Gamma and Neutron Dose Rates for Assembly 37

Distance from Core	γ Dose (mr/hr)	Neutron Dose (mrem/hr)
0 in.	146	59
6	121	-
20	10.8	-
29	5.2	0.9

b. Analysis - Resonance scattering effects of aluminum and stainless steel were considered in new critical mass predictions for a few ZPR-III assemblies (Table VIII). The core group transport and elastic removal cross sections were determined by use of the ELMOE Code (see Progress Report, June, 1961, ANL 6387, p. 15). SNG_4 critical mass calculculations were then carried out using the 16-group Yiftah set with core transport and elastic removal cross sections modified. Corresponding critical masses which would be obtained if the "conservative" $\nu^{U^{235}}$ values referred to by Yiftah are used are also listed. Recent measurements of ν^{25} by A. B. Smith (see Progress Report, May 1961, ANL-6374, and Section IV A 1 of this report) suggest that the lower $(d\nu/dE)^{U^{235}}$ may indeed be more valid. Comparison of the resonance-corrected, conservative ν^{25}, calculated masses with the experimental critical masses show that:

(1) the aluminum diluent assembly, No. 23, is now calculated of the order of 1% k overly reactive,

(2) the stainless steel diluent assembly, No. 32, still is computed overly reactive by about 2.7% k;

(3) the stainless steel-aluminum diluent assembly, No. 31, is overly reactive by about 2.1% k.

Table VIII. Critical Masses (kgs U^{235} in Core)

	Experimental		Calculated*			
			Regular ν^{25}		Conservative ν^{25}	
ZPR-III Assembly No.	Measured	Corrected	Without Resonance Corrected	With Resonance Corrected	Without Resonance Corrected	With Resonance Corrected
23	258	271	219	233	241	257
31	463	494	387	407	414	436
32	227.5	239	176	197	188	210

*Assumes shape factors of 0.94 for sphere to cylinder.

It thus appears that the stainless steel, 16-group cross sections are still too reactive (by about 1 or 2% k in assembly No. 32). An increase in the average stainless steel capture cross section by about 40% in the energy groups between 9 kev and 1.35 Mev would be necessary to reduce the assembly No. 32 reactivity by about 1% k.

2. ZPR-VI and ZPR-IX

a. Control and Safety Rods - A contract for the fabrication of 24 blade-type safety rod drive mechanisms has been placed. Delivery is expected to be started in late autumn.

b. Control Console and Instrumentation - The control console layout drawing has been received from the vendor for approval. With some modifications it should be possible to accept his layout. The schematic interlock circuit diagrams were also received from the vendor for approval. These drawings are now being carefully reviewed.

c. Procurement - All of the depleted uranium blanket slugs and core dilution plates have been received. Coating of the depleted uranium with a teflon material is in progress.

All of the control and safety rod drive mounting plates and support members for ZPR-IX have been received and inspected.

d. Building - The cycle of pressurizing, strain measuring, leak hunting, and then sealing continues on cell No. 5. The last test reached a pressure of 5 psig ($\frac{1}{3}$ atm). At this loading the wall deformation measured with six strain gauges was consistent with the design calculations, but new leaks were found around the door frames at the concrete interface. Humi-Seal epoxy resin was pumped into voids in the bond between the door frames and concrete wall and this treatment stopped most of the leaks.

Concurrently, experience with cell No. 5 is being used to ready cell No. 4 for its tests. Standard grease fittings are being installed on the door frames now so that epoxy seal can be injected without delay if leaks are found.

e. Building Leak Rate Tests - All strain gauges were applied on the surfaces to be tested. This involves the 44 active gauges and 44 compensating. All outside gauges were insulated and weather proofed, and all inside gauges were moisture proofed with beeswax.

To detect any discrepancies in the readings due to temperature variations between active and compensating gauges, the west wall (showing the largest temperature changes) was thermocoupled and the temperature

variation recorded over a 29-hour continuous period. Temperature readings were taken every half hour and the results were plotted. The largest discrepancy was observed between the reinforcement to be instrumented and the slugs provided for the compensators. A maximum possible error of 7 microinches/in. was observed.

Such a large error cannot be tolerated, since it amounts to more than 10% of the maximum anticipated strains. Thus, the slugs for the placement of the compensators were abandoned. The compensators were mounted instead on semicircular plates loosely attached to the reinforcing bars to insure proper heat conduction. Also, it was decided to run the tests between 11 p.m. and 6 a.m. in order to reduce temperature effects to a minimum.

The shell was pressurized twice (on different days) to 5 psi to check the compressor, the pressure gauges, the water column, and other devices. A number of strain gauges that were operative at the time of pressurization showed reasonable agreement between experimental and calculated strains at the seam between two pours and did not record unusual or unexpected strains. The final pressurization test is expected to be performed after the repair of some heavy leaks and modification of the pressurizing equipment.

B. EBR-I

1. Fabrication of Core IV Fuel Elements

Work was completed on depleted uranium filled blanket rods. A total of 121 acceptable blanket rods were produced by the same techniques to be used for the fuel elements. To date, 125 fuel rods have been loaded with plutonium-1.25% aluminum fuel alloy, NaK filled and bonded. Of these, 40 have been accepted, 5 have been rejected for weld leaks, and the remaining 80 are in various stages of inspection. The weld difficulties were traced to an excessively tight fit of the connector in the tube which caused pressure to build up behind the welds. The connectors were reworked to vent the volume below the weld and the welding problem appears to be solved.

Injection casting of fuel slug material has been resumed to provide a sufficient number of slugs for completion of the order. There has been unexpected difficulty in producing the ribs by the upsetting operation. This may be due to the necessity of rolling the cast slugs to a smaller size which produces an increased surface hardness. A slight modification of the upsetting procedure has resulted in some improvement of this operation.

2. Properties of Plutonium - Aluminum Alloys

Slump tests on the plutonium-1.25 w/o aluminum fuel alloy for the Mark-IV core have been completed. These tests were designed to determine the slump behavior of this alloy on repeated heating and cooling in the temperature range 400-640°C, under an axially applied load of 20.3 psi.

The test results on specimens measuring 0.205 in. in diameter by 1.5 in. long can be summarized as follows:

1. An homogenizing heat treatment, carried out under vacuum at 450°C for 145 hours, greatly improves the load carrying ability of the alloy. Following such a heat treatment specimen SB-2 showed no tendency to slump below 500°C.

2. The benefits of such a heat treatment are retained at 550°C for test periods up to 20 hours.

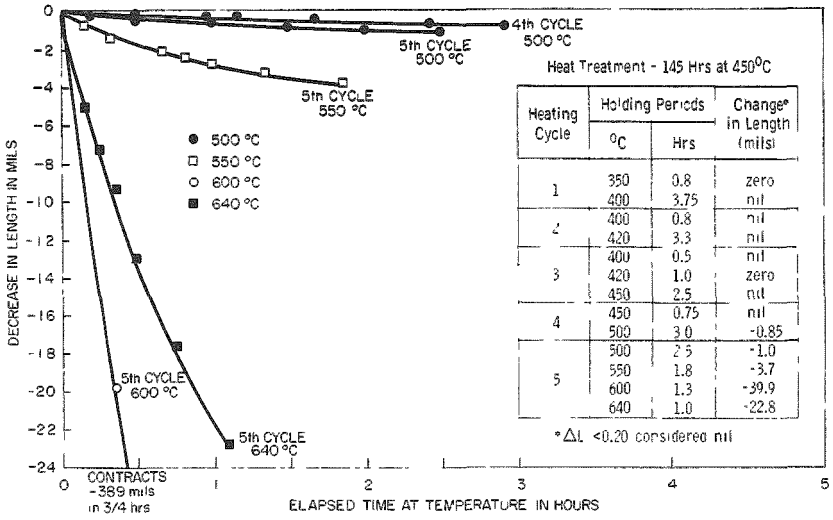
3. Approximately the same benefits can accrue from a three hour homogenization at 550°C.

4. With either homogenizing heat treatment the amount of slump occurring at any temperature in the range 400-550°C can be expected to decrease with each successive heating cycle.

5. Above 550°C the amount of slump increases by a factor of seven or more for all specimens regardless of heat treatment.

6. At no time did any specimen collapse suddenly and without warning even at temperatures up to 640°C.

The benefits of a long and an abbreviated homogenizing heat treatment are illustrated graphically in Figures 1 and 2.

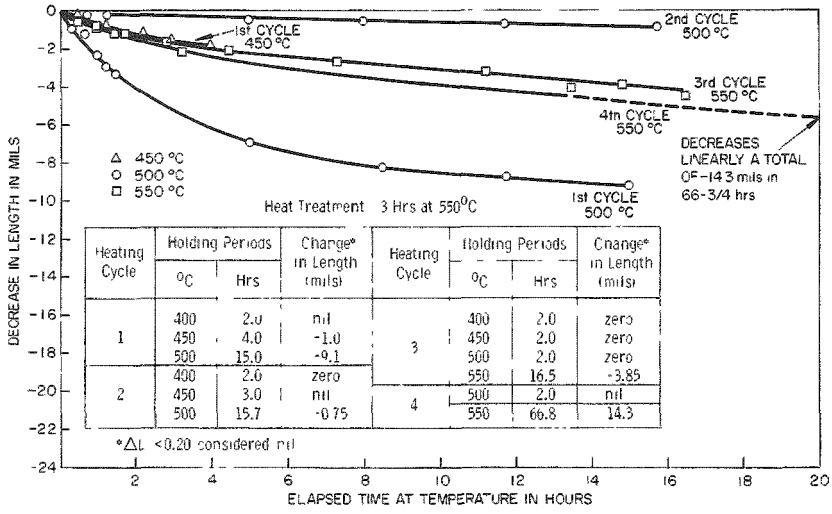


Heating Cycle	Holding Periods °C	Holding Periods Hrs	Change* in Length (mils)
1	350	0.8	zero
	400	3.75	nil
2	400	0.8	nil
	420	3.3	nil
3	400	0.5	nil
	420	1.0	zero
	450	2.5	nil
4	450	0.75	nil
	500	3.0	-0.85
5	500	2.5	-1.0
	550	1.8	-3.7
	600	1.3	-39.9
	640	1.0	-22.8

Figure 1

Decrease in Length of Specimen SB-2 with Time at Temperature

Specimen Load, 20.3 psi



Heating Cycle	Holding Periods °C	Holding Periods Hrs	Change* in Length (mils)
1	400	2.0	nil
	450	4.0	-1.0
	500	15.0	-9.1
2	400	2.0	zero
	450	3.0	nil
	500	15.7	-0.75
3	400	2.0	zero
	450	2.0	zero
	500	2.0	zero
	550	16.5	-3.85
4	500	2.0	nil
	550	66.8	14.3

Figure 2

Decrease in Length of Specimen UH-1 with Time at Temperature

Specimen Load, 20.3 psi

C. EBR-II

1. Construction

The approximate status of construction contracts as of July 18, 1961, was as follows:

Building	% Completion
Power Plant (Package 2)	100
Reactor Plant (Package 2)	100
Fuel Cycle Facility	93

2. Installation of Equipment - Package 4

a. Cleaning Operations - Cleaning of the secondary sodium system has started. The cleaning procedure consists of three major operations: 1. cutting open the system; 2. removing the foreign material from the piping; and, 3. welding the system closed. The pipe cutting operation is about 40 percent complete. The removal of foreign material has been initiated. Welding has not begun.

The pipe is opened with a cutting tool which faces simultaneously both sides of the cut with a $37\frac{1}{2}^{\circ}$ welding bevel, and removes only that material necessary for proper fit-up for rewelding. By this procedure no additional pipe material is required for reassembly of the system and fit-up of the piping for welding is facilitated. The use of a cutting tool rather than a saw, slitting wheel or cutting torch reduces to a minimum the quantity of foreign material introduced into the piping by this operation.

A possible cleaning technique was discussed in the Progress Report for June (ANL-6387) which utilizes standard pipe cleaning "pigs." This method is being employed. The small pig shown in Figure 2 of the June Report is used; however, the standard unit shown in Figure 2 is slightly modified. It is shortened for easier removal from the piping and the steel nose is replaced by a nylon one to avoid damage to the pipe wall.

The modified units have been used to clean one of the two evaporator sodium outlet headers. Figure 3 shows the header after it has been opened for cleaning. This line is a typical example of the type of piping configuration to be cleaned. The straight 10 in. portion of the header is cleaned by pulling a pig through the line. The four "S" shaped 8 in. lines, which are open at one end and attached to the 10 in. header at the other, are cleaned by propelling the pig through the line with air pressure. To do this, the 8 in. pig is loaded into the open end of one of the 8 in. lines followed by the pipe plug which effects a temporary pipe closure as shown in Figure 3. The enclosed volume between the pipe plug and pig is pressurized with air via an opening through the pipe plug. At 20 to 30 psig air

pressure the pig is propelled through the line and is stopped by the 10 in. header. The nylon nose protects the header from damage. The unit is withdrawn through the 10 in. header by pushing it through the line with a second pig which is manually manipulated.

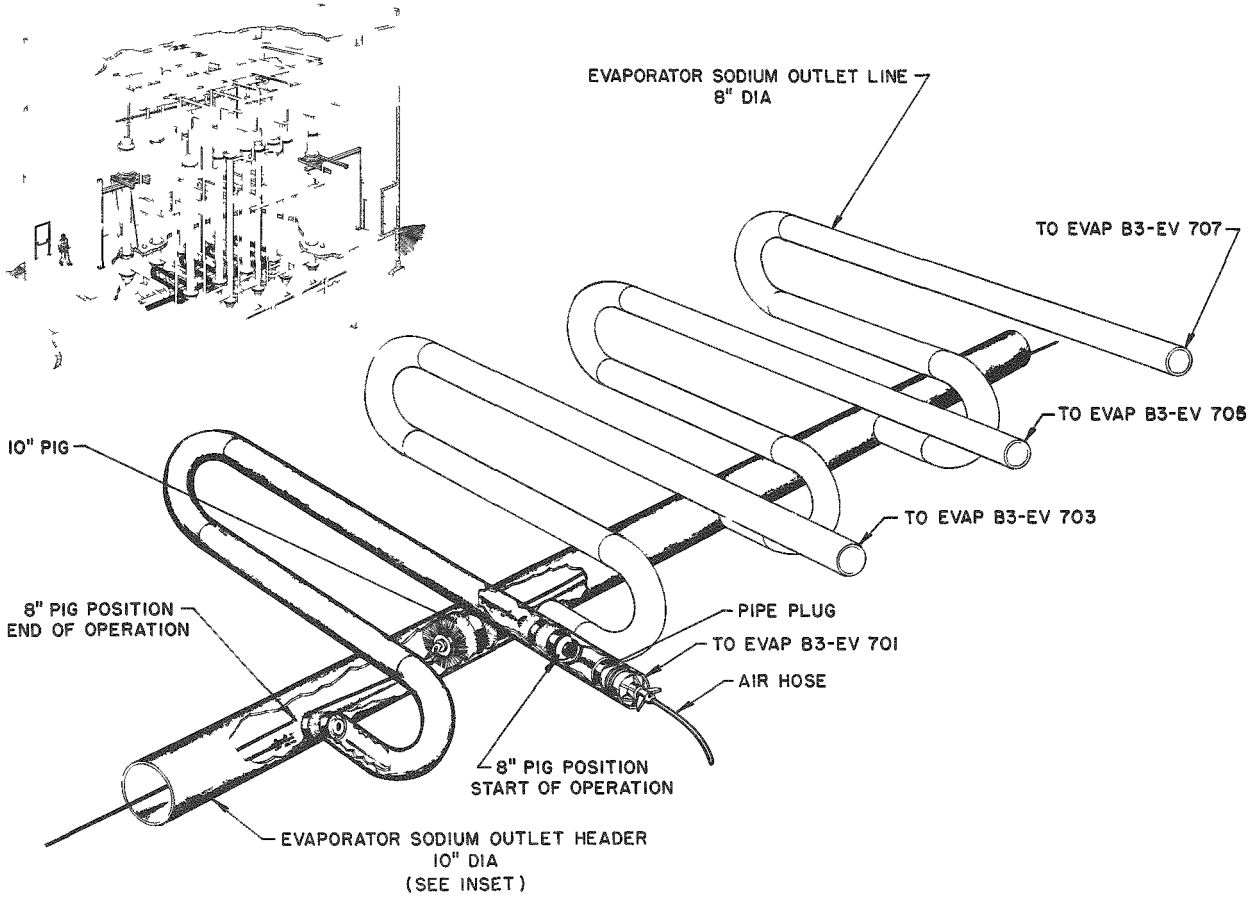


Figure 3

Cleaning Operation Evaporator Sodium Outlet Header

The cleaning procedure which is being used is as follows:

1 The line is opened and made accessible for cleaning. For example, the 10 in. header noted above is cut open at each end, the four 8 in. evaporator sodium outlet lines are cut free from the evaporator, and the header is moved to provide access to the pipe openings. The piping configuration shown in Figure 3 is then ready for cleaning.

