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REACTOR ENGINEERING DIVISION QUARTERLY REPORT

SECTION II

April, May, June, 1956

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TABLE OF CONTENTS

	<u>Page</u>
PART A. ADVANCED WATER REACTOR PROGRAM	
SUMMARY	7
I. EXPERIMENTAL BOILING WATER REACTOR (EBWR)	10
A. Reactor Design and Evaluation	10
1. Core	10
Water Corrosion Developmental Tests	10
2. Control Rods and Drive Mechanisms	12
3. Pressure Vessel and Thermal Shield	12
4. Shielding Structure	17
a. Reactor Vessel Pipe Shielding	17
b. Compressive Strength Tests on Heavy Concrete with Steel Ball Aggregate.	17
B. Status of Construction	18
1. Production and Corrosion Testing of Fuel Plates	18
2. Control Rods and Drive Mechanisms	20
3. Pressure Vessel	21
(1) Shop Hydrostatic Test	22
(2) Helium Leak Test.	23
(3) Closure Details after Shop Pressure Tests	24
(4) Miscellaneous	25
4. Hold-down and Shielding Components	26
5. Steam Plant and Auxiliary Systems	27
6. Power Plant and Service Buildings	28
II. SUPPORTING AND ALTERNATE DESIGN RESEARCH AND DEVELOPMENT	31
A. Stability of Bubbling Reactors.	31
1. General Theory	31
2. Evaluation of Constants	33
3. Lower Order Theories and Results	34
4. Conclusion	36
B. Two-Phase Pressure Drop.	36
C. Effect of Oxygen Addition on Water Decomposition	38

TABLE OF CONTENTS

	<u>Page</u>
D. Fuel and Cladding Materials.	41
1. Effect of Fused Salt Treatment on Corrosion Resistance of Zircaloy 2	41
2. Corrosion of Tests on Zircaloy 2 Ingot Samples.	41
3. Aluminum Alloys	42
a. Dynamic Corrosion Tests	42
b. Boiling Corrosion Tests	44
c. Corrosion of Spot-welded 2S Aluminum	47
d. Corrosion - Irradiation Tests.	47
4. Urania - Thoria Fuel Materials	49
a. Solution - Irradiation Tests	49
b. Fabrication of BORAX-IV Fuel Elements	57
E. Control and Thermal Shield Materials	57
1. Corrosion of Boron-Stainless Steel	57
2. Corrosion of Miscellaneous Control Materials.	61
a. Rolled Hafnium	61
b. Boron Carbide Solution Tests	61
c. Aluminum-Nickel Jacketed Cadmium	61
F. Structural Material Corrosion Tests	61
G. Control Rod Drive Mechanisms.	63
1. Hydraulic	63
2. Magnetic Jack	63
H. Argonne Water Loop at MTR	63
1. Operation	63
2. Examination of Second Irradiated Tube Tip	64

3
TABLE OF CONTENTS

	<u>Page</u>
PART B. FAST BREEDER REACTOR PROGRAM	
SUMMARY	69
EXPERIMENTAL BREEDER REACTOR-II (EBR-II)	72
A. Physics Calculation of EBR-II Design Study No. 2	72
B. Irradiation Tests of EBR-II-Type Fuel Elements	75
C. Iron-Uranium Compound Formation at Sodium-Wetted Uranium-Stainless Steel Interfaces	78
D. EBR-II Primary Sodium System Piping and Insulation. . . .	79
1. Insulation Requirements	79
2. Calculation of Transient Thermal Stresses	80
E. EBR-II Control Rod Mechanism - Mark-A.	83
F. Bellows Seal and Sodium Condensation Test Apparatus . . .	83

LIST OF FIGURES

<u>Number</u>	<u>Title</u>	<u>Page</u>
1	Corrosion in Zirconium End Plug of Coupon from Plate No. 153 After 2490 hr in 750F Steam at 1500 psig	11
2	Schematic of EBWR Thermal Shield Experiment	13
3	Direct Thermal Neutron Flux in Air Gap Surrounding Pressure Vessel vs Thickness of Water Gap behind Thermal Shield	14
4	Effect of Water Gap Thickness behind Thermal Shield and Thermal Shield Material on Thermal Neutron Flux at Pressure Vessel Wall	16
5	Macro and Microstructures of Fuel Plate Surface Defect after Pickling and Corrosion Testing for 2 Weeks in 550F Water.	19
6	Effect of Changing Test Environment on Corrosion of Aluminum-Nickel Alloy (M-388)	44
7	Effect of Change in Gas Condition Corrosion of Aluminum-Nickel Alloy (M-388) Exposed for 990 hr in 500F Water . .	45
8	Pits Originated by Inclusions in Aluminum-Nickel Clad (X50)	46
9	Corrosion-Irradiation of Aluminum-Nickel Alloy (M-388) in the Argonne Water Loop at MTR	48
10	Representative Areas of Aluminum-1% Nickel Samples Exposed in Argonne Water Loop at MTR (MTR-W-7)	50
11	Magnified Views of Al-Ni Encapsuled Urania-Thoria Fuel Pin Irradiated in CP-5 Autoclave Facility (Run No. 22)	52
12	Temperature Gradient Through Urania-Thoria Fuel Pin (CP-5 Irradiation-Run No. 22)	53
13	Stress Distribution in Urania-Thoria Fuel Pin Irradiated in CP-5 Autoclave Facility. (Run No. 22)	55

5

LIST OF FIGURES

<u>Number</u>	<u>Title</u>	<u>Page</u>
14	Urania-Thoria Pellets Irradiated in CP-5 Autoclave Facility (Run No. 22)	56
15	Type 347 In-Reactor Tube Tip No. 2 Irradiated in Argonne Water Loop at MTR	65
16	Bend Tests of Specimens From Type 347 Tube Tip Irradiated in Argonne Water Loop at MTR	67
17	EBR-II Plant Arrangement.	69
18	EBR-II Facility	70
19	Surface Appearance of EBR-II Type Fuel Elements Irradiated in CP-5 (Before Decanning)	76
20	Surface of Fuel after Removal of Cladding	77
21	Calculated Temperature Profiles through Inner Tube Wall	82
22	Air Cylinder Hydraulic Shock Absorber	84
23	Measured Scram Velocity vs Displacement for Mark-A Control Rod with Constant Cylinder Pressure of 85 psig . .	85
24	Displacement vs Time for Scrammed Mark-A Control Rod with Constant Cylinder Pressure of 85 psig	85
25	Measured Scram Velocity vs Displacement for Mark-A Control Rod with Cylinder Pressure Decreasing from 85 psig (start) to 23.6 psig (end of stroke)	86
26	Measured Displacement vs Time for Scrammed Mark-A Control Rod with Cylinder Pressure Decreasing from 85 psig (start) to 23.6 psig (end of stroke)	86
27	Bellows Seal and Sodium Condensation Test Apparatus . . .	87