2. The interior of the pipe is visually inspected through all openings. Any foreign material near an opening is recorded, photographed and removed.

3. The pipe cleaning pig is put through the line a minimum of three times. If at this time there is evidence that foreign material still exists in the pipe additional passes are made until such evidence is removed.

4. The pipe is wiped to remove loose, fine, particulate matter. The wiping action is accomplished by replacing the wire brush on the pig with cloth material.

5. The lines are blown clear with air and sealed closed to preserve the state of cleanliness.

6. All material removed from the system is carefully photographed, recorded and retained.

7. The piping is reassembled, rewelded and stress relieved, if necessary. During this operation the piping system is flooded with inert gas to facilitate welding and to prevent scaling of the pipe.

Figures 4 and 5 show the condition of the interior of two of the four evaporator sodium outlet nozzles prior to cleaning.

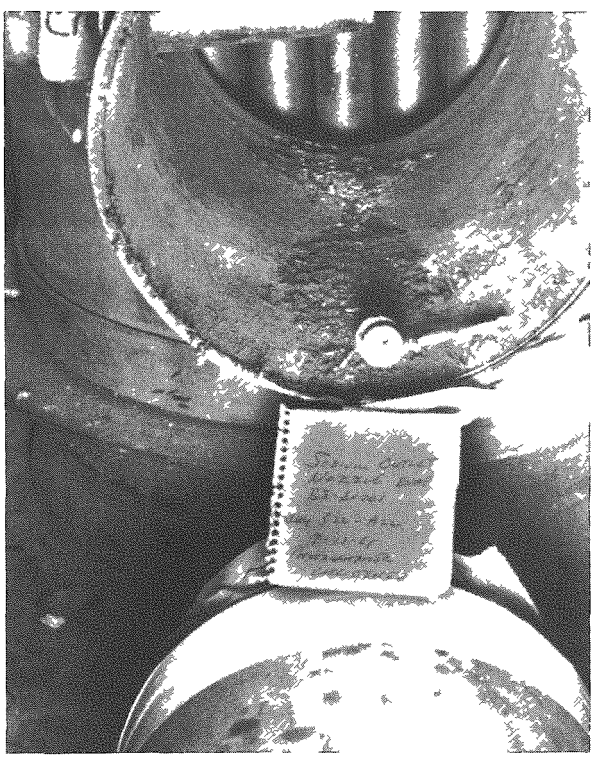


Figure 4

Interior of One of the Four Sodium Outlet Nozzles Showing Scale and Nut-Bolt-Washer-Wire Assembly. For X-ray of This Foreign Material Refer to Figure 3 page 20 of April Report (ANL-6355).

Figure 5

Interior of Another of the Four Evaporator Sodium Outlet Showing Scale, Ashes, and Foreign Material.

Note that in each instance the line is contaminated with a nut-bolt-washer-wire assembly which is laying in or near a heap of ashes and/or scale. Each of these units was a temporary pipe plug used by the construction contractor. The plug was used to effect a temporary blockage of the evaporator inlet in order to maintain an inert gas atmosphere within the pipe for welding the pipe to the evaporator nozzle. It is believed that removal of these plugs was intended to be via the 2 in. thermocouple well nozzle shown. (The thermocouple well was not in place at that time.) To accomplish this, a very flexible type plug would be necessary to be removed through a 2 in. opening. Difficulty at this stage could easily cause the loss of a plug. In Figure 3 of the Progress Report for April (ANL-6355) an X-ray photograph of a nut-bolt-washer-wire assembly was shown. The object in Figure 4 is that assembly.

b. Identification of Piping Material - Concern has existed that material other than the specified 2.25 w/o Cr-1.0 w/o Mo low carbon steel had been accidentally installed in the piping system. Unfortunately the situation was vague in that it was not known whether, 1) another material had definitely been installed, and 2) whether the sodium and/or the steam piping was involved. Accordingly, both systems were suspect. To minimize damage to the existing systems by removal of samples for chemical analysis, nondestructive techniques have been developed for identification purposes. During the present period, all the sodium piping of 4 in. diameter and larger has been checked with a thermoelectric device supplemented by spark testing. The thermoelectric device was developed by the Laboratory for this specific problem. After nondestructive checking was complete, a small number of samples were taken for chemical analysis. These indicate that one length of pipe was slightly out of specification, and that some of the welding rod used is suspect. Additional sampling of the welds is planned.

c. Containment Vessel - As described in the preceding Progress Report (ANL-6387), water seepage had been discovered from a localized area of the concrete wall lining the inside of the containment vessel. The area involved was located on the southeast side of the building immediately below the sub-basement ceiling. The concrete in this area had been removed to accelerate drainage of the remaining water trapped in the thin gap between the wall and the vessel. Although it was considered extremely unlikely that this water could have entered the building by passing through the containment vessel from the outside, a precautionary test was thought desirable for substantiation. This test is now essentially completed.

A curved trench about two feet wide and two feet deep was dug immediately outside the containment vessel in the southeast quadrant as shown in Figure 6.

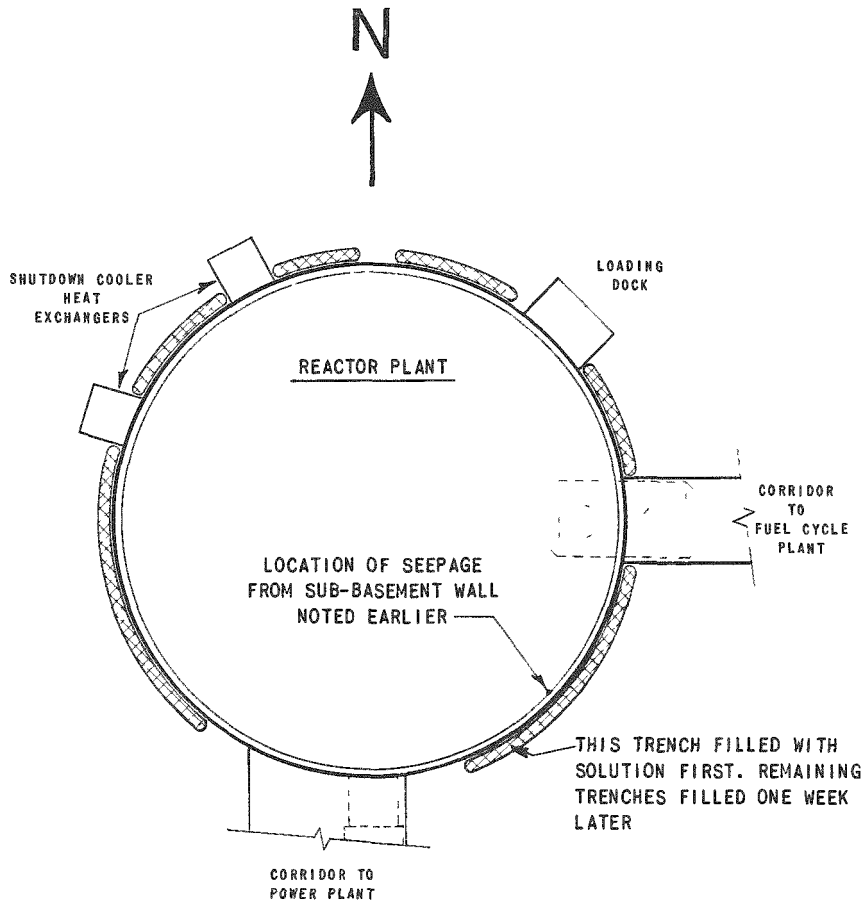


Figure 6

Trench Layout Employed in Fluorescein Solution Testing of Containment Vessel.

This trench, which encompasses the seepage area location, was filled with approximately 1200 gallons of water bearing 50-100 ppm of fluorescein. For a period of three days, additional solution was added to make up for that absorbed by the earth and maintain the trench filled. After one week with no new leakage and no trace of fluorescein discernible at the seepage area, additional trenches were dug surrounding the containment vessel as shown in the figure. These trenches, with the original trench, constituted an almost continuous ring around the entire vessel; only those areas which were inaccessible, such as at the Power Plant and Fuel Cycle Plant corridors, loading dock, etc., were not trenched. These newer trenches were initially filled with approximately 6000 gallons of fluorescein solution and then were maintained filled over a three day period. No new seepage and no trace of fluorescein at the seepage area inside the Reactor Plant has been found to date (three weeks from commencement of the test). The trenches are now to be refilled with earth and the test considered concluded, except that additional checks for fluorescein at the seepage area will be made for another week or two.

d. Leak Rate Test - A leak rate test was performed on the Package III isolation valve in the suspect exhaust line. After a 72-hour, 24-psi test, loss of pressure was only 0.2 psi and the valve and line to the building are considered leak-tight.

3. Preparation for Dry Critical Experiments

Preparations for the dry critical experiments by Laboratory personnel have been continued through the present period and are now close to completion. As in the preceding period, the work of this period consisted primarily of minor corrective actions and modifications, checking-out of the mechanical equipment and electrical circuitry, final alignments and adjustments of components and equipment, etc. The current status of this work is very briefly reviewed below:

The final checkout of the entire fuel handling system is approaching completion. A series of transfers of subassemblies between the storage rack and various positions within the reactor has been accomplished with no significant difficulty, although many minor adjustments and modifications were found to be necessary. With the exception of a small amount of rework still to be done on the transfer arm mechanism, there remains only a further series of transfers of subassemblies into and out of the reactor to be accomplished successfully before the fuel handling system may be considered ready for dry critical experiments.

Some difficulty has been experienced in obtaining smooth operation of the transfer arm mechanism. Intermittently "sticking" of the arm has occurred, apparently because of a faulty bearing. This is considered minor, because the transfer arm is manually driven and is readily disassembled for inspection. The arm will be disassembled and the difficulty corrected prior to starting dry critical experiments.

It has been found that several of the subassembly tubes in the storage basket were larger than called for in the design. As a result the control rods and safety rods fit too loosely in the storage basket tubes and are not held as close to the vertical position as is required. Since the storage basket is not required to serve its normal function during the dry critical experiments, modification to the basket will not be made until a later date.

Preparations have been under way for introduction of the neutron source into the reactor. One neutron source thimble and one neutron source shield thimble were inserted in their proper positions in the reactor. All dry critical, in-core instrumentation has been installed. Practice runs, using a dummy neutron source, have been completed to check out the procedure for introducing sources into the reactor from the temporary source storage holes in the operating floor of the Reactor Plant.

A minor modification to the lower adaptor of each of the twelve control rods was made to eliminate difficulty initially encountered with operation of the control rod gripper jaw drive. The modification consisted of lengthening the lower adaptor seating pin by approximately $\frac{3}{16}$ in. One control rod drive was tested in place, using a dummy control rod, to determine the scram time-displacement characteristic. Its performance was completely satisfactory.

Although the dummy loading of the reactor was completed in the preceding period, no photographs of the loading were available at that time. As a matter of general information, Figure 7 shows the reactor with a portion of the outer blanket and the central (thermocoupled) subassembly loaded. Easily seen also are the twelve control rod drive shafts with grippers attached at their lower ends, the holddown mechanism, a portion of a reactor vessel cover lock mechanism (at the right of the photo), and the reactor coolant outlet nozzle.

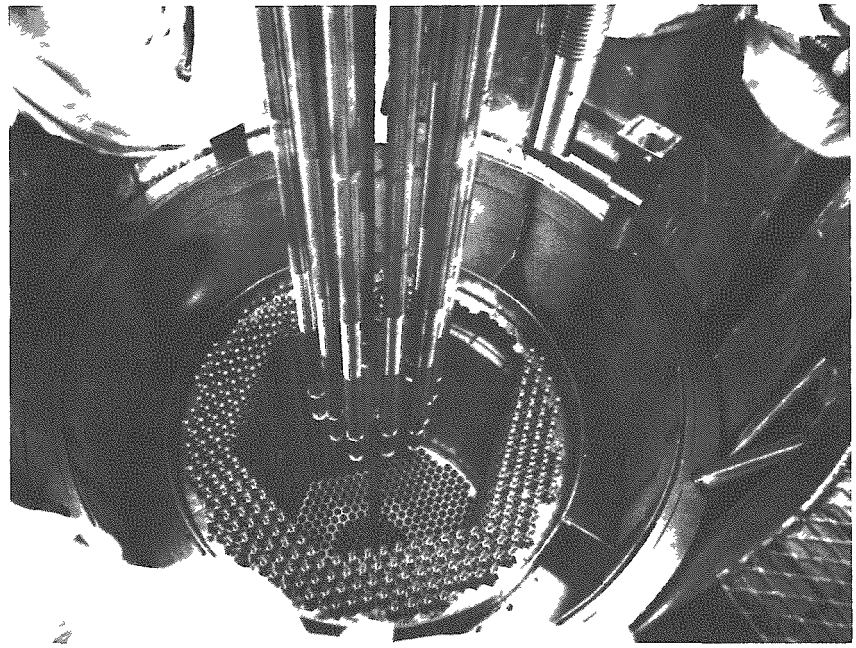


Figure 7

View of Reactor Core during Dummy Loading Showing the Outer Blanket and Central Thermocouple Subassemblies Partly Loaded.

Minor modifications were made to the control circuits as the result of a safety review. These modifications have been checked out and satisfactory operation of the system achieved.

A detailed checkout was also made of the shutdown system to be used for the dry critical experiment. The control functions initiated by

Channels A and B, Channels A and B level bypass, and Key Switch No. 6 (Reactor Startup Key) were installed. The system was operated and each control function checked to assure initiation of a scram.

A light panel and switch were installed in the control room to perform nonautomatic functions and indicate isolation valve positions. A valve-position indicator has also been installed on the instrument air valve to indicate position in the control room.

The automatic data logger has been programmed for those signals pertinent to the dry critical experiments.

Systems have been placed into operation for operator training, and reactor checkoff sheets have been reviewed.

Common-key switches on the operating console have been replaced with individually-keyed switches.

Reactor Technology course for new technicians is progressing satisfactorily and is about $\frac{1}{3}$ complete. Technician training on the operation of the electrical power system is continuing. A technician training program has been started to train technicians for the initial sodium tank car unloading and system filling. These trainees will ultimately be the sodium system operators.

4. Engineering

A final experimental flow test of an inner blanket subassembly has been completed. This test was required to verify the calculated flow rate versus pressure drop characteristic of the as-built subassemblies, which incorporated one final design change not previously tested. The change consisted of a modification of the flow restrictor strip geometry to increase coolant flow rate through the peripheral channels of the subassembly. The test was conducted in an instrumented water loop using an exact, full scale model of the subassembly, and the data converted analytically to the EBR-II sodium system. The results indicate that at full reactor coolant flow, the flow rate per subassembly to be expected in each row of the inner blanket is: sixth row, 30.1 gpm; seventh row, 20.3 gpm. These flow rates are very satisfactory. They compare with flow rates before the final modification of 22.8 and 16.5 gpm, and with predicted flow rates of 32.2 and 19.5 gpm, respectively.

5. Procurement

Fabrication of the sodium-to-sodium heat exchanger is complete and it is en route to the site.

Manufacture of about 825,000 lbs (ten tank cars) of sodium coolant is now complete. The chemical analyses of the sodium coolant have been received and are acceptable. The specified chemical composition of the sodium, and the results of the chemical analyses of a sample from each of the ten (10) tank cars as reported by the sodium manufacturer, are shown in Table IX. Delivery of the tank cars to the site is expected next month.

Table IX. Analyses of EBR-II Sodium Coolant

Element (ppm)	Specified (ppm max)	Tank Car GATX No. 67900	67902	67905	67906	67907	67908	67909	67978	67979	67981
Calcium	10	<5	<3	<5	<3	<5	<2	<2	5	8	<5
Carbon	50	20	21	5	10	16	26	16	17	29	12
Chlorides	30	2	5	5	8	11	22	13	3	3	13
Sulphur	10	<2	2	1	2	6	1	<1	<2	<2	<2
Boron	5	<1	<1	<1	<1	<1	2	2	2	2	2
Lithium	20	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Silver	20	<2	<2	<2	<2	<2	<2	<2	<2	<2	<2
Gold	30	<2	<2	<2	<2	<2	<2	<2	<2	<2	<2
Cadmium	20	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Indium	30	12	11	9	13	12	8	11	7	11	11
Potassium	1000	124	190	135	150	138	132	151	121	104	119
Alkalinity, as Na, %	99.97 ± 0.	100.01	99.98	99.98	99.98	99.97	100.00	99.97	99.95	99.99	99.99

6. Component Development - Instrumentation

All wiring revisions necessary for dry critical operation have been made. Final checkout of nuclear instrumentation, the control rod circuits, and the emergency reactor shutdown circuits, is being carried out.

a. Fuel Handling System - Field testing of the fuel handling control system has necessarily been integrated with concurrent mechanical testing of the mechanisms, and other activities such as the loading of blanket subassemblies. Some difficulty has been experienced with operation of the gripper force-limiting devices. Operation is satisfactory for dry critical experiments, but some modifications may be required after these experiments have been completed. Circuit modifications and wiring to accommodate the recently added rotating plug rotational locking devices are approximately 90% complete.

b. Cable Connectors - The 100-contact connectors used for connecting cables to the rotating plugs have in some cases failed to make reliable contact. This is at least partly due to the weakness of the aluminum connector shells. Replacement shells made of steel have been ordered, and will be installed as time permits.

c. Refueling Machine - The refueling machine, except for its associated carriage, has been delivered to Argonne, Illinois. The control circuit design is being modified to accommodate the revised cooling system, port seal, and port seal purging scheme.

d. Fuel Element Failure Detector - A continuous monitoring station for EBR-II fuel rupture has been under development for about one year. After the decision was made to base the detection scheme on delayed neutron monitoring, a neutron source with appropriate gamma background was constructed for testing purposes by insertion of a long source containing irradiated sodium into a heavy water jacket, so as to simulate the coolant bypass on which the actual equipment will be located. Measurements were then made with graphite stacks of different size, and with different detector locations, to ascertain signal strength and background. Among other information, it was thus found that it would be necessary to keep the detector pulses as short as possible in order to minimize gamma pileup; further, it became apparent that detectors with rather special characteristics would eventually be desirable. Consequently, two fast, transistorized amplifiers were ordered.

The original intention of locating the graphite stack inside the sodium purification cell was abandoned, and a separate loop, with somewhat faster flow speed, will be installed. This loop is expected to be utilized by other experiments as well, and will be readily accessible at all times. Some thought has been given to the potential benefits of increasing the signal-to-background ratio of the monitor by means of neutron multiplication, that is, through insertion of fissionable material into the graphite moderator. Preliminary calculations suggest that a multiplication of the order of 5 to 10 will make a very considerable improvement in the over-all reliability of the monitor while minimizing the potential hazards which might be anticipated for stronger multiplications. With some multiplication, improved detectors, and fast electronics, the gamma background may eventually be eliminated.

A second series of equipment tests and possibly multiplication measurements is planned for September. All the equipment will then be moved to the National Reactor Testing Station, where further tests may be made and the complete monitor installed at its EBR-II site.

7. Component Development - Steam Generators

In the Progress Report for February, 1961 (ANL-6328) it was reported that a decision had been made to use two modified evaporators as superheaters in the steam plant. Late delivery of the special quality 2.25 w/o Cr-1.0 w/o Mo tubing for the second unit has delayed fabrication of the two units until this month. All of the components have now been received and assembly of the units has started. This involves modifying a completed evaporator fabricated as a spare, and assembling a second unit from the starting components.

8. Component Development - Fuel Reprocessing Facilities

a. Fuel Cycle Facility - The Fuel Cycle Facility Building is about 93 percent completed. Installation of services is now the main concern.

Equipment for use within the Facility is being designed and fabricated. The window shutters for the Argon Cell are being machined and assembled. The scrap-handling coffin has been delivered. The small scrap-handling containers which will be used for storage of low-level wastes have beenfound satisfactory.

A collapsible stand for manipulator carriages has been fabricated. The stand will be used to support a manipulator carriage after its removal from the bridge for purposes of inspection and maintenance.

Films have recently been discovered deposited on the internal polished glass surfaces of the shielding windows. This will cause a further delay in their installation. The new tentative installation period is between October 1 and November 30.

Design of equipment for a 5-kg scale skull reclamation process is continuing.

b. Remotely Controlled Methods and Equipment for Fuel Fabrication - Assembly drawings were completed for two remote indicating balances for the melt preparation station. Detailing of this equipment is under way. Specifications were completed for the balance readout indicators and these were submitted for procurement action. Designs, details and specifications were completed on the fuel pin processing and inspection equipment. The fuel element assembly and welding station details are estimated at 88% complete. Specifications were prepared and submitted for procurement of the dry argon compressors required to power the argon cell equipment.

Design work has been completed on leak detection equipment. Designs were 90% completed on the bonding machinery and specifications are being prepared. Fuel element construction and welding equipment is estimated at 76% complete.

The design of control equipment to be located in the subcell is estimated at 90% complete. Installation drawings for the basement area are estimated at 50% complete and of the operating annulus at 40% complete.

The injection casting furnaces are being constructed by the Laboratory and are estimated to be approximately 65% complete. The furnace vacuum-pressure system has been ordered. The vendor will submit manufacturing prints for approval and is to deliver the system

within 60 days of receipt of the order. Most of the component parts of the five station leak detector have been fabricated and are being assembled. Nine of twelve cabinets for pushbutton control stations in the operating annulus have been received. Pushbuttons and controls are being installed in these. Two 2,000 watt-second capacity stored-energy welding power supplies were assembled and tested.

9. Process Development

The initial core loading of EBR-II, which consists of 50 percent enriched uranium alloyed with noble metal fission product elements, will be recovered by melt refining. After the stainless steel jackets have been removed mechanically, the fuel pins are chopped and charged to a zirconia crucible where they are melted and held at a temperature of 1400°C for three to four hours. Approximately two-thirds of the fission products are removed by this treatment through mechanisms of volatilization and selective oxidation by the crucible. The purified product is collected in the form of an ingot by pouring the molten metal into a graphite mold.

A supplementary process involving a liquid metal system will be used to recover uranium and plutonium from the melt refining skulls. The skull reclamation process will also permit control of the concentrations of noble fission products in the recycled fuel. The process presently involves removal of a melt refining skull from a crucible by oxidation of the skull to convert it to a powder, selective extraction of noble metals (ruthenium, rhodium, palladium, molybdenum, and technetium, whose oxides are reducible by zinc) into molten zinc from a slurry of the oxides in a molten chloride flux, reduction of uranium oxides by magnesium in a zinc solution, uranium precipitations to enable removal of various fission products in the supernatant solutions, and a retorting step to isolate a uranium metal product.

A second liquid metal process will be used to concentrate the plutonium from the reactor blanket. In the current process, the about one percent plutonium-uranium alloy is dissolved in zinc containing 10 to 15 percent magnesium. Additional magnesium is then added to precipitate metallic uranium. The plutonium dissolves in the magnesium-rich phase and the desired separation is thereby accomplished. After separation of the phases, the uranium and plutonium are recovered by retorting.

Process studies include the determination of chemical and physical properties of substances of process interest, the development of engineering operations into workable plant-scale techniques and demonstrations of processes under levels of radioactivity anticipated in plant usage. Work to improve various process steps and to simplify the processes is continuing.

a. Melt Refining Process Technology - A fourth melt refining experiment with highly irradiated EBR-II-type fuel pins has been completed. A 364-gram charge of ten percent enriched uranium-five percent fissium, irradiated to an estimated burnup of one percent and cooled for 35 days, was melt refined in a zirconia crucible for one hour at 1400°C. This experiment thus duplicated the conditions of the third experiment with respect to holding time, temperature, crucible composition, and furnace atmosphere. A yield of 74 percent was obtained in the fourth run as compared to a 52 percent yield in the third run. The higher yield is attributed to certain precautions that were taken to limit the exposure of the fuel pins to air insofar as this was practicable. The pins were blanketed with argon whenever feasible; the silicon carbide abrasive used to clean the pins prior to melt refining was outgassed; the pins were sparged with argon during the cleaning operation; and the melt refining crucible was outgassed for $6\frac{1}{2}$ hours at 860°C before it was installed in the melt refining furnace. Although these procedures resulted in an improved yield, it is recognized that the improvisations which were used to simulate more closely the conditions within the EBR-II Argon Cell could not be entirely successful because of experimental limitations and that exposure of the pins to air at various times during the handling operations could not be avoided.

Analytical results from the second melt refining experiment with highly irradiated material are in agreement with results from the first. The removal of fission product yttrium and rare earths exceeded 99 percent; over 95 percent of the tellurium was removed. About 18 percent of the zirconium was removed; this amount is greater than the amount (9.2 percent) removed in the first run with highly irradiated material. These zirconium removals, although modest, are of particular interest since they were not observed in earlier work with unirradiated and low-activity materials. Analytical data for the third and fourth experiments are not yet available.

Additional experiments were made to determine the effect of storage conditions on melt refining yields. Fissium fuel pins were stored for 24 hours at 350°C in an argon atmosphere that initially contained six percent nitrogen and 1.7 percent nitrogen at the conclusion of the experiment. The melt refining yield was 84 percent. In comparison, a yield of 91 percent was obtained with pins that were stored under a pure argon atmosphere at the same temperature for two hours. In the EBR-II Fuel Cycle Facility, the argon atmosphere is expected to contain nitrogen in concentrations up to 5 percent.

Further studies were made of the transport of fission product activities through the off-gas handling equipment used in melt refining. The only elements found in larger than trace amounts were noble gases

and iodine-131. Samples of the argon atmosphere in the furnace at the completion of melt refining indicated that about 0.01 percent of the total iodine-131 present was in the gas phase.

b. Skull Reclamation Process - In a new demonstration of the skull reclamation process, successful removals of the noble metal fission products (ruthenium, rhodium, palladium, and molybdenum) were again obtained. The feasibility of recycling a crucible containing a heel of uranium and magnesium in zinc solution from a previous reduction step to the noble metal leach step was tested. No problems were encountered in the recycling operation.

The use of beryllia crucibles in retorting uranium concentrates has given reproducibly high recoveries (90 to 99 percent) of the uranium charged to the crucible. The main difficulty appears to be crucible quality. In the most recent run an isostatically-pressed beryllia crucible (about 90 percent theoretical density) was evaluated in an experiment under conditions essentially the same as in the intermetallic precipitation, decomposition, and uranium retorting steps of the reclamation process. The crucible cracked in the side wall during the precipitation step; however, it remained intact and readily released 99 percent of the uranium as product.

Laboratory studies confirmed difficulties observed earlier (see Progress Report, June, 1961, ANL-6387) in demonstration runs in obtaining rapid and quantitative reductions of uranium oxides with high magnesium concentrations (12 weight percent) in the zinc phase. In addition, it was found that fragments from the zirconia melt refining crucible are reduced under these conditions. More favorable operating conditions for the reduction step are being sought.

Preliminary design on integrated process equipment for pilot plant skull reclamation equipment is in progress. The equipment would be contained in an argon glovebox and have a capacity of 2.5 kilograms of skull oxide.

c. Direct Processing of Clad EBR-II Fuel Pins - Survey runs were made to determine the feasibility of dissolving stainless steel-canned EBR-II fuel pins in zinc and extracting the uranium into a fused salt by means of an oxidant such as zinc chloride. In $10\frac{1}{2}$ hours at 600°C all of the uranium was extracted by the flux while all of the iron and nickel and 83 percent of the chromium were dissolved by the zinc. At 500°C the reaction rates were too low and the experiment was discontinued after three hours.

d. Blanket Processing - The blanket process which is currently being demonstrated in small-scale runs consists of dissolving the 1 percent plutonium-uranium alloy in a 12 percent magnesium-zinc solution,

precipitation of the uranium by addition of magnesium to 50 percent concentration and cooling to about 450°C, and recovery of the plutonium from the supernatant by evaporation of the magnesium and zinc. The flowsheet for the blanket process is shown in Figure 8.