LIST OF TABLES

<u>Number</u>	<u>Title</u>	<u>Page</u>
I	BF ₃ Counter Readings in Air Gap	14
II	Compressive Strength of Concrete Shield Specimens	18
III	Summary of Water Decomposition Experiment - Run No. 3	40
IV	Effect of Fused Salt Treatment on Corrosion of Zircaloy 2 in Static Degassed Water at 680F	41
V	Corrosion of Zircaloy 2 Ingot Samples in Static Degassed Water at 550F	42
VI	Summary of Corrosion-Irradiation Tests on Aluminum- Nickel Alloy (M-388)	48
VII	Dynamic Corrosion of Boron-Stainless Steel in Degassed Water at 550F	58
VIII	Static Corrosion of Boron-Stainless Steel	59
IX	Static Corrosion of Welded Boron-Stainless Steel.	60
X	Static Corrosion of Miscellaneous Structural Materials	62
XI	Dynamic Corrosion of Miscellaneous Structural Materials	62
XII	System Characteristics	80

REACTOR ENGINEERING DIVISION QUARTERLY REPORT

SECTION II

April, May, June, 1956

PART A. ADVANCED WATER REACTOR PROGRAM

SUMMARY - J. M. West

Experimental Boiling Water Reactor (EBWR)

Production of fuel plates for the reactor by the Argonne Metallurgy Division began during March. About 400 plates have been made to date.

Corrosion testing of EBWR plates revealed small spots of white corrosion product on the Zircaloy 2 surfaces. Production was suspended while this was investigated. The spots were found to occur at small, raised, granular-like areas on the surface. Chemical analyses showed these raised areas to contain significant amounts of iron. At present the most feasible explanation appears to be that iron diffused from the steel can to the Zircaloy 2 cladding during the hot rolling operation. The spots containing iron were affected less than the other parts of the surfaces by the acid pickling bath and therefore remained as raised areas after pickling. The presence of iron increased the corrosion rates of these spots in the autoclave.

This difficulty with local corrosion is being corrected by carefully scraping off the raised spots before pickling. Plates which had already been pickled once will be scraped and re-pickled. Scraping alone does not eliminate the corrosion.

To date only one cladding failure has occurred during corrosion testing of about 200 EBWR plates. Plates were exposed to 550F water for a period of two weeks. Welding of fuel plates to Zircaloy 2 side plates to form fuel assemblies has not begun.

A small-scale fuel assembly containing plates made by EBWR production techniques has been exposed to a burnup of 0.35% of uranium atoms in the Argonne Water Loop at MTR. No evidence of failure has been noted. The assembly will be removed for examination during the next quarter.

Because a high oxygen content was found in the steam from the BORAX-III reactor and may lead to excessive corrosion, the EBWR turbine has been dismantled and portions of it are to be nickel plated (0.005 in. thick) by the Kanigen process. The portions to be plated will be the main turbine casing, the diaphragms, and the gland seal casings. The rotor will

not be plated because of the complexity of its structure. Carbon steel steam piping will also be plated. There appears to be barely adequate time to accomplish the plating before these components are required in the construction sequence.

Laboratory tests have shown that the rate of evolution of hydrogen and oxygen is indeed greater when boiling water is exposed to radiation than when the water is static. There now appears to be little doubt that the oxygen content of steam from boiling reactors will be higher than desired. Preparations are being made for tests to evaluate catalytic recombination of hydrogen and oxygen in steam before it reaches the turbine.

Except for the top closure and gaskets the reactor pressure vessel was completed and shipped by Babcock and Wilcox. Following installation of the heavy reactor support and containment beams, the vessel was lowered into place in the reactor shield. Hydrostatic tests made prior to shipment showed excessive leakage through the gasketed closure. Further developmental work and testing, aimed at reducing leakage and preventing scarring of the head by the gaskets, is being carried out by Babcock and Wilcox.

General construction work on the plant continues. Many of the major mechanical components have been set in place and piping installation is under way. Selective overtime work in limited amounts is authorized whenever appreciable savings in construction time can be realized.

Pumps have been ordered for conversion of EBWR to forced circulation after an initial period of operation with natural circulation. Piping and other forced circulation equipment designs are being made.

All control rod drive mechanisms have been fabricated and tested under conditions closely simulating EBWR operation. All drives are ready for installation when other construction reaches a stage where this is feasible.

Fabrication of nuclear instrumentation is essentially complete.

Concrete shielding blocks containing heavy aggregate for use in pipe tunnels are being made at Argonne.

BORAX-IV

About 500 ft. of tube-plates made of aluminum-nickel alloy (M-388) have been extruded satisfactorily on an experimental basis. A trial batch of 40 kg of ThO_2 pellets has been made, and production for the BORAX-IV core can begin shortly.

9

Some small unbounded areas have been observed in sample ThO_2 tube-plate fuel elements. Lead apparently does not always fill the gap between pellets and tube when the gap is very thin. Methods of centering pellets in the tubes to provide a uniform gap are being investigated.

Research and Development

Additional corrosion test results on aluminum-nickel (M-388) alloys show less difference between in-pile and out-of-pile samples than observed in previous tests. Radiation, however, still appears to reduce the corrosion rate significantly.

A boiling corrosion test on M-388 alloy was run for 661 hr at 422F and a heat flux of 25,000 Btu/(hr)(sq ft). The average corrosion rate was 2.8 mils/yr. Pitting corrosion occurred at points where inclusions were noted at the beginning of the test.

Samples of 2S aluminum which had been spot welded showed no accelerated corrosion in the weld areas after 3343 hr at 420F.

A test fuel element containing four ThO_2 (and 10% enriched UO_2) pellets (0.250 in. dia. x 0.48 in. long) inside an aluminum tube was irradiated to about 0.5% burnup of the metal atoms. Lead was used as a bonding agent. The temperature and pressure were approximately 500F and 800 psig, respectively. A small hole was drilled in the tube prior to irradiation. Post-irradiation examination revealed that the capsule had increased about 9% in length. One end of the capsule had increased very slightly in diameter. The drilled hole had been sealed by extrusion of lead. All pellets had typical thermal stress cracks but had not changed dimensionally. The distortions observed in the capsule may have been due to gas pressure buildup. The strengths of aluminum and lead are very low at a temperature of 500F.

A control rod drive mechanism of the magnetic jack-type has been operating satisfactorily under simulated EBWR conditions.

Work on hydraulic control rod drive mechanisms has been discontinued temporarily because of the pressing nature of other work.