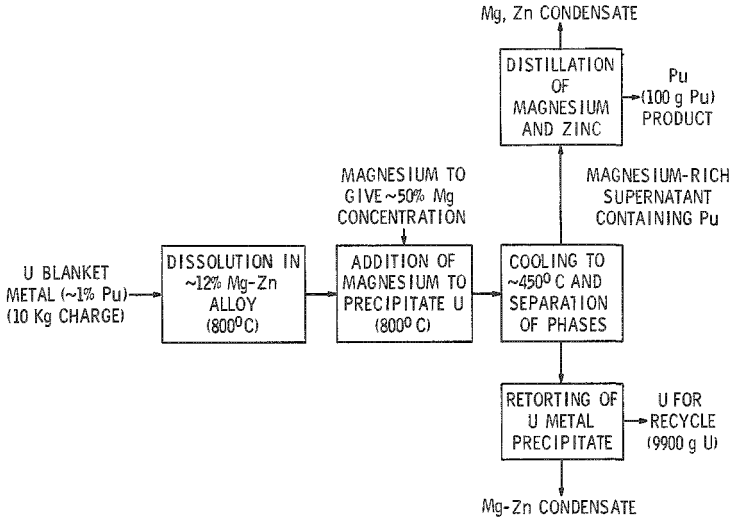


Figure 8

Liquid Metal Process for Uranium Blanket Metal

e. Plutonium Recovery Processes - Experiments to determine the distribution of cerium between liquid plutonium and calcium at 860°C and between plutonium and calcium-zinc solutions at 860°C and 705°C have been completed. Analytical data on cerium distribution are not yet available. It was found, however, that the solubility of plutonium in the calcium phase increases as zinc is added.

f. Materials and Equipment Evaluation - Solution stability and material demonstration runs with zinc-magnesium-uranium solutions were completed in impregnated graphite and tantalum crucibles. As in previous runs with other grades of graphite, some uranium was removed from solution by carbiding in the run made with the impregnated graphite crucible, thereby indicating that this material is unusable for skull reclamation process crucibles. The tantalum crucibles performed well in three-day long runs except that one of the crucibles failed at the weld around the bottom when the frozen ingot was remelted after the run had ended. A pyrolytic graphite coated crucible failed because the 30-mil pyrolytic layer separated from the base ATJ graphite.

Unalloyed molybdenum did not show promise as a container material for zinc-rich solutions. An arc-cast molybdenum coupon disintegrated during a 500-hour corrosion test in zinc-5 percent magnesium at 850°C.

g. Liquid Metal Distillation - Four additional runs were completed with the large-scale cadmium distillation unit. The maximum evaporation rate attainable for the present coil-susceptor combination is limited by the maximum allowable coil temperature (800°C) to 67 kilograms of cadmium per hour. A graphite susceptor has been ordered which will probably enable the design capacity (100 kilograms per hour) to be met.

Equipment to measure vapor-liquid equilibrium in liquid metal systems has been assembled in order to determine activity coefficients in binary metal systems such as the cadmium-magnesium and zinc-magnesium systems and in ternary systems such as the magnesium-zinc-cadmium system. In a preliminary test of the equipment, the boiling point of cadmium was measured over the pressure range 5 to 760 mm Hg with a precision of ±0.4°C.

h. Special Projects - The rate of solution of particulate metal solutes in agitated metal solvents under various conditions of agitation is being studied. Analysis of data for uranium dissolution in cadmium indicates only approximate agreement with the correlations of Hixson and Baum for nonmetallic systems.

Measurements of the diffusivity of uranium in liquid cadmium are being made in which uranium is diffused from a uranium-cadmium solution contained in tantalum capillary tubes immersed vertically in liquid cadmium.

10. Fuel Development - Core II

a. Fast Reactor Fuel Jacket Development - Fuel jacket material for future fast reactors must be superior in strength to stainless steel because of proposed higher operating temperatures. Several niobium and vanadium base alloys possess some of the required properties at temperatures in excess of 900°C. Requirements for these alloys are compatibility with sodium coolant and fuel, good weldability and fabricability, and high strength and low creep at elevated temperatures. Vanadium base alloys have been developed by Armour Research Foundation for fabrication development work at ANL. Commercial suppliers have been contacted to supply high strength alloy tubing or base material which can be fabricated into tubing.

Of the seven niobium base alloys hot rolled last month, only two could be further fabricated into 0.028-in. thick sheet by cold rolling. Reductions slightly in excess of 80% were possible with only minor edge cracking. A third alloy was severely edge cracked after only a 10% reduction by cold rolling. The remaining four alloy strips had broken up during

hot rolling and had to be scrapped. Further fabrication of the 0.028 in. thick strips will take place as soon as they can be annealed. Samples of these alloys will be tested for compatibility with sodium and plutonium fuel.

Four other niobium base alloys were cast, jacketed in 304 stainless steel, and hot rolled at 1150°C. The alloy strips have been removed from the cans and are in the process of being cold rolled into sheet.

Strips of 0.025 in. thick Rene' 41 sheet have been formed into $\frac{3}{8}$ in. O.D. tubes. The tubes are now waiting to be welded prior to being cold drawn.

III. REACTOR SAFETY (040117)

A. Thermal Reactor Safety Studies

1. Fuel-coolant Chemical Reactions

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

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

The series of condenser discharge runs with uranium wires in water at 100°C was completed. It was not possible to study the reaction in water above 100°C because of rapid corrosion. Reaction reached 100 percent in heated water when the initial metal temperature was 3000°C at a mean particle diameter of 200 μ. Runs in room temperature water gave only 40 percent reaction under identical conditions.

A brief study of the sodium-steam reaction at 200°C by the pressure-pulse method was begun. It was assumed that the rate of the sodium reaction would be limited only by gaseous diffusion and mixing processes with the reaction cell. Rates with sodium at 200°C were only slightly greater than those with uranium in the 1200-1600°C range suggesting that observed uranium rates were also diffusion limited. Rates with sodium were considerably above those previously reported with aluminum. The aluminum data are therefore not limited by diffusion in mixing.

In further studies of metal-steam and metal-water reactions by the levitation melting method, improvements in the two-color pyrometer have been achieved. Experiments have demonstrated the feasibility of cooling the levitation-induction coil with flowing oil at 110°C, thus avoiding the condensation of steam on the cold inlet side of the coil.

Preparations were completed for a series of experiments in TREAT on uranium dioxide core, stainless steel-304 fuel pins. The parameters to be studied are the behavior of a water-logged core and of fuel pins with a high-density core (as a function of the reactor period).

An out-of-pile test was made to check the reliability of an autoclave to be used in future work on fuel meltdowns in water at a high pressure. The water in the autoclave was maintained at 1000 psia (corresponding to temperatures of 285°C for water) for six hours. The temperature control was satisfactory and there was no indication of leakage.

Measurements and calculations were made on the internal pressures generated in oxide core fuel pins during reactor transients. It appears that a possible mechanism of importance is the pressure from the gas retained in the pores of the ceramic core. At high temperature, this gas pressure can cause swelling, or possibly rupture, of the cladding.

Computations were made on the IBM-704 to give estimates of the temperature as a function of time and radial position in an oxide core fuel pin during a reactor transient. The results indicate that a transient on a 60 millisecond period is fast enough so that the central core temperature can be calculated assuming an adiabatic process using the measured fission energy input. These computer calculations also showed the presence of a temperature lag between the clad and core and steep thermal gradients near the edge of the core. The problem of flux depression was evaluated for the case of a 17.7 percent decrease in flux from the surface to the center of the core; with this self-shielding of the neutron flux, the point of maximum temperature did not occur in the center of the fuel pin (as it did with uniform heat generation) but at a point of approximately one-half of the radius of the core. However, the total temperature difference with and without attenuation of the neutrons was only about 100°C so that the influence of flux depression is quite small.

2. Kinetics of Ignition and Oxidation of Reactor Materials

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

In theoretical studies to relate isothermal oxidation rates to observed ignition behavior isothermal oxidation rates of uranium were obtained in the region from 300° to 500°C.

Earlier studies showed that the addition of halogenated hydrocarbon compounds to air causes a marked decrease in the burning propagation velocity of uranium foil strips. Studies of the effects of these additives on ignition temperatures showed that CF_3Cl and CH_3Br lowered ignition temperatures while CHF_3 did not. This finding suggests that the exothermic reaction between Cl or Br with UO_2 may be involved.

Studies of ignition of zirconium powder in the vacuum-shielded apparatus have shown that the ignition temperature of 270/325 mesh spherical zirconium powder is between 275° and 325°C. This is approximately 350 degrees lower than observed for single foil (1 mil thick) which has a similar specific area.

An electronic device for transmitting the temperature of a sample without interfering with simultaneous operation of an analytical balance has been tested and proven to be adequate for anticipated plutonium oxidation and ignition experiments.

B. Fast Reactor Safety Studies

1. Core Meltdown Studies - TREAT Program

TREAT in-pile meltdown experiments are being performed on fast reactor fuel samples in order to obtain information on the types of sample fuel movement and failure, and to survey the mechanisms producing such phenomena.

a. Oxide Sample Experiments - The first two meltdown transients on oxide fuel were reported in the June, 1961, Progress Report, ANL-6387. Both tests were performed on a cylinder of fuel (about 1 cm long) inside an EBR-II steel jacket. Subsequent inspection showed that the sample had melted, but that there was no evidence for fuel fragmentation or explosive rupture. Steel cladding around the fuel section had melted. More detailed examination of the specimen is underway to attempt to determine the existence of zones of fuel-cladding interaction and possible oxide movement inside the cladding remains.

Four additional experiments on full EBR-II length oxide samples clad with standard EBR-II steel tubes were run. The experimental conditions were as follows:

Sample No.	Reactor Energy Release	Maximum Cladding Temperature	Est. Max. Fuel Temperature (Based on Cladding Temperature)
2	44	900°C	1450°C
3	62	1200	1950
4	84	1375	2280
5	117	1500	2475

Sample temperatures were measured by two or four Pt-Pt Rh thermocouples welded to the cladding. Preliminary analysis of the time vs. temperature curves indicates that effective fuel-cladding interface thermal conductance was of the order of 0.1 watts/(cm^2)(°C). Pre-experiment calculations of sample temperatures had covered the range of 0.04 to 0.2 watts/(cm^2)(°C). Direct calculation of this parameter would require knowledge of the details of surface contact between the fuel and cladding, and is being treated at present as a parameter to be obtained from the experiments.

Samples 2 and 3 have been inspected. In both cases, the cladding appeared to be undamaged. No fuel cracking was observed, although slight surface roughness was noted for sample 3. Pieces from the two elements were taken for metallographic examination.

b. EBR-II Elements in Stagnant Sodium - Three capsules of the third series of experiments on meltdown in stagnant sodium were run. A fourth capsule, instrumented with a pressure transducer, had been stored temporarily at TREAT, pending specification of new transient settings. It was discovered that the transducer had been damaged in shipment. Specification of transient conditions for this capsule will be changed, based on examination of transient records from the three tests completed. It had originally been planned for the capsule to be given a transient identical with that used for a similar sample instrumented with an internal thermocouple instead of a sodium bath pressure transducer. Assembly of the fifth capsule, containing the cast-in tantalum-molybdenum thermocouple was delayed, and it is now anticipated that this capsule will be shipped to TREAT during the next month.

c. Photography of Dry Meltdown Experiments - The first two transparent capsules of the current experimental series have been received at TREAT. Each contains a 10% enriched, half-length Fermi-I sample. The remaining three capsules (two containing 6% enriched EBR-II elements, and one containing a 6% enriched EBR-II pin clad with niobium) are being shipped. In conjunction with the tests, special small charcoal adsorber traps will be inserted into the gas purge system to test the adsorption of gas-borne fission product activity by traps operating at higher gas flow rates than used previously. Both room temperature and liquid nitrogen-cooled traps will be tested.

d. Pre-irradiated EBR-II Fuel - Seven of the 5 wt-% fissium cast EBR-II bonded and clad fuel elements irradiated in the MTR to burnups in the range of 0.5 to 2 atom-percent were removed from the capsules and examined. No cladding deformation or swelling was found. These seven samples will be stored for use in the future TREAT experiments on pre-irradiated fuel.

e. Package Sodium Loop - A prototype of the package sodium loop designed to fit inside the volume of a TREAT fuel element has been filled with sodium and given flow tests. Excessive oxide precipitation occurred at low temperature (~200°C), and plans are being made for incorporation of a device to remove oxygen from the loop sodium after filling. Work is underway to instrument a second prototype with miniature pressure transducers.

IV. NUCLEAR TECHNOLOGY AND GENERAL SUPPORT (040400)

A. Applied Nuclear and Reactor Physics

1. 3.0-Mev Van de Graaff

a. Determination of $d\bar{\nu}/dE$ for U^{235}* - These measurements were completed throughout the energy range from 40 kev to 1.58 Mev. The final results are given in Figure 9 which shows

$$\bar{\nu}(U^{235}) = 2.43 + 0.10\ E\ (\text{Mev})$$

and

$$\frac{d\bar{\nu}}{dE}(U^{235}) = 0.10 \pm 0.07\ \text{Mev}^{-1}$$

All of the measurements were relative to the thermal U^{235}-$\bar{\nu}$ value of 2.43. The $d\bar{\nu}/dE$ value obtained from these measurements (above) is significantly smaller than the value of about 0.14 Mev^{-1} previously used in reactor calculations. This smaller value is expected to extend from thermal to about 6 Mev where second chance fission becomes appreciable.

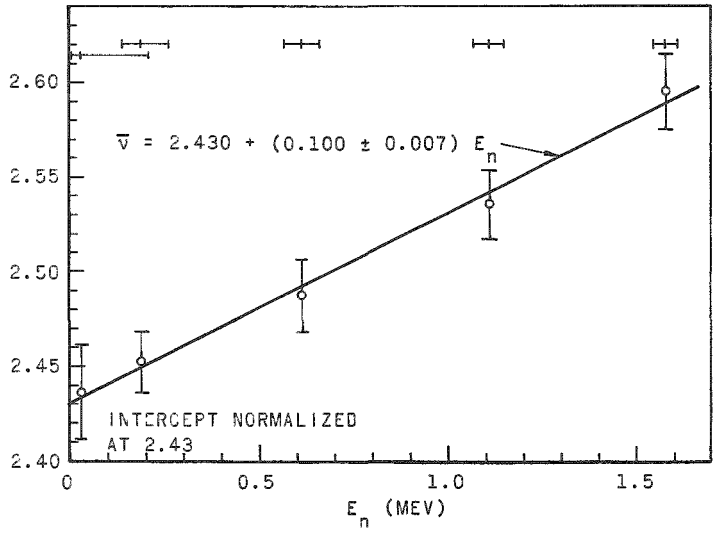


Figure 9
$\bar{\nu}$ for U^{235} as a Function of Neutron Energy

The experimental apparatus is being utilized for studies of the $\bar{\nu}$ of other fissile materials.

b. Delayed Neutron Measurements - A careful examination of delayed neutron emission from U^{238} fission is in progress. The results are of considerable interest from the applied point of view and are

*$\bar{\nu}$ is defined as the average number of neutrons emitted per fission.

significantly related to the theory of fission. The measurements thus far completed indicate that there is a sharp dependance of the U^{238} delayed neutron yield on the energy of the incident neutron in the region of the fission threshold. Indeed, the total delayed neutron yield changes by approximately 50% in going from incident neutron energies below the "knee" of the fission cross section to neutron energies corresponding to the plateau portion of the cross section. Such abrupt shifts could be quite important in fast reactor systems containing large amounts of U^{238}. They also indicate that sub-threshold fission is characterized by a considerably different fragment mass and/or charge distribution due to the effect of penetrating the fission barrier.

c. Fast Neutron Scattering, U^{235} - Studies of the elastic and inelastic scattering of fast neutrons from U^{235} have started. The first measurements indicated a quite clear residual nuclear level structure at ~ 50 kev. Measurements such as these, which clearly define inelastically scattered neutron groups, have not previously been made. As a result the inelastic scattering cross section of U^{235} has been uncertain. The work now in progress will provide greatly improved elastic and inelastic cross sections for U^{235}.

d. Scattering from Th^{232} - The elastic and inelastic scattering of neutrons from thorium have been studied throughout the incident neutron energy interval 0.3 Mev to 1.6 Mev. Fast time-of-flight techniques were utilized in conjunction with an electromagnetic pulsing and bunching system for the production of short and very intense neutron bursts. The differential elastic cross section was determined as a function of incident neutron energy. The cross sections for inelastic scattering resulting in the excitation of residual nuclear levels at 50 ± 2, 170 ± 10, 720 ± 20, 790 ± 20, 820 ± 20, 1050 ± 50, and 1150 ± 50 kev were measured. The angular distribution of inelastically scattered neutrons resulting in the excitation of the lower two residual nuclear levels was also determined. The measured individual inelastic excitation functions may be combined to obtain the total inelastic scattering cross section as a function of incident neutron energy. The experimental results are being compared with theoretical calculations of the Hauser-Feshback type and with the results of other experiments.