I. EXPERIMENTAL BOILING WATER REACTOR (EBWR)

A. Reactor Design and Evaluation - L. W. Fromm

1. Core

Water Corrosion Developmental Tests - C. R. Breden

Pre-production Plate Coupons

Twenty-four coupon samples representing Zircaloy to Zircaloy bond areas from the trimmed off ends of pre-production development plates have been tested for 2757 hr in 750F steam at 1500 psig. Although all samples show white or grey areas, actual pitting or chalking appears to be limited largely to a very few areas in which non-corrosion resistant zirconium had been used for end plugs (Fig. 1). Since the ratio between corrosion rate in 750F, 1500-psig steam and 550F water is about 40 to 1,¹ this is equivalent to about 13 years' exposure in 550F water.

Production Plate Coupons

Twelve coupons representing nine production plates are being corrosion tested in 750F, 1500-psig steam. After two weeks there were a few small white and grey areas, and the core areas showed large numbers of inclusions. By the end of 5 weeks most of these inclusions were showing white corrosion products. After 6.6 weeks there was not much additional change. If the 40 to 1 ratio between 750F steam and 550F water is valid, this is equivalent to about 5 years' exposure in 550F water. If the ratio remains valid under actual operating conditions, the outlook is quite good for the use of Zircaloy 2 in the production plates.

EBWR-Type Subassembly

An assembly (No. 6) fabricated by proposed EBWR techniques, but containing one non-acceptable plate and a side plate with a lamination-type crack at the lower end, is being subjected to long-term corrosion tests in 550F water. After the first 72 hr, the plates looked quite free from corrosion product except at spots where the chromium-plated "fingers" of the holding jig had been in contact with the plates during the assembling operation. These spots were white. There were also some areas of white corrosion product in the spot weld areas at the ends of the assembly.

¹K. M. Goldman, D. E. Thomas, "The Corrosion Behavior of Zircaloy 2," WAPD-92 (February 19, 1954).

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FIG. 1
CORROSION IN ZIRCONIUM END PLUG OF COUPON
FROM PLATE NO. 153 AFTER 2490 hr IN
750 F STEAM AT 1500 psig. X25

After an additional 312 hr, there was loose white corrosion product in the "finger" contact area. The white corrosion product in the spot weld area appeared to be less evident. The "lamination" crack in the lower end of the side piece showed the presence of white corrosion product. In other respects the appearance of the assembly was quite satisfactory.

2. Control Rods and Drive Mechanisms

Zircaloy 2 - Stainless Steel Contact Corrosion - C. R. Breden

Static and dynamic tests² on samples of Zircaloy 2 and stainless steel Types 304 and 347 bolted together to study contact effects are continuing. No contact or crevice effects were detectable on dynamic samples after 10,257 hr in water flowing at about 18 fps. Water temperatures during different periods in this test were 600F, 550F and 525F, depending on the requirements of the boiling corrosion facility attached to the test loop.

Static samples in air-saturated and in degassed water have been tested at 500F for 11,013 hr, and at 600F for 10,195 hr. The Zircaloy 2 is showing tiny pits associated with white corrosion product, uniformly distributed on the exposed areas. This does not appear to be associated with contact or crevice effects.

3. Pressure Vessel and Thermal Shield

Experiments on EBWR Thermal Shield Mock-up - R. J. Rickert, A. D. Rossin

Shielding calculations of thermal neutron flux based on diffusion theory are not adequate in regions where small thicknesses of moderator are sandwiched between heavier absorbing materials. This case arises in thermal shielding for water-cooled power reactors such as the EBWR. Detailed knowledge of the thermal neutron flux in the pressure vessel and thermal shield is needed since these components act as major sources of capture gamma rays. The design of thermal shielding is strongly dependent on the relative amount of heating in the pressure vessel wall from core gamma rays, thermal shield capture gammas and gammas due to neutron capture in the vessel wall.

²Reactor Engineering Division Quarterly Reports: ANL-5571 (July, 1956); ANL-5371 (January 15, 1955).

13

A mock-up of the pressure vessel and part of the shield of the EBWR (Fig.2) was installed in the Bathtub Shielding Facility of the Argonne Research Reactor (CP-5). Experiments were performed to determine the relative effects of mild steel and of boron-stainless steel thermal shields on thermal neutron flux at the pressure vessel wall as a function of the thickness of the water gap between the thermal shield and the vessel wall.

LEGEND

A	THERMAL SHIELD	MILD STEEL, 1 in. THICK
B	THERMAL SHIELD	2 wt- % BORON-STAINLESS STEEL, 1 in. THICK
C	STEEL BOX	ONE 2 1/4 in. AND ONE 3/4 in. THICK FACE WITH 6 in. AIR GAP BETWEEN TO SIMULATE PRESSURE VESSEL WALL, AIR SPACE, AND REACTOR WELL STEEL WALL
D	LEAD SLAB	3 in. THICK
E	LIGHT CONCRETE	THREE SLABS, EACH 4 in. THICK
F	LIGHT WATER	
G	WATER GAP	VARIED BETWEEN 0 AND 1 1/4 in.

FOILS WERE PLACED ON AXIAL CENTERLINE IN AIR GAP, WATER GAP, AND IN WATER AHEAD OF THERMAL SHIELD

FIG. 2
SCHEMATIC OF EBWR THERMAL SHIELD EXPERIMENT

Indium foils and a BF_3 counter were used to measure the neutron fluxes and to determine the cadmium ratios. The foils were standardized against a foil activated in the Argonne Standard Pile. Foils were mounted on Lucite discs for ease in handling, the Lucite effectively simulating the water it displaced. The foils were arranged in stacks and staggered to minimize self-shielding.

Count rates were observed with the counter bare, cadmium covered, and with a cadmium sheet behind the counter to eliminate thermal neutron backscatter from the outer shielding. Table I gives

the count rate of the BF_3 counter in the air gap surrounding the pressure vessel wall. As constructed, the gap in the reactor will be packed with thermal insulation.

Table I
 BF_3 COUNTER READINGS IN AIR GAP

Run	Thermal Shield Material	Water Thickness, in.	Count Rate, counts/min				
			Total	Cd Behind Tube	% of Total due to Back-scatter	Cd Enclosed Tube	Cd Ratio
1	None	-	76023	58629	0.23	19196	3.96
2a	Mild Steel	1	66423	50018	0.25	22000	3.02
3a	Mild Steel	1/2	60030	44366	0.26	21426	2.80
4a	Mild Steel	0	50911	35333	0.31	18622	2.73
2b	B-SST	1-1/4	58406	42976	0.26	20094	2.91
3b	B-SST	3/4	49872	36036	0.28	19504	2.56
4b	B-SST	1/4	43244	29848	0.31	17768	2.43

The direct thermal flux (below cadmium cut-off) in the air gap was determined by subtracting the BF_3 counter reading, cadmium enclosed, from the reading with cadmium behind the counter tube. The latter reveals that a substantial contribution is due to scattering from the concrete and lead. The data obtained are plotted in Fig. 3 as a function of the water thickness behind the thermal shield for steel and boron-stainless steel slabs.