2. Argonne Thermal Source Reactor (ATSR)

An attempt to establish the relationship between reactor drifts and moderator temperatures is being made. Thermocouples were placed in the dump, core, and shield tanks of ATSR. Two all day runs have been made in which the temperature changes and automatic control rod drifts have been observed and recorded. Considerable difficulty has been encountered in maintaining the calibration of the six thermocouples being used and a new set of thermocouples will probably be necessary.

3. ZPR-VII Facility

a. High Conversion Critical Experiment - The initial loading of 3 wt-% enrichment (Hi-C) UO_2 fuel was replaced by a loading of 4.95 wt-% enrichment (BORAX-V) UO_2 fuel. Criticality was obtained with 715 fuel elements loaded in a 21 x 35 row array and spaced 1.27 cm (0.5 in.) center to center, and having a volume ratio of H_2O/UO_2 of 1.5/1. Twenty lattice positions are occupied by control elements, one five-fingered element remaining 33 cm withdrawn and the others completely withdrawn at criticality. The UO_2 pellet diameter is 0.871 cm (0.343 in.), and density is 10.29. The fuel is stacked in 61-cm columns inside Type 304 stainless steel tubing of 0.876 cm (0.346 in.) ID and 0.038 cm (0.015 in.) wall thickness.

A cylindrical core of 621 BORAX-V elements was critical without the top reflector, in comparison with the prediction of 625 elements fully reflected. The prediction was based on a three-group computation, and small corrections are needed to take account of a measured UO_2 pellet density of 10.29 rather than the expected value of 10.2.

Experimental work on the 621 element core includes relative distribution measurements of neutron density, U^{235} fission, and U^{238} fission, and absorption. Reactivity worths are being measured for core components including H_2O, Type 304 stainless steel, and UO_2 of 0.22, 4.95, and 9.9 wt-% enrichments.

Further examination of data from the 3 wt-% enrichment UO_2 core indicated that a bare critical cylinder would require about 985 fuel pins if loaded on 1.24 cm (0.488 in.) center-to-center square lattice. This spacing yields a volume ratio of H_2O/UO_2 of 1/1, or 3/1 atomic ratio of H/U^{238}. Measurements of control rod worths gave 1.73% ρ and 1.48% ρ for the five-fingered and the blade rods, respectively. About 0.28% ρ of the finger-rod worth was the result of water displacement. Removal of a central fuel pin resulted in a gain of 0.056% ρ. This may indicate an unexpected dependence on epithermal interactions.

4. Experimental Reactor Physics

a. Thermal Reactor Spectra - An investigation on optimizing the design parameters for neutron choppers has been completed. One interesting conclusion is that it is not desirable to cover a large energy interval at a single chopper speed. The optimum is a design objective of about 1.7 runs per energy decade, which corresponds to changing the rotor speed by a factor of two from one run to the next.

b. Pile Noise Technique - The pile noise technique for effective beta measurements has been found to be valid for simple two-group and reflected reactor kinetics models, provided that measurements are made in the thermal group and in the core. A new preamplifier required for this technique has been built and checked out.

c. Fission Neutron Yield for U^{235} - Measurements were made of collimator absorption of fission neutrons, and the main series of measurements using a manganese bath at CP-5 were performed. In this series four different fission chambers, with backing of either U^{235} or depleted U^{238}, were used. The fission channel included a fast preamplifier, A-159 amplifier-discriminator, and 20-megacycle scaler. The total counting efficiency of this channel will be determined in a coincidence experiment, which will be the last of those measurements requiring the thermal column of CP-5.

The fission chambers mentioned above are double hemispheres of 0.007 in. thick magnesium; two of these have a spacing between hemispheres of 0.125 in. and two have a spacing of 0.062 in. The fissionable coating on the inner electrode was vacuum-deposited for two of these and painted on for the other two. All chambers turned out to have surprisingly good resolution; pulse rise times were observed to be between 10 and 15 nanoseconds, adequate for our purpose. The vacuum deposited coatings were evaporated from a U^{235}-charged tungsten spiral with the inner electrode of the fission chamber rotating on an axis parallel to the spiral, while a mask of suitable shape, interposed between source and target, ensured uniform deposition over the whole surface. This technique appears to work rather well but is time consuming, since three to four charges are required to put on 100 microgram/cm^2. The heavier coatings were therefore painted on.

d. Analog-to-Digital Techniques for Auto- and Cross-Correlation of Reactor Dynamics Data - Six REDCOR Model 361 amplifiers have been received to replace the Beckman amplifiers for input to the ADC unit. A series of tests (noise, drift, linearity, frequency response, and common mode rejection) have been made and the amplifiers were found to meet specifications. Two power supply cards of the above amplifier were found defective and sent back to the manufacturer for replacement. The amplifiers have been installed in the ADC unit.

e. Reactivity Measurement in Subcritical Cores - A literature survey was completed and the various possible methods of subcritical reactivity measurements are being examined.

5. Theoretical Reactor Physics

a. CP-5 Hot Spot Factor - A further analysis of the design parameters (see Progress Report, ANL-6343, March, 1961) of a converter tube for use in CP-5 has been completed. It was recommended that the fuel loading be increased to 1200 gm of U^{235}. This would result in a 50% gain in fast flux compared to a facility containing 600 gm U^{235}. The total fast flux ($E > 9.12$ kev) at the sample will be 10^{12} n/(cm^2)(sec); and the flux with an energy greater than 1.35 Mev will be 0.25×10^{12} n/(cm^2)(sec).

The thermal neutron flux within the experimental space can be reduced to about 10^7 n/(cm^2)(sec) by inserting inside the fuel sleeve a 0.32 cm thick stainless steel sleeve containing 2 wt-% B^{10}.

b. Analysis of Moderator Scattering from Experiment - Additions have been made to a set of programs which process the data from the moderator scattering law experiment. The programs now include routines which compute the Scattering Law $S(\alpha, \beta)$ which represents the differential cross section, in terms of α = absolute value of momentum change, and β = energy change. They also include routines for computation of standard deviations on both the corrected time of flight distributions and the $S(\alpha, \beta)$.

More data for H_2O, D_2O, Be, BeO, and UO_2 have now been accumulated and processed. The major portions of the programs can now be operated independently.

c. Mathematical Numerical Methods Analysis - In connection with the development of methods for computing the elementary functions on a computer with automatic square root and with significance control, approximations to the function $f(x) = \tan^2 x/x^2$ have been investigated. Sin 2x, cos 2x, and tan 2x can be obtained from this function by no more than one square root, one division, and two additions. It was found that for $-\pi/8 \leq x \leq \pi/8$, the approximation

$$f^*(x) = \beta_0 + \cfrac{\alpha_1}{x^2 + \beta_1 + \cfrac{\alpha_2}{x^2 + \beta_2}}$$

with parameters

i	α_i	β_i
0	-	0.00711 63123 97111 485
1	-0.85129 77476 368449	+2.12722 27085 48637
2	20.99432 39488 2288	-7.03416 51669 79883

agrees with f with a maximum deviation of less than .31 x 10^{-11}. Since, for this range, f lies between 1.0 and 1.1, the relative accuracy of this approximation is acceptable for a computer with a 40-bit fraction.

B. Reactor Fuels and Materials Development

1. Corrosion Studies

a. Corrosion of Sintered Aluminum Powder Tubing - Armour Research Foundation powder tubing extruded through a bridge die has now been exposed to water at 290°C (as atomized, for 120 days and at 360°C and milled - 70 hours) for 126 days. There have been no additional tubing failures, even though metallographic examination of extruded tubing revealed defects at the junction where the metal streams join. The pieces of the milled powder tubing (which had failed after 92 days) have been continued in test for a total exposure to date of 106 days at 290°C. As of this time, the material being tested at 290°C reveals only a slight hint of end swelling. The phenomenon of end swelling (associated with porosity) is a sensitive indication of material quality, and powder material previously produced has been badly swelled after about 90 days. Tubing under test thus represents the best material made to date.

The eutectic bonded (Atomics International) plus "motor arc" welded tube end closures have been exposed for an additional 31 days (total time is 77 days except for one tube which has an exposure of 81 days). [Note: The exposure times given in the last Progress Report (ANL-6387) were slightly in error, being shorter than the actual time by about four days.] As of this time there have been no additional gross failures but small blisters have appeared on two of the powder tube closures. Additional samples will be prepared using powder product plugs and perhaps stronger tubing.

b. Zirconium Alloys for Superheated Steam - Alloys containing copper, iron, and nickel have been exposed in various conditions of heat treatment for periods varying from about 36 to 45 days to steam at 540°C and 600 psi. Rates are relatively insensitive to heat treatment and composition. In all cases the rate is decreasing with time and in many instances corrosion behavior can be expressed by the equation $W = Kt^n$. For the most part K is in the range 60-80 and n in the range 0.30-0.35 (W in mg/dm^2 and t in days). Rates are thus about 2 mdd after 40 days exposure.

These alloys are also being tested (as hot rolled and quenched from 900°C and this heat treatment followed by aging at 700°C) at 650°C and 600 psi. Exposure has reached 15 days and under these conditions the corrosion rate is much more sensitive to composition and heat treatment. The best alloys are those containing 1) 3.1 Ni-0.6 Fe; 2) 2.1 Ni-0.6 Fe; 3) 1.1 Cu-0.7 Fe; and 4) 1.1 Cu-1.2 Fe. Only two observations have been made, so rates cannot be discussed quantitatively. Within these limitations it appears that rates are about three times higher than at 540°C and 600 psi and that the aged samples are corroding about 30% faster than the quenched samples.

A new group of alloys has been fabricated in an attempt to reduce the absorption of corrosion product hydrogen, and to reduce the corrosion rate. These alloys contain titanium to promote electronic conduction in the oxide, platinum to discharge corrosion product hydrogen and molybdenum or tantalum (in equiatomic concentration to the titanium) to immobilize vacancies caused by the titanium. After seven days exposure at 540°C and 600 psi the best alloys are Zr-0.02 w/o Ti-0.04 w/o Mo and Zr-0.02 w/o Ti-0.076 w/o Ta-0.21 w/o Pt. Their appearance is comparable to the copper-iron-nickel alloys discussed previously (lustrous film) but the slope of the log-log plot (weight gained versus time) is up to $2\frac{1}{2}$ times greater. Duplicate samples of these alloys are now in test at 650°C and 600 psi.

2. Improved Fabrication Procedure for Zirconium Alloy Tubing

Zirconium alloy tubing is a desirable structural material for use in thermal reactors which are light or heavy water cooled or moderated. Systems have been designed which require tubing for coolant water, moderator columns, fuel jackets, and instrumentation. Large diameter thin wall tubing and sound small diameter tubing are not commercially available.

Defect-free zirconium alloy tubing is available in many sizes down to about $\frac{3}{8}$ in. OD. However, careful inspection is required to separate the defective material from commercially produced lots of tubing in order to arrive at defect-free tubing. The reduction of tubing to diameters less than $\frac{3}{8}$ in., principally by sinking, results in increasing roughness of the inner surface and in numerous though intermittent cracks. These cracks are oriented principally along radial planes. Cracks which start at the ID surface vary in depth but may penetrate to $\frac{2}{3}$ of the wall thickness or even further.

An attempt is being made to determine the mechanism responsible for initiation and propagation of internal cracks and to devise methods for eliminating this phenomenon. The present goal is to produce a small diameter, (0.080 in. OD x 0.015 in. wall), tube with a satisfactory inside surface free from cracking.

Although commercially available 0.299 in. OD tubing which is free from indicated defects has been used for development work, the complete fabrication background from starting ingot stock to finished product should be known. For this reason base extrusions from ingot stock and forged slab have been prepared as starting material.

Six Zircaloy-2 hollow billets were successfully extruded into tubing from powder metallurgy product ingots prepared by the General Electric Co.,

Detroit, Michigan. The powder ingots were produced by a modified vacuum hot-pressing technique. The ingots were machined to 0.934 in. diameter billets, jacketed in copper, and extruded at 725°C. The extrusion ratio was 10:1 and the extruded Zircaloy-2 tube measured 0.314 in. ID x 0.411 in. OD.

The extruded tube with the copper jacket was drawn on a relatively soft, solid, copper rod "core" which coelongated with the tube. This prevented the excessive wall thinning associated with drawing on a mandrel or a rod. Drawing was continued with and without low temperature intermediate anneals at 500-600°C. After reducing the tube to size and final annealing, the completely softened core was stretched plastically by pulling the ends and was freely removed from the tube.

Tubes finished to 0.080 in. OD and 0.058 in. OD with 0.011 in. wall by this process have been free from apparent OD and ID cracks. Additional tubes are being prepared for fabrication in this manner using different starting material and varying the reductions in order to control the final OD and wall thickness and to compare variables introduced in the starting material.

3. Examination of Irradiated EBR-I Mark III Fuel Elements

A postirradiation examination of two EBR-I Mark III fuel rods, F53 and F72, which received reactor exposures of 1625 and 2682 Mwh, respectively, was conducted in order to determine if the observed diameter increase in the fuel section was due to high temperature mechanical deformation such as compressive creep or due to fuel swelling. Since rod F72 was operated for 1269 Mwh in the sheared condition (stabilizing ribs removed) an analysis was also possible of the influence of ribs on radial restraint.

Density measurements by immersion were performed on one-inch sections cut from the fuel portion of each rod and the resultant percent density changes were then calculated using a preirradiation figure based upon the density of the bottom portion of the fuel. Volume changes resulting from diameter increases (assuming no change in length) and from density decreases were calculated and plotted as a function of distance along the fuel portion of the rod. These are shown in Figures 10 and 11 for rods F53 and F72, respectively. Strain levels imposed upon the Zircaloy-2 cladding along the length of the fuel due to diametral expansion are illustrated in Figures 12 and 13. The values of the density decreases, diameter increases, volume changes, and cladding strains are given in Table X.