FIG. 3
DIRECT THERMAL NEUTRON FLUX IN AIR GAP AROUND PRESSURE VESSEL WALL VS THICKNESS OF WATER GAP BEHIND THERMAL SHIELD

15

The thermal flux as determined from foil activations is plotted for various thermal shield materials in Figs. 4(a) and 4(b) for one-inch and one-half-inch water gaps, respectively.

The only check against EBWR shield calculations which can be made at this time is for the case with no thermal shield. The thermal shield in EBWR was designed for D_2O core conditions, hence was never included in H_2O calculations. The calculated³ thermal flux four inches in front of the pressure vessel wall is 2.4×10^{12} , and in this experiment 3×10^5 , as shown in Fig. 4(a), or a ratio of 8×10^6 . The calculated flux in the air gap is 1.8×10^{10} which, when divided by 8×10^6 , yields 2.2×10^3 for the air gap flux to be expected in this experiment. The observed value in the air gap is 3×10^2 , which is a factor of seven below calculations. This result is reasonable since calculations represent pessimistic assumptions concerning the geometry (the Bathtub employs a finite diameter beam hole while calculations used one-dimensional techniques) and removal theory.

The conclusions which can be drawn from the results of this experiment are as follows:

- (1) The use of steel or boron-steel as thermal shielding in light-water reactors has very little effect in reducing the neutron flux at the pressure vessel wall. The water displaced by the steel does almost as well.
- (2) The primary advantage of boron-stainless steel over ordinary steel for thermal shielding is the reduction of capture gammas from the thermal shield since the neutron flux in the shield is sharply reduced by the boron.
- (3) Reduction of the water thickness behind the thermal shield (from 1 in. to 1/2 in.) reduces the thermal flux in the air gap outside the pressure vessel about 20% per 1/2 in. for a steel thermal shield, and about 30% per 1/2 in. for boron-stainless steel.
- (4) The cadmium ratio in the air gap is small compared to that observed in moderating materials. Much of the scattered radiation in the air gap may be intermediate energy neutrons which are not fully described by removal theory.

It appears that for a light water reactor, boron-stainless steel does not offer enough advantage over ordinary stainless to warrant the

³M. Grotenhuis, J. W. Butler "Experimental Boiling Water Reactor (EBWR) Shield Design," ANL-5544 (March 1, 1955).

16

FIG. 4
EFFECT OF WATER GAP THICKNESS BEHIND THERMAL SHIELD AND THERMAL
SHIELD MATERIAL ON THERMAL NEUTRON FLUX AT PRESSURE VESSEL WALL

RE-7-19101-B

17
increased cost. For a D₂O reactor this conclusion will change since the D₂O behind the thermal shield will not build up the thermal flux as light water does after the boron has depressed it.

4. Shielding Structure - E. H. van Zylstra

a. Reactor Vessel Pipe Shielding

At points up the side of the reactor vessel, where pipes must penetrate the shielding and attach to the vessel, large stepped rectangular holes are left in the bulk shielding by inserting permanent forms made up from 1/4-in. thick carbon steel plate prior to pouring the bulk shielding. These permanent forms are welded at their inner end to the 3/4-in. thick inner shield form. Pipes are brought out from the vessel through holes in the inner shield form, and, after going through a triple offset to eliminate direct paths for radiation streaming, the pipes pass through the outer concrete retaining wall through larger pipe sleeves set in the concrete. The larger pipes are welded at one end to the 1/4-in. thick permanent forms, and at the other end (through bellows or closure plates) to the process piping. Shielding is provided in these pipe tunnels in the form of concrete blocks.

The pipes are insulated with Type 430 stainless steel wool strapped in place with stainless steel straps. In order to provide space around each pipe for downward movement of both pipe and pressure vessel due to thermal expansion, sectional troughs have been designed. The straight portions of these troughs are made from 1/4-in. plates bent into an "U" shape. Covers for these straight portions are rectangular flat plates with angle iron stiffeners on the under side. The corners at the pipe offsets are made from "L"-shaped plates with the two legs bent at 90 degrees and welded. These corners have flat plate, angle iron braced, covers to fit. The troughs are not made airtight since they are within the airtight steel tunnel forms. They are installed by assembling the various trough pieces on top of the shielding blocks after the blocks have been placed to a level approximately 2 in. below the bottom of the insulation on the various pipes.

b. Compressive Strength Tests on Heavy Concrete with Steel Ball Aggregate - E. H. van Zylstra

The fortuitous availability of steel balls (5/8 in. dia.) prompted a series of tests to determine the effect of partial or total substitution of the balls, in lieu of steel punchings, on the compressive strength of the heavy concrete biological shielding of the EBWR.

As evidenced by Table II, no serious reduction in strength was effected by the substitution; therefore, steel balls were used as part of the aggregate for the heavy concrete.

18

Table II

COMPRESSIVE STRENGTH OF CONCRETE SHIELD SPECIMENS

Concrete Mix	Curing, Days	Density, lb/ft ³	Compression Strength, psi
Magnetite & Punchings (Jones & Laughlin fine aggregate)	7	268	1586
Magnetite & Punchings (National Lead fine aggregate)	18	298	3272
	28	293	4597
Magnetite & Punchings (National Lead fine aggregate Punchings: $\frac{1}{2}$ regular, $\frac{1}{2}$ steel balls)	7	293.4	3572
	7	294.0	3660
	14	287.8	4033
	14	289.3	3961
	28	293.6	4244
	28	285.8	4632
Magnetite & Steel (National Lead fine aggregate Coarse aggregate: steel balls)	7	292.0	2971
	7	291.7	2989
	14	296.0	3360
	14	294.2	3660
	28	262.8	3801
	28	304.2	3819

B. Status of Construction - L. W. Fromm, A. S. Jameson1. Production and Corrosion Testing of Fuel Plates -
C. R. Breden, C. K. Soppet

Fabrication of fuel plates by the Argonne Metallurgy Division has been under way since March. A total of 409 plates have been delivered to the Reactor Engineering Division for corrosion tests. An additional 265 plates have been rolled and are in the final stages of completion. The casting of fuel billets for the 888 plates needed for the initial core loading is essentially complete.