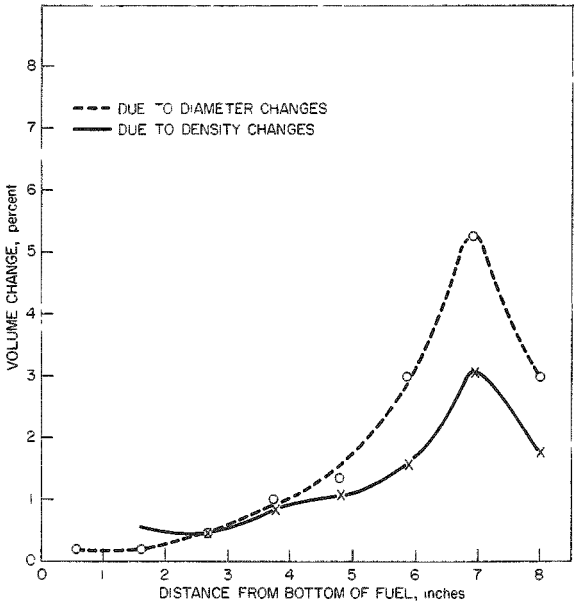


Figure 10

Comparison of Volume Changes Due to Diametrical Expansion with Volume Changes Due to Decreases in Density for EBR-I, Mark III Fuel Rod F53

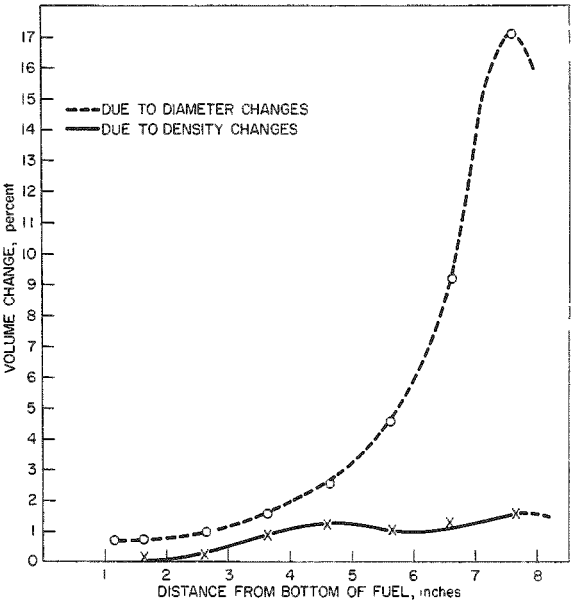


Figure 11

Comparison of Volume Changes Due to Diametrical Expansion with Volume Changes Due to Decreases in Density for EBR-I, Mark III Fuel Rod F72

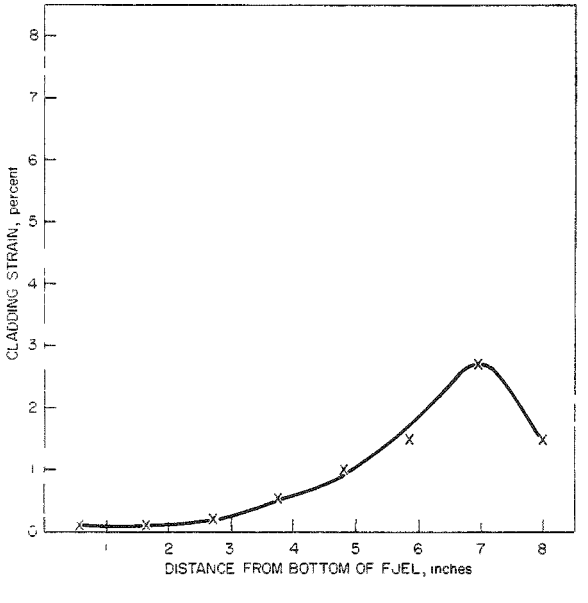


Figure 12

Zircaloy-2 Cladding Strain in EBR-I, Mark III Fuel Rod F53

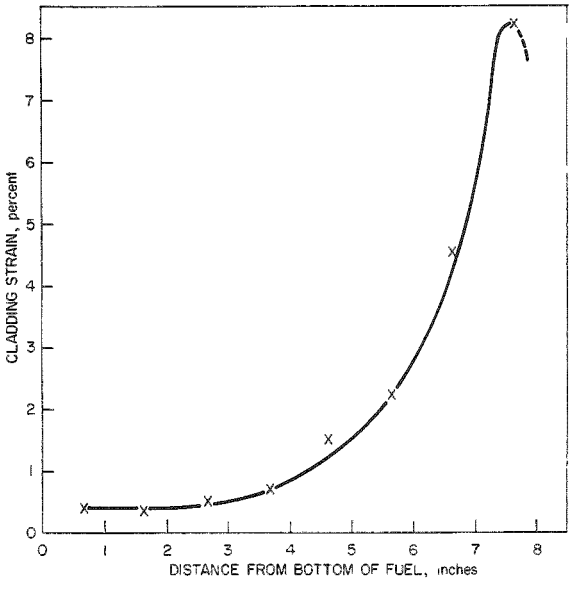


Figure 13

Zircaloy-2 Cladding Strain in EBR-I, Mark III Fuel Rod F72

Table X. Postirradiation Data on EBR-I Mark III Fuel Elements F53 and F72

Rod No.	Position from Top of Fuel to Center of Specimen (inches)	Density (gms/cc)	Density Decrease (%)	ΔV_f[a] (%)	ΔV_d[b] (%)	Diameter[c] Increase (mils)	Cladding Strain (%)
F53	1/2	15.2108	1.75	1.78	3.0	6	1.5
	1-9/16	15.0170	3.00	3.09	5.5	11	2.7
	2-5/8	15.2781	1.32	1.33	3.0	6	1.5
	3-11/16	15.3166	1.07	1.08	1.4	4	1.0
	4-3/4	15.3516	.84	.85	1.0	2	.5
	5-13/16	15.4072	.48	.48	0.5	1	.2
	6-7/8	15.3930	.57	.57	0.2	0.5	.1
	7-15/16	15.4820	-	-	0.2	0.5	.1
F72	7/8	15.6059	1.46	1.48	17.0	33	8.2
	1-7/8	15.6635	1.22	1.23	9.1	18	4.5
	2-7/8	15.6839	.97	.98	4.5	9	2.2
	3-7/8	15.6532	1.16	1.17	2.5	6	1.5
	4-7/8	15.7078	.82	.83	1.5	3	.7
	5-7/8	15.8027	.22	.22	1.0	2	.5
	6-7/8	15.8272	.06	.06	0.7	1.5	.4
	7-7/8	15.8371	-	-	0.7	1.5	.4

[a] ΔV_f = Change in volume due to density decreases where $\Delta V = \Delta \rho / 1 - \Delta \rho$.

[b] ΔV_d = Change in volume due to diameter increases assuming all expansion was diametrical.

[c] Diameter increases taken from diameter profile curves by F. D. McGinnis, February 1960 (F53) and R. G. Matlock and J. D. Leman, December 1960 (F72).

Although rod F72 was in the reactor for a total of 2682 Mwh (approximately 0.18 a/o burnup) and rod F53 was exposed for 1625 Mwh (approximately 0.13 a/o burnup), the density changes in rod F72 were only half as severe. This may be related to the fact that rod F72 was located at 2.06 in. from the reactor center, compared to 0.90 in. for rod F53. Hence, it operated at a lower burnup rate and at a lower temperature (465°C vs. 415°C, see Figure 18, ANL-6301). In addition, the greater amount of upsetting in rod F72 (see Table X) may have caused a collapsing of the voids formed in the upper regions of the fuel during irradiation, resulting in a lowering of the actual observed fission product induced density decreases. The large amount of upsetting in rod F72 occurred when the rod was operated for the last 1269 Mwh in the reactor in the sheared condition.

The diameter changes were almost twice as great for rod F53 and nine times as great for rod F72 as that which would have resulted from decreases in density. This indicates that the diameter increases in the fuel were due more to elevated temperature mechanical deformation than to fission product induced swelling. The greater amount of deformation observed in rod F72 as compared to rod F53 was probably due to the absence of radial restraint caused by the part-time operation of the rod without its ribs, as mentioned previously.

The level of strain imposed upon the Zircaloy-2 cladding by diametral expansion was approximately 8.2% for rod F72 and 2.7% for rod F53. No splitting of the cladding due to these strain levels could be observed visually, although the actual presence or absence of cracks in the cladding will not be known until metallographic examination is complete. The performance of the Zircaloy-2 was quite good considering the diameter increase which had occurred and the subsequent level of strain imposed upon the cladding.

4. Behavior of Braze Material in Superheated Steam

Stainless Steel (Type 304) assemblies brazed with Coast Metals 60 (Ni-20 w/o Cr-10 w/o Si-3 w/o Fe) and Nioro (82 w/o Au-18 w/o Ni) are being tested in superheated steam at 540°C and 650°C (600 psi in both cases). The joints appear to be in good condition after the exposure times indicated below:

	Temp., °C	
Material	540	650
Coast Metals 60	68 days	20 days
Nioro	30 days	20 days

5. Lightweight Alloy for Liquid Mercury

The surface treatment of titanium has been continued.

The effect of localized attack on the corrosion behavior of anodized titanium was observed in mercury at 370°C. It was found that there was no accelerated attack on the scratched and bent areas of the titanium specimens during 336 hours of exposure (static). A series of anodized films of different thickness on titanium is now being prepared.

Titanium was also nitrided in purified nitrogen at 850°C. Preliminary results revealed that nitrided titanium was immune to mercury at 370°C after 336 hours exposure under static conditions. These samples are now being observed in mercury at higher temperatures.

Metallographic examinations of these tested specimens are in progress.

6. Ceramic Fuels

a. Plutonium Carbide Melting - Six additional melts of plutonium carbide were prepared and the attempt was made to cast these into 0.144 in. diameter graphite molds. The PuC lacked fluidity to fill these molds. There also appeared to be surface reaction between the graphite and the PuC. Tests of mold and crucible coatings continue.

b. Uranium-Plutonium Carbide - An investigation is being made into possible processes for the preparation of high density solid solutions of UC-PuC. One method involves the simultaneous reaction of UO_2 and PuO_2 with carbon, using petroleum pitch as the carbon source. A mixture of UO_2, PuO_2 and petroleum pitch was reacted in the form of a loose powder at 1400°C in vacuum. The reacted powder was pressed into pellets and sintered at 1750°C. The fired pieces had a geometric density of approximately 80% of theoretical.

c. Uranium-Thorium Sulfide - Work has continued on the determination of properties of US, ThS and solid solution bodies.

The most recent batch of US is the best produced to date, analyzing 99.8 w/o US phase and only 0.2 w/o insoluble UO_2 residue. The improvement is attributed to multiple hydriding of the U turnings in the low temperature sulfiding step.

Groove etching of US, described in the March Progress Report (ANL-6343 page 65), is now better understood. It is thought that the grooves result from the selective etching of stacking faults which occasionally occur in face centered cubic structures when the close-packed (111) planes fault from the ABCABC cubic sequence (US) to the ABAB close packed hexagonal sequence (γ US_2). The apparent stepping of the wider grooves was found to be purely an optical effect and the curvature of the grooves toward the pores is believed due to surface rounding of the rather porous specimens during polishing.

The oxidation behavior of a US sample of 92% theoretical density in air at one atmosphere was observed. A thin surface coating was formed which appeared black at 250°C and seemed to protect the sample from further oxidation until 460°C when the sample ignited to form a mixture of UOS and UO_2. A more refined oxidation study will be run in a differential thermal analysis apparatus which permits the O_2 pressure to be varied over a wide range. Steam corrosion tests will also be made in this apparatus.

Vicker's hardness at a 100 gm load was measured. The US phase in different samples showed a hardness varying from 228 to 276 gm/mm^2 while ThS was only slightly softer at 219 to 246 gm/mm^2. The small oxysulfide areas gave values ranging from 325 to 500 gm/mm^2. These values indicate that the monosulfides are relatively soft materials with some promise of machinability. In support of this, a half inch long US sample of 92% theoretical density was drilled through, turned and faced on a lathe and smoothly filed, using considerable care.

ThS-US mixtures having molar ratios of 0:100, 27:75, 50:50, 75:25 and 100:0 were fired in vacuum for one hour at 1815°, 1935° and 2050°C. Densities ranged from 79.7 to 83.7% of the theoretical density at 1815°C, 81.6 to 87.6% of the theoretical density at 1935°C, and 72.5 to 97.4% of the theoretical density at 2050°C.

At 1815°C, lattice parameters showed negative deviation from Vegard's law up to about 30 m/o ThS and positive deviation at greater than 30 m/o; at 1935°C these deviations remained to a much lesser extent, and at 2050°C essentially a straight line relationship between US ($a_0 = 5.491$Å) and ThS($a_0 = 5.685$Å) was achieved. Two things were shown: (1) a complete series of solid solutions can be formed between US and ThS, but, in this case, not until 2050°C, and (2) ThS is more easily soluble in US than is US in ThS.

Modulus of rupture measurements on the solid solution series showed a steady increase in strength from US to ThS. The 1815° and 1935°C series both showed values increasing from about 20,000 psi for US to 30,000 psi for ThS. The 2050°C series had too wide a density range to be of significance.

Electrical resistivity measurements did not show a linear decrease from US to ThS as might be expected. The 1815°C series increased from a value of 158 microhm-cm for US to 227 microhm-cm for $U_{0.50}Th_{0.50}S$, then dropped to 22 microhm-cm for ThS. The 1935°C series showed similar values but reached a maximum at $U_{0.75}Th_{0.25}S$. The 2050°C series had much higher values for the solid solution members.

7. Differential Thermal Analysis Investigations

The oxidation behavior of oxides containing uranium is being investigated using differential thermal analysis (DTA) techniques.

Preliminary experiments are concerned with the surface area and composition of powdered samples, as well as the gaseous environment in which the samples are tested. Figure 14 for example, illustrates the effect of dynamic gas flow of oxygen and nitrogen through a UO_2 sample with respect to the normally reported static air DTA curve. The DTA curve in static air was quite similar to those reported by other investigators. DTA peaks were observed at 225° and 370°C. The introduction of flowing nitrogen changes the DTA curve slightly. The nitrogen probably provides a slight oxidation barrier to the sample thus broadening and shifting the peaks to 257° and 400°C. The DTA curve under a dynamic oxygen flow also gave the two characteristic peaks for UO_2 oxidation. The first peak was observed to shift from 225° to 245°C whereas the second peak temperature remained constant at 370°C. The peak shape of the second, however, was much sharper and more vigorous, indicating rapid oxidation of the sample to U_3O_8 over a narrow temperature range.

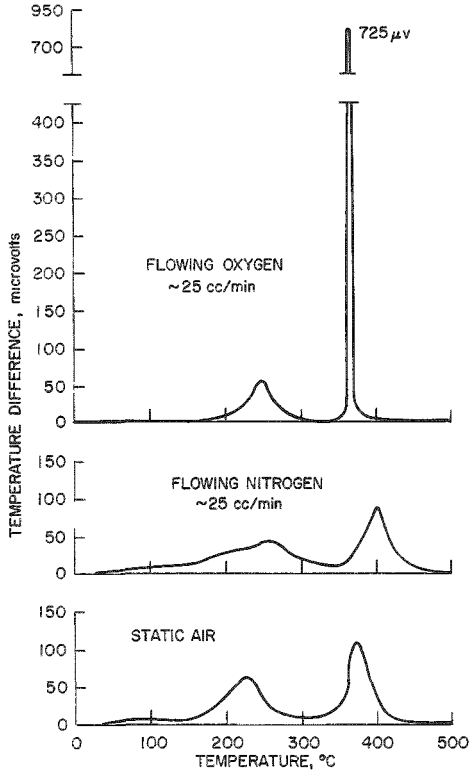


Figure 14

DTA Curves for Spencer "Ceramic Grade" $UO_{2.06}$

8. Nondestructive Testing

a. Ultrasonic Techniques - The work discussed in the June Monthly Report is being continued. The tests on the 0.030 in. thick Al plates are complete. The series of artifical flaws were 1.0 in. long, 0.032 in. wide and had depths varying from 0.001 in. to 0.009 in. After the tests were completed, the depths of the nominally 0.001 in. and 0.002 in. deep flaws were measured with a microscope. They were found to have actual values of 0.0004 in. and 0.0014 in., respectively.