All of the production plates corrosion tested have shown one or more small spots which had white corrosion product on the surface (Fig. 5). A characteristic of these spots was that they were raised from the surface and were granular in appearance. In all cases examined metallographically, penetration by corrosion extended less than 0.0005 in. below the surface of the plate [Fig. 5 (b)]. The spots are raised because they are affected less by

19

MICRO 19419

(c)

X250

SECTION THROUGH DARK AREA SHOWING
SLIGHT SURFACE CONTAMINATION

MICRO 19419

(b)

X250

SECTION THROUGH LIGHT AREA SHOWING
DEPTH OF CONTAMINATION

MACRO 19422

(a)

X9

FIG. 5
MACRO AND MICROSTRUCTURES OF FUEL PLATE
SURFACE DEFECT AFTER PICKLING AND CORROSION
TESTING FOR 2 WEEKS IN 550 F WATER

the pickling solution than is the remainder of the cladding. Some of the raised spots have been scraped off and subjected to chemical analysis. The results showed as much as 0.6% iron. The explanation proposed for the presence of these defects is that they represent local areas where iron diffusion from the can into the Zircaloy 2 cladding has occurred during rolling. The high percentage of iron in the upper surface of these spots accounts for the increased corrosion. It has been found that if these corroded raised spots are carefully scraped to remove the surface material high in iron, and then pickled, white corrosion product does not reappear on subsequent corrosion testing. Scraping without pickling is not sufficient. All of the first lot of plates (about 300) are being scraped, vapor blasted, and pickled before corrosion testing again.

The integrity of the cladding process as indicated by autoclave tests seems to be quite good. Only one failure, one of the first production plates, has occurred during the test period, an aggregate of approximately 45,000 plate-hours.

2. Control Rods and Drive Mechanisms - C. F. Bullinger

The nine control rod drives to be used in operation of EBWR have been proof tested and are stored pending installation on the reactor. Each unit was connected to an operating control rod and tested for approximately 240 hr at 600 psia.

Six of the units experienced varying degrees of vibration on the driven down stroke at the beginning of test. This vibration, at certain locations on the stroke, was traced to a minute difference in pitch diameter between the male and female acme threads of the lead screw and carriage nut. This difference resulted in improper lubrication and consequently high drag in the affected regions. Continuous operation and re-lubrication of the threads has proved effective in wearing the nut in to match the lead screw, with complete elimination of the vibration.

All valve seats on the control rod connector have proved watertight under static head of reactor water level height. Removal of the control rod, seal, and drive without draining the test vessel has been demonstrated successfully in repeated tests.

The Zircaloy 2 follower of one control rod was observed to have warped more than 1/2 in. out of line in two places. These areas were highly burnished as were the contacting areas of the Zircaloy 2 rod guide in the vessel. There was no evidence of fretting corrosion or change in the rod drive mechanism performance characteristics.

The seal assemblies and nylon guide bushing have shown good ability to absorb transport corrosion products without impairment of

21
performance. The seal rings have exhibited self-cleaning tendencies as indicated by a decrease in leakage rate by 50% or more after standing idle over a weekend. After several scrams the leakage rate returns to normal.

As a final check on the effect of irradiation on the FM-10001 nylon guide bushing, a spare unit was placed in CP-5 and exposed to a thermal neutron flux of 10^{12} n/(cm²)(sec) for a period of 48 hr. The only visible effect was a slight change of color. Resilience was unaffected. The unit was then subjected to a hard gamma dosage of 250,000,000 roentgens. The color changed to tan, and the surface was slightly sticky. The stickiness disappeared after about 5 days exposure to atmosphere. A growth of approximately 0.0015 in./in. on all surfaces was observed. When placed in a seal housing and operated in the usual manner, however, the performance was the same as for the unirradiated nylon. A pessimistic estimate of the equivalent radiation received in the EBWR would allow a minimum lifetime of 10 years for the bushing, or equal to the rest of the drive.

3. Pressure Vessel - N. Balai

The construction of the EBWR pressure vessel continued to present difficult fabrication problems which required additional tests to prove the reliability of the fabrication procedures. Although the vessel was shipped on May 18 for installation inside the reactor biological shield, the closure remains a problem which must be solved in the field. The vessel was erected on its support on June 16. Unforeseen difficulties included:

a. Interpretation of dye-check results, particularly on the interior cladding. The cladding was known to be sound from the bond-integrity tests on the clad plates prior to forming operations. (The cladding was soap tested while maintaining a 900-psi nitrogen pressure in the crevice between the base material (SA-212 Grade B) and the stainless steel Type 304 spot welded sheet.) A second bond-integrity test after the final stress relief of the vessel revealed numerous small cracks and pin holes in repaired regions where the spot cladding had been removed for internal attachments and replaced by weld metal. In too many cases, dye-checking did not reveal the flaws. For practical purposes, the gas-soap test, where it can be applied, is much more reliable than the penetrant dye method.

b. Interpretation of the corrosion resistance of the cladding. Shop heat-treating records show that portions of the cladding have been heated to 1150/1200F three and four times for periods up to 3 hr/cycle. In addition, the clad lower head plate has been heated to the forging temperature of SA-212 Grade B plate for spinning. Samples of the clad lower head plate have been heat treated in accordance with the fabrication history and are currently under test in static water at 550F.

c. Interpretation of weld X-ray data. All welded joints with the exception of small nozzle perforations in the cylindrical shell were X-rayed. All films show unusually sound carbon steel weldments. Inconel welds between carbon steel and stainless steel Type 304 show very fine porosity (ASME Code definition), extremely uniformly dispersed. The integrity of all nozzle welds (Inconel and carbon steel) was established in a helium leak test at 800 psi.

(1) Shop Hydrostatic Test

Preparations for the 1200-psig hydrostatic test were started on May 5. With the vessel in the horizontal position, the problem of retaining the fragile Flexitallic gaskets was solved by cementing them into the grooves with molasses. Two outer gaskets were ruined when the outer edge scraped a bolting stud. Both gaskets were compressed uniformly from the 3/16-in. thickness (nominal) to 0.150 in., at which point the flanges made metal to metal contact without a measurable elongation of the stud bolts. The bolts were then stressed to 20,000 psi - corresponding to bolt elongation of 0.009 in. Gasket seating was checked with a 15-psig air test, at which time it was discovered that the tape covering the leak-off connection was slightly puffed up. A soap test immediately established a minute leak. Air pressure was then reduced and the vessel filled with demineralized water having a resistivity of 6 megohm-cm.

Critical sections of the vessel were white-washed to prove the strength of the design. They were:

- (a) A 12-in. stripe across the diameter of the flat head.
- (b) An area 30 in. square including the upper ring forging, transition knuckle, weld zone and portion of shell.
- (c) A 30 in. x 24 in. area including the lower head, knuckle radius, weld and portion of shell.

While filling operations were in progress on May 8, thickness gage surveys were made between the flange faces of the closure since it was noted previously (by reflections of lighting) that the cover plate appeared to be slightly waved and crowned. The separation was a fairly uniform 0.002 in. with an occasional gap of 0.005 in.

A test pressure of 1200 psi was maintained for a 2-hour period to permit a number of detailed examinations of the entire vessel for leaks. An inconsequential leak at a temporary screwed plug connection in addition to closure leakage was found. Two 30-minute samples

showed that the closure leaked at a rate of 354 cc/hr and 326 cc/hr. The collected samples were colored by the molasses used for sealing the gaskets in the groove. Thickness gage measurements showed the flange separation had increased to 0.010 in. with no additional elongation of bolting. None of the white-washed areas showed signs of distress and it was concluded that the design met ASME Code requirements for strength. The hydrostatic test was witnessed by the Hartford Inspector.

After the pressure was released (1180 psig final pressure), thickness gage measurements established that the flange separation varied from 0.005 to 0.008 in. (Additional thickness gage measurements on the following day were approximately the same as in the original condition. A number of bolts were understressed as determined by elongation measurements.)

These data indicated that the closure gasket had failed by local buckling of the stainless steel ribbon winding of the gasket aided by the rubbing action between elastic expansion of the ring forging in contact with a static cover plate. The vessel was emptied and made ready for the Helium Leak Test which was designed to prove the integrity of pressure vessel (welds and plate) with only the closure excepted. Leakage through the closure was to be measured. Prior to pressurizing with helium, the studs were again torqued up to the 20,000 psi stress level (0.009 in. bolt elongation).

(2) Helium Leak Test

With the pressure vessel in the horizontal position, a Pliofilm housing was constructed around the vessel to collect helium leakage through welds and plate imperfections, but not including the closure and the lower head nozzle connections. The bag was attached to the steel walls and the painted concrete floor with pressure-sensitive tape. At the fabricator's request, all welds inside and outside the Pliofilm housing were individually bagged and probed as a precautionary measure for the exact localization of leakage. Leakage through the gasket was metered through a Fischer and Porter Flowrater. The 1200-psi helium test was relaxed to 800 psi because of the gas procurement problem. The leakage test was started on May 13, 1956 with the vessel pressurized to 850 psi and was discontinued approximately 48 hr later. The atmosphere inside the Pliofilm housing contained no detectable concentrations of helium.

A Consolidated Engineering Type 24-101A mass spectrometer was used for the detection of helium gas. It had been previously calibrated in the laboratory and was known to be able to detect a leak of 0.5×10^{-6} cc/sec at a differential pressure of 15 psia. The calibrated leak (0.5×10^{-6} cc/sec) produced a 70% scale reading in approximately 45 min.

Helium leakage through the inner gasket of the flanged head closure was 9600 cc/min, which is far in excess of the desired equivalent D₂O steam leakage rate of 1.3 cc/hr. Atmospheric in-leakage through the outer gasket was not measured. Gas pressure (at a final pressure of approximately 820 psi) was discharged to the atmosphere and the cover plate was removed for a visual examination of the closure joint.

(3) Closure Details After Shop Pressure Tests

In removing the cover plate, the inner Flexitallic gasket had to be secured inside the small annulus, by taping to prevent its complete deterioration prior to examination. In contrast, the outer gasket was still firmly cemented into its groove by the molasses used previously to hold both gaskets in place. Both gaskets were found to be permanently distorted in the thickness direction, to have failed by local buckling, i.e., the laminations were "waved" to the left and, alternately, to the right of the nominal arc described by the spiral stainless steel ribbon comprising the core of the gasket. Additionally, the outer gasket had two areas where the ribbons were sheared at an angle of 90 degrees to the nominal arc of the metallic ribbon core.

The flange facing of the cover plate retained the pattern of the inner and outer gaskets and confirmed the slight "waving" and "crowning" observed prior to the installation of the cover plate for pressure testing. In addition, a blurred radial pattern of the gaskets was found to be approximately 1/32 in. wide instead of the 0.010 in. steel ribbon core width. This indicated rubbing or scuffing action across the gaskets and confirmed the flange rotations observed by thickness gage measurements before, during, and after pressure tests. The inside gasket retaining ring was found burnished.

The bottom of inner gasket groove was free of scoring which was to be expected since the gasket completely filled the groove in the compressed condition. This was affirmed by the "dimples" in the gasket at the six intergroove leak-off points. All of the molasses was washed out of the inner gasket groove. No evidence of molasses, as determined by stickiness, was found on the inner gasket. The outer gasket was still sealed into its groove by molasses and was not removed. "Dimpling" of the outer gasket was also observed at the intergroove leak-off points.

From the observations of the performance of the closure under pressure conditions the following conclusions were drawn:

(a) That Flexitallic gaskets are unsuitable for the closure because of their failure by local buckling.

(b) That stainless steel Type 304-jacketed, asbestos-filled gaskets could be expected to perform better than the Flexitallic-type gaskets since flange rotations - under pressure - add to the sealing forces. Three different types of gaskets were ordered from the Akron Metallic Gasket Company:

Type DS - Double Jacket - Asbestos Filled

Type FDS - Full Double Jacket - Asbestos Filled

Type DC - Double Corrugated - Asbestos Filled

They are to be installed in the EBWR facility without benefit of previous performance data since no other test facilities of this size and pressure rating are available.

(c) That cover plate maintenance problems would be extremely difficult as revealed by the cover plate flange facing re-machining process. It was found that the precision required to obtain flatness across the face (to eliminate crowning) and circumferential flatness (to eliminate waving) necessitated new techniques and procedures, i.e., grinding spindles and tool holders were not stiff enough to eliminate chattering. A phonograph finish with a 32-microinch roughness was achieved with an "oil of wintergreen" coolant - but only after three intermediate operations, one of which was an attempt at hand-lapping.

The design of the cover plate gasket facing was modified to eliminate the machining of a radioactive cover plate. Any scuffing or galling is to be transferred to the gasket, which is expendable. A Stellite 6 hard facing was selected and the fabricator (The Babcock and Wilcox Company) is determining welding procedure and machining techniques to be used in finishing the cover plate.

The pressure vessel, sans cover plate, was shipped from the Barberton plant on May 18 and arrived at the Laboratory approximately two weeks later.

An inspection of the vessel after its transportation by the railroad showed evidence of rough handling in transit, despite requests and placarding to the contrary. Rod anchors were found stretched an additional 3 in. and bands were loose. No structural damage was apparent otherwise. The shipping incident illustrates the need for more careful considerations of this problem for future reactor pressure vessels.