The plate was positioned so that the defects were not facing the transducer; they were thus simulating inner surface defects. All these defects were easily detected using Lamb waves. Furthermore, in this ideal situation (i.e., the flaw always oriented perpendicular to the plane of the plate) there is a correlation between amplitude of the received pulse and the depth of the flaw. Using shear waves, the smallest defect that could be found was 0.003 in. deep. In cases where the defect can be found with both types of wave motion, the amplitude of the received signal is always much greater with Lamb waves than with shear waves.

When the plate is reversed in position so that the flaw is facing the transducer, all flaws are readily and easily detected by both methods, and there appears to be no advantage in using Lamb waves when looking for outer surface defects.

Additional plates were made with narrower and shorter flaws. These had 0.002 in. widths, 0.5 in. lengths, and depths varying from 0.001 in. to 0.009 in. The same results as given above were found with these smaller flaws. However, on these smaller flaws, there is a much greater difference in the response of Lamb waves and shear waves to a defect. On the 0.009 in. deep flaw the shear wave gave a response of 3.1 volts and a Lamb wave a response of 16.0 volts at the same gain settings.

b. Neutron Techniques - Continuing the study of X-radiographic intensification methods as applied to neutron radiography, it has been found that the use of a single lead screen does offer some intensification with all the metal neutron converter screens. The lead screen data reported last month indicated that sandwiching film between 0.005 in. lead screens resulted in a greater neutron radiographic speed only with cadmium and gadolinium converter screens. Rhodium, indium and silver used in this manner decrease the speed. The apparent reason for this loss in speed is that the beta emission resulting from neutron activation of the converter screen is absorbed in the lead screen between the converter and the film. Some verification for this is found in the presently reported data which involve the use of film sandwiched between the converter screen and a single lead screen (0.005 in. thick).

Neutron radiographic speed increases over the use of only the converter screen and film have been found for cadmium and gadolinium (approximately 50%) and for rhodium, indium and silver (10 to 20%). There appears to be no significant loss in image quality with the use of the lead screen.

The recording of neutron diffraction patterns on film has advantages under some circumstances. An initial effort to record such data has been made by photographing a reflection from a single crystal of alpha uranium. The method employed a B-10 loaded scintillator and Type F X-ray film. A good exposure was obtained in 30 seconds. Further studies of this method of recording neutron diffraction patterns are planned.

C. Reactor Materials Development

1. Radiation Damage in Steel

The problem of radiation damage to pressure vessel steel is of primary importance to the industry, in particular in the design and safety evaluation of water-moderated power reactors. Realizing that a clear explanation of radiation embrittlement will require bridging of many gaps in current technology, work has been proceeding on three phases of the problem. These are described below.

a. Theoretical Model - A theoretical model of the damage mechanism is needed which will yield a generally applicable method of reporting neutron exposure, taking proper account of differences in neutron spectra. While the simple model in which damage production is proportional to the incident neutron energy and effective scattering cross section appears reasonable, it lacks experimental verification. The model has been revised to provide more detail in the high energy region of the spectrum and calculations of damage rates in different facilities are being made. These will be applied to experimental results to see if correlations can be obtained. Irradiations are being planned for CP-5, and dosimetry capsules are now being exposed in EBWR in preparation for irradiation during high power operation.

b. Brittle Fracture - Since embrittlement is the most feared property change due to irradiation, tests involving this property are used to evaluate radiation damage. A review has been started on the theory of brittle failure and methods of testing. It is proposed to evaluate testing methods now in use and investigate new methods. Should a new technique appear sufficiently attractive, experimental work would follow on unirradiated material, with the hope of ultimate application to irradiated samples.

c. Magnetic and Dynamic Properties - Radiation-induced changes in these properties would be of great interest and practical value. If correlations could be developed between these properties, irradiation exposure, and mechanical properties, irradiation tests could be performed nondestructively and on small samples to measure damage rates at points of interest in reactors.

Work on magnetic properties has been discussed in the last several Progress Reports. Current work involves measurements of coercive force and magnetic inductance on unirradiated heat-treated, and irradiated specimens of pressure vessel steel SA-212B.

Eight magnet bar specimens were prepared from unused EBWR SA-212 plate through the 4-in. gauge thickness direction to determine the influences of machining, and location within the gauge thickness.

Measurements with the inductance comparator bridge on all eight bars were equal to that for an identical specimen which had been fully annealed by slow cooling from 1550°F/1575°F.

The coercive force of SA-212B of the heat-treated bars described in the May 1961 Progress Report (ANL-6374, p. 46) was measured. This measurement (the magnetic field strength required to restore the residual magnetization to zero) confirms the previous findings of the 100-cps bridge comparator measurements. In both measurements, the mechanically harder materials were also magnetically harder. Both sets of measurements indicated that the steel was magnetically harder than fully annealed material for all annealing temperatures up to and including 1350°F, and independent of thermal heat treating time up to 3 hours at temperature.

Inductance comparator bridge measurements of 34 additional samples of SA-212B cut from unbroken irradiated specimens (ANL-26 irradiation in MTR, described in the June Progress Report, ANL-6387, p. 55) showed that the magnetic properties at 100 cps were that of fully annealed (from 1550°F/1575°F) and unirradiated SA-212B. Preparations are in progress for the measurement of the d-c coercive force of these bars, which include specimens exposed to four levels of irradiation. At a later date, these irradiated bars are to be thermally heat treated and examined magnetically for clues to the post-irradiation thermal heat treatment re-embrittlement phenomena.

2. Irradiated Boron-Stainless Steel

Samples of chips and cores from impact slug remnants of irradiated 2% boron-stainless steel (from the ANL-30 irradiation in MTR) were analyzed for lithium and boron to check the burnup of the B^{10} isotope. The concentrations of boron in the surface region and the core were lower in two of the three slugs analyzed; the lithium analyses showed a higher concentration in the surface region as would be expected from the greater burnup of the boron-10. Lithium-7 is found as a result of the n,α nuclear reaction on B^{10}.

The third sample shows a peculiarity which is ascribed to the accidental inclusion of Type 308-L metal used in joining the welded tool plate. The assumption of the presence of austenitic weld metal is strongly supported by the order of magnitude greater radiation levels of the samples themselves. Similar measurements of activity levels between irradiated Type 304 stainless steel and the 2% boron-stainless steel were recorded in routine cave monitorings. The results of the analyses are summarized in Table XI.

The boron analyses show that the original 2.26% boron-stainless steel was almost completely depleted of its B^{10} isotopic content. The Li^{7} concentrations are comparatively low and indicate that a leaching of the

light alkali metal from the boron-stainless steel matrix by the NaK coolant took place during the irradiation. It is concluded that the ANL-30 irradiation did indeed burn out all of the B^{10} isotope originally present.

Table XI. Analyses of Irradiated Boron-Stainless Steel Samples

Sample	Weight (gms)	Li (%)	B (%)	Radiation Level, at 2 in.	
				Hard-Soft	Hard
1 - surface chips	0.3437	0.084	1.83	160	110
1 - core	0.6431	0.080	1.93	250	190
2 - surface chips	0.3080	0.040	0.61	1300	1100
2 - core	0.7230	0.029	0.39	4000	3100
3 - surface chips	0.3571	0.048	1.87	130	100
3 - core	0.7165	0.042	1.95	250	200

D. Heat Engineering

1. Double Tube Burnout Test

A double tube burnout test section which will permit burnout data to be taken at constant quality is being constructed. This will permit burnout data to be taken while maintaining constant flow conditions in the channel during the burnout transient. A loop has been modified to accept the new test section and a series of tests have been run to pick the optimum location of the burnout taps, and to verify that the burnout detector truly measures the temperature excursion at burnout.

2. Hydrodynamic Instability

A dual purpose program to reduce raw data to useable form and to compare the reduced data with present correlations has been programmed and debugged. The data for the series of runs on the 1.6 cm ID test section and riser has been reduced.

The transient behavior of the loop following a step increase and a step decrease of power has been measured. A test of the loop operation during sinusoidal variation of power will be performed and the results of all three tests will be compared with an analog model of the loop hydrodynamics.

3. High Void Natural Circulation Study

The flow characteristics and hydraulic resistances for a high void natural circulation loop are being evaluated. Experiments were completed

using first one downcomer and then two downcomers, thus giving two sets of conditions. Three orifice plates of various diameters were added in the downcomer to vary the single-phase flow resistance for each condition above.

4. Boiling Liquid Metal Experiment

The small scale loop designed for measurement of vapor volume fractions and two-phase frictional behavior of flowing alkali metals is still under construction. Fabrication of loop components is nearly complete. All auxiliary equipment and instrumentation has been received with the exception of the pressure and differential pressure transmitters. The vendor's original delivery date has been delayed by fourteen weeks. Temporary equipment will be used in the interim.

5. Packed Bed Reactor Studies

Design and selection of components for a small in-pile loop for packed bed fuel studies is continuing. The heart of the helium loop design is a small oil-free, "maintenance free" commercially available rotary compressor. This machine introduces a practical limit on the system pressure loss of about 0.8 atm which must be apportioned among the lines, valves, disconnects, flow meter, heat exchanger, packed bed and filter. This limit is in opposition to restrictions imposed by the in-pile thimble size as well as the desire to keep the system small to minimize dump tank size requirements. The final solution (which will dictate maximum power of the experiment) has not been reached and is dependent upon the loss characteristics of the components selected.

Generalized pressure drop characteristics of the packed bed and for various heat exchanger passages have been calculated and work has begun to determine what type of gas filter and/or fission gas adsorber is required, and where it can be located in the system. The problem of disconnecting the external loop from the radioactive in-pile equipment appears to have a solution in the use of self-sealing, quick-connect unions, which would allow a minimum of manual effort and gas spillage during the disconnect operation. The design philosophy with respect to flow and pressure instrumentation has been to avoid extending small diameter pressure leads beyond the in-pile thimble or the main coolant line. Commercial turbine-type flowmeters and miniature strain gauge differential pressure transducers satisfy this criterion.

E. Separations Processes

1. Fluidization and Fluoride Volatility Separations Processes

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

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

Processes for recovering uranium from zirconium-enriched uranium alloys are being developed. Use of hydrogen chloride for zirconium removal and decladding followed by direct fluorination of uranium is being tried. The reactions between various alloys and zirconium tetrachloride are being examined in corrosion studies.

The chemistry of plutonium hexafluoride is being investigated in order to obtain information useful in establishing process flowsheets as well as information of a more fundamental nature.

a. Direct Fluorination of Uranium Dioxide Fuel - Most of the previous fluorinations of uranium dioxide pellets have been made on unclad pellets. Two fluorination runs have been made in the 3-inch diameter column in which the pellets were contained in one-inch long open-end pieces of Zircaloy tubing to simulate chopped fuel elements. The conditions for both runs were as follows: bed temperature, 500°C, off-gas recycle at 0.6 cu ft/min. The feed stream was pure fluorine, but the off-gas recycle contained oxygen produced in the reaction which acted as a fluorine diluent. One run was made with a fluidized bed of alundum grain and the other without a fluid bed. While appreciable fluorination was carried out, operational difficulties appeared in both runs due to ignitions of the Zircaloy and temperature surges. In the case of the fluid-bed run, severe caking developed due to sintering of the bed material.

The rate of fluorination of uranium dioxide pellets was observed as a function of temperature in the range of 350° to 500°C. Both oxygen and nitrogen diluent (~ 80% diluent) experiments were made; an alundum grain fluid bed was used in all cases. Although the temperature effect on the fluorination rate was the same for both gas diluents, the absolute rates for oxygen dilution were about 5 times higher than those for nitrogen dilution over the range of experimental conditions. The UF_6 collection rate rose from 10 g/hr at 350°C to 230 g/hr at 500°C for nitrogen diluent; from 100 to 700 g/hr for oxygen diluent. For the rate constant expressed as a function of temperature in the form

$$k = Ae^{-E^1/RT} ,$$

values of E^1 of about 14 kcal/gmol were indicated for both the nitrogen diluent runs and for the oxygen diluent runs. These temperature coefficients agree with those obtained in earlier laboratory experiments.

b. Halogenation Studies on Uranium-Zirconium Alloys - Engineering-scale studies on a fluid-bed volatility process for recovering uranium from enriched uranium-zirconium alloy fuels are being resumed in view of favorable results obtained in recent laboratory experiments. The first step, direct hydrochlorination of the alloy, conducted in fluidized beds of inert solids for heat removal has been demonstrated in the two-zone reaction studies. Adequate uranium removal from the bed by fluorination, heretofore not attained for reasons attributed to the presence of zirconium tetrafluoride and the type of bed solid, is expected to be achieved by changes in the process. These changes involve use of specific bed material and complete removal of the zirconium as the volatile tetrachloride. During hydrochlorination the uranium reacts to form the solid trichloride which is expected to be retained in the fluid bed or in packed beds of alundum.

Further laboratory work has shown that pure crystalline alumina is a suitable inert solid for use as a fluidizable medium in the fluoride volatility processing of enriched uranium, zirconium alloy fuels. Three grades of Norton's Alundum (crystalline aluminum oxide) were investigated for use. Best results were obtained using Norton's "Blue Label" Grade RR 60 mesh Alundum. An average of 99.5 percent removal of uranium was obtained on fluorination of the fluid bed following the hydrochlorination step for 5 hours at 350°C. The uranium concentration in the initial sample was between one and two percent. The experimental procedure was representative of what would be used in process application.

c. Corrosion Studies - A parallel study has been initiated to permit evaluation of metal filter materials and potential materials of construction for use in this work. Preliminary results of bomb-type corrosion experiments in which specimens were exposed to zirconium tetrachloride vapor at about 370°C and 2000 mm Hg pressure showed corrosion equivalent to 0.3 to 0.4 mil/mo in 35 hour tests and 0.1 or less mil/mo over a 170 hour period for A-nickel and Inconel, somewhat higher values for stainless steel and Monel. A single test of sintered nickel, using the apparent surface area rather than the actual area (these may differ by a factor of ten) gave a corrosion rate of 2 to 4 mils/month for a 170-hour test.

d. Plutonium Fluoride Studies - Laboratory-scale fluorinations of solid solutions of uranium and plutonium dioxides mixed with solids which are being considered for use as inert material in a fluidized-bed fluorinator are being continued. Pure crystalline aluminum oxide has been shown to be a suitable for use as an inert solid. In duplicate static-bed experiments at a starting concentration of 1.4 percent plutonium and 27 percent uranium the solids were fluorinated for 10 hours at 450°C. The plutonium was 98.5 ± 0.4 percent removed and the concentration of plutonium in the residue was 0.03 percent. The concentration of uranium in the residue was 0.006 percent. These residual concentrations are identical with those found earlier in runs at fluorination temperatures of 350°C. A greater retention

of plutonium in the alumina residue results at a fluorination temperature of 550°C, at which temperature an average of 93 percent of the plutonium is fluorinated from the solid mixtures.

2. General Chemistry and Chemical Engineering

The attainment of lower cost nuclear power depends on the development of more economical methods of manufacture of nuclear fuel materials. Research and development for devising new chemical methods of improving chemical engineering techniques or equipment is being conducted.