(4) Miscellaneous

Prior to shipment of the grid plate, an exact grid plate template was fabricated for reference use for future cores. Anchor bolting and the positions of the control rod passages were located from the grid plate master.

The contract for fabrication of the thermal shield was let to the Superior Steel Corporation, Pittsburgh, Pa. Since the melting and rolling techniques for 60 in. wide plates of 1% boron-stainless steel are not yet developed, the design of the cylindrical section was changed to a 10-piece construction instead of the 5-piece configuration originally envisioned. All of air-melted ingots had been rolled to 1 in. gauge plate by the end of June. Preliminary ladle analyses (verified by check analyses) showed that the 1.0-1.2 wt-% natural boron specification had been met.

4. Hold-down and Shielding Components - E. H. van Zylstra

The components of the hold-down structure (except the hold-down wedges, beams and blast shields) were placed in the power plant building, welded together, and set in final position. The blast shield support brackets and shelves were welded in place to the hold-down columns. The inner shield form was delivered in two halves and was lowered into the reactor opening. The halves were moved out against the hold-down columns and set down on the blast shield shelves for temporary support. The blast shields will be placed later as soon as the inner shield form halves have been lifted off of the blast shield shelves and placed in their proper position on top of the bottom shield plug. The hold-down beams will not be placed until the reactor vessel cover has been received.

The top shield tank and 6 in. thick back-up shield plate assembly was delivered to the building site early in the quarter.

The reactor vessel top shielding segments, support blocks and lead shield forms are being fabricated at Rock Island Arsenal. Some of these segments are now completed and delivered to Plant Engineering where they will be filled with punchings concrete.

The canned insulation segments and pads of steel wool for the reactor vessel cover have been completed.

The bottom shield plug and thimble and inlet tube form assembly was delivered and placed on timber shoring in the sub-reactor room. All tube forms have been adjusted and welded. The asbestos rope insulation has been applied to the forced circulation inlet nozzles on the pressure vessel. It is expected that installation of the bottom shield plug will be completed early in the next quarter.

Prior to placing the reactor vessel the two main beams and one secondary beam of the vessel support structure were welded in place between the hold-down columns. After the vessel was lowered into position the second secondary beam was welded in place. The anti-torsion beams will not be welded in place until the blast shields have been lowered into position.

27

All of the coarse magnetite required for the EBWR shielding has been received from Montana.

The two lead-shield forms for the 12-inch recirculation outlet nozzles have been completed and are scheduled for installation in the EBWR power plant building early in the next quarter.

Application of the 3-in. thick stainless steel wool insulation on the reactor vessel had been started at the end of the report period.

5. Steam Plant and Auxiliary Systems - A. S. Jameson,
V. C. Hall, E. A. Wimunc, C. B. Zitek

Laboratory findings on the BORAX-III reactor indicated a high water decomposition rate.⁴ In order to check possible excessive corrosion anticipated in EBWR due to these findings, the turbine and the 6-inch main steam piping were Kanigen processed. This process deposits a nickel and nickel phosphide (95% nickel) coating of a fairly uniform thickness over the prepared surfaces; a thickness of 0.005 in. was placed on these components. Surfaces which were not to be plated were masked with a special compound.

Parts of the turbine which were plated included the upper and lower halves of the turbine casing, all the diaphragms (22 halves), the Curtis admission shield, the 6 halves of the high-pressure gland seal casings, and 4 halves of the low-pressure gland seal casings. The turbine rotor assembly was not coated because it contained too many dissimilar metals as well as cracks (between blade roots and tops and the rings), neither of which is conducive to good plating.

The demineralized water storage tank has been installed on the main floor of the power plant building. Other items to be installed on the main floor will be moved into the plant as soon as the working schedule permits. These items (on the site) include the turbine, generator, exciter, and the high-pressure boric acid tank.

The following is a list of locations and the items installed at these locations:

⁴Reactor Engineering Division Quarterly Report - Section I, ANL-5571 (July, 1956), p. 83.

Condenser Floor

Condenser, desuperheater, and piping.
Steam Dryer-Emergency Cooler.
Reactor water column.
Air ejectors.
Condenser circulating water pumps.
Shield cooling surge tank.
Surge protection cubicle.

Pump Floor

Start-up heater.
Reactor Feed pumps.
Cooling water pumps.
Fluid recovery system.
Steam by-pass valve oil pumps.
Vacuum tank.

Basement

Coolers and Regenerative Coolers.
Retention Tanks.
Shield Cooler and Pumps.

Basement (Service Building)

Condensate Return Pumps.
Retention Tank and Pump.
Low-Pressure Boric Acid Tank.

Cooling Tower

The cooling tower basin is completed along with the structural redwood skeleton. Piping between the plant and the cooling tower is essentially completed.

6. Power Plant and Service Buildings - A. H. Heineman

The cable exit pans through the power plant shell were installed and welding completed. During subsequent pouring of concrete in other areas, the openings in the 24-in. thick outer envelope which were left for these pans, and for the small temporary opening below grade, were reinforced, formed, and poured. Form work, placing of reinforcing steel, and pouring of concrete continued throughout the quarter.

The 24-in. thick outer wall, the fuel storage pit, the octagonal wall around the reactor, and turbine support columns were completed below the main floor.

29

The main floor slab and turbine support beams were completed in a single pour. Following a 10-day curing period the hold-down columns were lowered into the reactor opening with their footings in the anchorage pockets which had been left in the basement floor. When the columns had been accurately positioned, top and bottom reinforcing rods of the basement slab were welded to the footings, as designed, to obtain full anchorage. The pockets were then filled with concrete.

The 3/4-in. thick steel plates which form the outer portion of the sub-reactor room ceiling around the bottom shield plug, and a permanent bottom form for the punchings concrete in the lower portion of the biological shield have been installed and welding is in progress.

Back-filling and compacting of the fill under the service building was completed and basement floor beams, basement floor slab, basement walls, main floor slab, loading platform, and platform for outdoor electrical equipment were poured. Sequenced with these operations were forming and pouring of the underground concrete ducts for main generator cables, auxiliary transformer cables, and other inter-connecting lines.

Between operations in the power plant and service building, the cooling tower foundations, main and auxiliary slabs, basin walls, and column supports were formed and poured, complete.

All pipe lines, pipe sleeves, conduits, tunnel liners, and anchor bolts which are embedded in concrete were installed in advance of the various pours.

All structural steel support beams and columns were installed in areas in the power plant which are to be floored with steel grating or checkered plate. Erection of the two stairways in the power plant was nearing completion at the end of the report period.

All structural steel work in the service building was completed. The exterior brickwork was started. The pre-cast concrete roof slabs were laid complete preparatory to application of the finish roofing.