The fluid-bed conversion of uranium hexafluoride into uranium dioxide is an alternative method for the production of enriched oxide for nuclear reactor fuels. The process involves either a single-step pyrohydrolysis and reduction using steam and hydrogen simultaneously, or a two-step process using alternatively steam or hydrogen, followed by the other reactant.

A small diameter fluid-bed calciner is being developed for the direct conversion of plutonium nitrate to plutonium dioxide.

Conditions for liquid metal reduction of the oxides of uranium, plutonium, and thorium are being explored. Preparation of refractory uranium and plutonium compounds is being investigated.

a. Preparation of Uranium Compounds - The compound UB_2 was prepared by reaction of dissolved uranium with a large excess of boron in a zinc-9 weight percent magnesium solution for 6 hours at 800°C.

Refractory compounds prepared to date by syntheses in liquid metal media include the following: UC, USi_3, U_3Si_2, UB_2, Pu_2C_3, and CeC_3.

b. Liquid Metal Reduction Studies - The effect of flux/metal ratio on the reduction of uranium oxides by magnesium-zinc solutions was investigated. Using a calcium chloride-magnesium chloride-magnesium fluoride flux (47.5, 47.5, 5.0 mole percent), increasing the flux/metal ratio increased the reduction rate and reduced the uranium loss to the flux.

3. Chemical-Metallurgical Studies

Continued progress is being made in the development of new pyrometallurgical methods for reprocessing nuclear reactor fuels. Important to the development of these methods are fundamental studies of liquid metal systems that are being used as media in the various process steps. Measurements of solubilities in liquid metals of fissionable and fission product elements and of the elements of container materials are yielding information that are basic to the development of new processes. The work that is

being done in delineating the phase diagrams of the uranium-zinc and the uranium-magnesium-zinc systems is providing information of fundamental importance to the development of pyrometallurgical processes for the EBR-II fuel cycle. Also of process interest is the information that is being obtained from studies of the distribution of elements between immiscible liquid metal pairs. The evaluation of thermodynamic functions for elements in liquid metal solvents and for the more important solid intermetallic phases is adding to the fund of information needed for a thorough understanding of the processes under study. Effusion studies are yielding useful information concerning binary systems in which one of the components is volatile. The program in calorimetry consists of determinations of heats of formation by means of oxygen bomb calorimetry and fluorine bomb calorimetry. The latter is being used for the combustion of refractory materials and gives promise of overcoming many of the gaps in information that have existed in thermochemistry because of the difficulties that hitherto were encountered in making the necessary measurements.

a. Liquid Metal Solvent Studies - It has been found that over the temperature range 450° to 650°C the solubility of vanadium in zinc ranges from 0.16 to 0.7 weight percent. There is evidence for retrograde solubility above about 690°C, but this has not yet been firmly established. Several new intermetallic solid phases have been found; the most zinc-rich phase is probably VZn_3 which is isostructural with $NbZn_3$. Work is continuing on the characterization of VZn_3 and identification of the other phases in the system.

Crystals of the epsilon zinc-uranium phase ($UZn_{11.5}$) and uranium were compacted by powder metallurgy and subjected to prolonged exposure to elevated temperatures. The delta zinc-uranium phase ($UZn_{8.5}$) formed after four weeks at 600°C, but did not form in three weeks at 500°C.

A 25 weight percent uranium-zinc charge was heated for 17 days at 875°C. The product was identified as the delta phase with residual zinc. Chemical analysis of the delta crystals showed the zinc:uranium atom ratio to be 8.75.

The effect of the addition of magnesium on the distribution of uranium between liquid zinc and lead has been determined. Magnesium additions increase the mutual solubilities of zinc and lead and consequently decrease the distribution coefficient of the uranium. Values of the distribution coefficient (weight percent uranium in zinc-rich layer/weight percent uranium in lead-rich layer) at 650°C varied with the magnesium concentration of the zinc phase in the following manner: K = 226, % Mg = 0; K = 55, % Mg = 1.5; K = 24, % Mg = 2.9; and K = 9, % Mg = 4.3.

The lanthanum-zinc and praseodymium-zinc systems are being investigated by means of the effusion method.

b. Calorimetry - Experimental study of the combustion of titanium in fluorine in the calorimeter has been completed. Preliminary studies of combustion techniques for burning tantalum and niobium in fluorine have been completed. Such studies for vanadium are currently being done.

Experiments for determining the heat of formation of boron nitride are continuing.

Exploratory studies of the combustion of silicon in fluorine have been started.

Preliminary experiments on the combustion of zirconium dihydride in oxygen are complete and a series of calorimetric runs with both ZrH_2 and ZrD_2 are to be started.

F. Advanced Reactor Concepts

1. Fast Reactor Test Facility (FARET)

The general engineering parameters for an experimental facility to test advanced fast reactor cores are being examined to prove the feasibility for such a device.

A conceptual design for a safety rod drive mechanism that can be adapted to a variety of core geometries has been completed. Located at the reactor top, the drive uses a rack and pinion for accurate position control. Pneumatic boost is used to reduce the gravitational scram time.

The versatility demanded of the test facility also requires that the method for changing fuel be adaptable to a variety of core configurations. The proposed layout for handling the FARET fuel elements emphasizes simple manual techniques to gain flexibility.

Still in the headache stage of conceptual design is the problem of making FARET suitable for new experiments which may require different coolants and consequently different structural materials. No easy solution has been found. Of several possible schemes examined, it appears that the cheapest way to make the changeover is simply to replace the original pressure vessel and associated primary piping. Other equipment such as the heat sink, controls, containment and handling devices would be retained.

A series of critical radius computations for various volume fractions of a uranium (93% enriched) carbide fuel, sodium cooled and niobium clad system have been completed for the FARET study. The results are presented in Table XII.

Table XII. Critical Radius Calculations

Volume Fraction UC/Na/Nb	Core Radius (cm)	Core Height (cm)	Diameter: Height Ratio	Volume of core (liters)	Critical Mass kg of U(93% enriched)
50/25/25	42.1	30	2.81	167	916
	24.8	45	1.1	87.1	478
	21.7	70	0.62	103.8	569
40/35/25	49.3	35	2.82	267	1170
	32.2	45	1.43	146.6	645
	26.2	70	0.75	151.2	664
30/45/25	50.6	45	2.25	362.5	1193
	33.6	70	0.961	248	816

A study of the theoretically allowable specific power (Mw/kg) in fuel based on high temperature creep properties of cladding materials such as niobium alloys, molybdenum, Inconel X, Hastelloy and SS 347 is presented in Table XIII. The computations assume that the specific power depends on the thermal stress in the clad. The specific powers are tabulated for a clad having the dimensions of the EBR-II fuel pin.

Table XIII. Allowable Specific Power, Mw/kg

Clad Surface Temperature (°C)	SS 347	Hastelloy	Nimonic 80	Inconel X	Mo-0.5% Ti	Nb
600(1100°F)	1.16	-	2.38	4.72	-	-
650(1200°F)	0.78	1.18	1.85	3.72	22.0	24
700(1300°F)	0.54	-	1.36	2.60	-	-
760(1400°F)	0.14-0.28	0.48	0.72	1.60	9.8	10

2. Direct Conversion Survey for Mobile Systems

As a first approach to find new ways to use nuclear energy for surface transportation, reactors for the production of chemical fuels suitable for either internal combustion engines or fuel cells are being considered.

Most of the abundant chemical compounds which contain elements suitable for large scale fuel production are found in nature in forms which are the result of exothermic reactions; for example, light metal oxides and nitrides. Reduction of these compounds by electrolysis at central station nuclear plants is one obvious way to utilize nuclear energy for the production of chemical fuels.

A potentially more efficient scheme is the direct use of fission heat to make the reverse endothermic reaction. Also, ionizing radiation may enhance reaction rates but probably not enough to increase conversion significantly.

As an interim measure to extend the supply of fluid fuels, the well-known methods for production of gasoline or synthetic gas from coal are possible applications for nuclear heat. Some work in this direction is underway at the Bureau of Mines. But when fossil fuels become scarce (or more valuable for other purposes) it will be necessary to synthesize alternate fuels to sustain mass transportation.

The large energy storage capability that we now enjoy with hydrocarbon fuels - 10.4 kcal/gm for gasoline - can be matched with some of the lighter elements. Three with high oxidation energies are boron, lithium and aluminum. Of these, aluminum is of particular interest because it is relatively abundant in the earth's crust. Boron hydride, on the other hand, is also of interest because of its high energy to weight ratio. Efforts are now directed toward screening the light element compounds for their possible production in reactors and their subsequent utility as power sources for transportation.

3. Compact High Power Density Reactors

A cost estimate has been completed for the equipment required to modify EBR-I to test a compact, 11 liter core. This experiment would supply information on the physics and engineering needed to develop a fast reactor which would use sodium vapor as the working fluid in a direct power cycle.

4. Research and Test Reactor Design

As part of a continuing interest in improved research reactors, some work was done to investigate the use of D_2O reflectors with light water moderated, flux trap cores. The usual objections to such a system are that it requires a high degree of structural integrity and a second independent heat removal system that can make D_2O inventory costs prohibitive. To eliminate these problems a canned D_2O reflector without external circulation was considered. The reflector heat sink in this case was the inlet header bringing cold water to the core. Heat transfer calculations indicate that it is feasible to cool the D_2O by natural circulation although hot spots may be encountered at points of high gamma heat generation, such as beam hole liners.

V. PUBLICATIONS

Papers

CRITICAL HEAT FLUX FOR WATER IN SWIRLING FLOW
R. Viskanta
Letter to the Editor, Nuclear Sci. and Eng. 10, 202-203 (1961)

TEMPERATURE DISTRIBUTION IN COUETTE FLOW WITH RADIATION
R. Viskanta and R. J. Grosh
ARS Journal 31, p. 839 (June 1961)

HEAT TRANSFER BY SIMULTANEOUS CONDUCTION AND RADIATION IN AN ABSORBING MEDIUM
R. Viskanta and R. J. Grosh
ASME Paper No. 61-SA-34

THERMOGRAVIMETRIC DECOMPOSITION OF THORIUM 8-HYDROXYQUINOLATE
C. E. Crouthamel and C. E. Johnson
Talanta 8 (6), 377 (June 1961)

FLUORINE BOMB CALORIMETRY
W. N. Hubbard, J. L. Settle, and H. M. Feder
Pure and Applied Chemistry, 2, 39-44 (1961)

MORE ZIRCALOY PROPONENTS
C. F. Reinke
Letter to Editor under column, Nucleonics 19 (2) 6 (February 1961)

GLASSY MATERIALS FOR NUCLEAR REACTOR APPLICATIONS
J. H. Handwerk, E. D. Lynch and V. K. Moorthy
Bull. Am. Cer. Soc. 40 (4) 269 (April 1961) Abstract

US AND ThS BODIES
P. D. Shalek
Bull Am. Cer. Soc. 40 (4) 271 (April 1961) Abstract

OXIDATION AND PHASE STABILITY IN THE SYSTEM URANIUM OXIDE-LANTHANUM OXIDE
D. C. Hill
Bull. Am. Cer. Soc. 40 (4) 271 (April 1961) Abstract

URANIUM AND URANIUM CARBIDE IGNITION STUDIES
M. Tetenbaum, R. Wagner, L. Mishler, and J. G. Schnizlein
Trans. Am. Nuclear Soc., 4 (1), 89 (June 1961) Abstract

The following papers were presented at the American Nuclear Society 1961 Annual Meeting, June 4-8, Pittsburgh, Pa., and were published in Transactions of American Nuclear Society, Volume 4, No. 1, June 1961.

CRITICAL EXPERIMENTS FOR THE PRELIMINARY DESIGN OF THE ARGONNE HIGH FLUX REACTOR, PART A
J. W. L. DeVilliers, Q. L. Baird, J. Juliano, C. N. Kelber, R. Kiyose, and K. E. Plumlee

CRITICAL EXPERIMENTS FOR THE PRELIMINARY DESIGN OF THE ARGONNE HIGH FLUX REACTOR, PART B
J. O. Juliano, C. N. Kelber, and K. E. Plumlee

CONCEPTUAL DESIGN OF A COUPLED FAST-THERMAL STEAM SUPERHEATING REACTOR
R. R. Rhode, R. Avery, W. V. Dewey, and B. J. Toppel

LEAK TESTING OF EBR-II FUEL RODS
A. P. Grunwald

ASSEMBLY, SODIUM BONDING, AND BOND TESTING EBR-II FUEL RODS
T. C. Cameron and N. F. Hessler

PRELIMINARY DESIGN OF A BASIC RADIATION EFFECTS REACTOR (BRER)
D. R. MacFarlane, I. Charak, R. R. Rohde, B. J. Toppel, and H. Unger

CONVERGENCE OF TRANSPORT SOLUTIONS FOR THIN SLAB CELLS
D. Meneghetti

FOIL ACTIVATION DATA HANDLING WITH AUTOMATIC COUNTERS AND A HIGH SPEED COMPUTER
K. E. Plumlee and M. T. Wiggins

IMPROVING THE DYNAMIC BEHAVIOR OF ION-CHAMBER CURRENT AMPLIFIERS
C. E. Cohn

REFLECTED REACTOR KINETICS
C. E. Cohn

HEAT TRANSFER TO SUPERHEATED STEAM
J. B. Heineman

The following material has been contributed to the "Reactor Handbook" 2nd Edition, Volume II, Fuel Processing, Eds. S. M. Stoller and R. B. Richards.

FLUORIDE AND OTHER HALIDE VOLATILITY PROCESSES (Chapter 6)
R. K. Steunenberg and R. C. Vogel

PYROMETALLURGICAL PROCESSES (Chapter 7)
H. M. Feder and I. G. Dillon

GASEOUS WASTE (Chapter 13)
W. A. Rodger, B. S. McCauley and R. A. Keeler

LIQUID WASTE TREATMENT AND DISPOSAL (Chapter 14)
W. A. Rodger and P. Fineman

SOLID WASTES (Chapter 15)
W. A. Rodger and R. A. Keeler

ANL Reports

ANL-6067 DESIGN OF THE ARGONNE LOW POWER REACTOR (ALPR)
Edited by Eberhard E. Hamer

ANL-6242 A GEOMETRY CORRECTION FOR PLANE SOURCE AGE MEASUREMENTS IN HYDROGENOUS MEDIA
Edgar F. Bennett and Jack Haugsnes

ANL-6325 STEAM-WELDED TUBULAR FUEL ELEMENT FOR USE IN CP-5
L. C. Hymes

ANL-6340 PREPARATION OF HIGH-DENSITY, SPHERICAL THORIUM OXIDE PARTICLES WITH UP TO 10 ATOM PERCENT URANIUM
C. E. Crouthamel, W. G. Knapp, S. B. Skladzien, and J. W. Loeding

ANL-6347 THE PERFORMANCE OF A MOTOR, A SWITCH, AND TWO TYPES OF PRESSURE PICKUP IN A HIGH-GAMMA-FLUX ENVIRONMENT
J. E. Ayer and G. J. Pokorny

ANL-6350 DESIGN AND DEVELOPMENT REPORT ON TREAT CONTROL ROD DRIVE II
R. V. Batch

ANL-6365 FLUX AND TEMPERATURE PEAKING EFFECTS RESULTING FROM THE USE OF SEAMED FUEL TUBES IN CP-5
D. P. Moon