Back-filling was brought up to approximately grade level at most points around the entire facility. Prior to final back-filling adjacent to the building walls, a sub-fill of gravel was laid to five feet below grade and soil drain tile installed. The layer of Foamglas insulation (3 in. thick) was then applied to the power plant shell from five feet below grade to a point slightly above grade. The trench above the tile was then filled with coarse crushed stone to about one foot below grade.

The reactor vessel was delivered to the site, moved into the power plant and lowered into position in the reactor opening on its permanent support structure. The reactor vessel is provided with 13 thermocouple blocks welded to the outer surface. These blocks are on a common vertical center line at elevations from a few inches below the top of the vessel neck, at approximately 2 ft-0 in. spacing, to a point near the vessel centerline on the bottom head. Each block has two inclined holes drilled in the top for insertion of dual thermocouples, and one drilled and tapped horizontal hole in the outer face for attaching a protective wireway for the lead wires.

The thermocouples were installed and soldered in place and the lead wires strung to the outside face of the octagon. The wireway was bolted in place, and the lead wires were bound with asbestos cloth to protect them from possible damage by the stainless steel wool insulation on the vessel. The electrical contractor then checked the thermocouple circuits for continuity and grounds. Laboratory personnel then checked each thermocouple for EMF response at room temperature. All thermocouples were found to be intact and operative.

Installation of permanent water and steam supply lines was virtually completed to service building, power plant and cooling tower. Permanent lighting in the power plant is about 75% complete.

II. SUPPORTING AND ALTERNATE DESIGN RESEARCH AND DEVELOPMENT

A. Stability of Bubbling Reactors - J. A. Thie

It has been known for some time that power oscillations in heterogeneous or homogeneous reactors containing bubbles of steam or radioactive gas can occur when one has the proper phasing of the system's delays:

- (1) neutron emission and thermalization times;
- (2) time lag between fission and bubble formation;
- (3) bubble departure time.

Differential equations describing the system and their solutions according to small oscillation theory are given here. A statement of the equations is a statement of the theory's assumptions. It should be noted that coolant temperature and flow variations are ignored. Although a heterogeneous boiling reactor is specifically in mind here, the theory may be applicable to some homogeneous reactors.

1. General Theory

The space-independent linearized kinetics equations for one delayed group (extension to more than one group being trivial) are

$$\ell \frac{dn}{dt} = pn = N_0 (1 - \beta)k + \lambda \ell c - \beta n, \quad (1)$$

$$\ell \frac{dc}{dt} = pc = \beta N_0 k + \beta n - \lambda \ell c, \quad (2)$$

where n , c , and k are small deviations from equilibrium values of N_0 , C_0 , and K_0 (the neutron density, delayed emitter density, and reactivity, respectively). A time dependence of $\exp(pt/\ell)$ is assumed for all variables for the purpose of finding the eigenvalues, p , of the system of differential equations.

The heat transfer rate at the surface of the metal follows the equations:

$$\frac{d}{dt} (H_0 + h) = K_1 (N_0 + n) - \alpha (H_0 + h)$$

$$K_1 = \alpha \frac{H_0}{N_0}$$

$$\frac{p}{\ell} h = \alpha \frac{H_0}{N_0} n - \alpha h, \quad (3)$$

where H_0 and h are equilibrium and deviations from heat transfer per unit time per unit area; K_1 is the proportionality constant between neutron density and rate of change of heat transfer rate; and $1/\alpha$ is the heat transfer time lag between fission and bubble formation for the geometry in question, and, in general, could be a function of p .

The equilibrium void fraction due to bubbles (V_0), and its variations (v), obey the equations,

$$\begin{aligned}\frac{d}{dt}(V_0 + v) &= K_2 (H_0 + h) - \gamma(V_0 + v) \\ K_2 &= \frac{\gamma V_0}{H_0} \\ \frac{p}{\ell} v &= \frac{\gamma V_0}{H_0} h - \gamma v \\ \frac{p}{\ell} k &= \frac{\gamma K_0}{H_0} h - \gamma k,\end{aligned}\tag{4}$$

where K_2 is the proportionality constant between heat transfer rate and rate of change of void fraction; $1/\gamma$ is the residence time of a bubble in the core; and K_0 is $V_0(dk/dv)$; K_0 is the equilibrium reactivity in voids if the void coefficient, dk/dv , is constant. It should be pointed out that a more accurate representation is possible by having several groups of bubbles with different residence times - a trivial extension of the theory.

If Eqs. (1) through (4) are to have a solution, p must be one of the eigenvalues of the system, i.e., roots of the equation:

$$\begin{vmatrix} -\beta - p & \lambda \ell & 0 & N_0(1 - \beta) \\ \beta & -\lambda \ell - p & 0 & \beta N_0 \\ \frac{\alpha \ell H_0}{N_0} & 0 & -\alpha \ell - p & 0 \\ 0 & 0 & \frac{\gamma \ell K_0}{H_0} & -\gamma \ell - p \end{vmatrix} = 0$$

or, more simply,

$$\begin{vmatrix} -\beta - p & \lambda \ell & 0 & (1 - \beta) \\ \beta & -\lambda \ell - p & 0 & \beta \\ \alpha \ell & 0 & -\alpha \ell - p & 0 \\ 0 & 0 & \gamma \ell K_0 & -\gamma \ell - p \end{vmatrix} = 0\tag{5}$$

This equation shows the stability of a reactor, within the framework of the equations assumed here, to be determined by $\lambda \ell$, $\gamma \ell$, $\alpha \ell$, and $K_0 = V_0 (dk/dv)$.

2. Evaluation of Constants

The terms dk/dv , λ , and ℓ are conventional reactor physics quantities. The terms V_0 , γ , and α are basically thermohydrodynamical quantities which can be determined directly from non-reactor experiments, or can be calculated from existing theories. Regarding the latter approach, techniques for calculating V_0 are well known.⁵

One possible method of evaluating γ is to note that the reactivity associated with a particular bubble, i , formed at a certain location will have the time dependence:

$$k_i = k_{i0} \exp(-\gamma t)$$

according to the definition of γ , but also has a time dependence readily computed from the knowledge of void coefficient as a function of position and the bubble velocity:

$$\text{velocity} = \frac{(\text{Slip Ratio}) (\text{Inlet Coolant Velocity})}{1 - \text{Void Fraction}} .$$

Averaging these time functions over all bubbles and fitting to $\exp(-\gamma t)$ gives γ . Application of this procedure to a core whose lower third does not boil, along with a number of simplifying approximations, gives

$$\gamma \approx \frac{\text{Mean Steam Bubbles Velocity}}{\frac{1}{5} (\text{Core Height})}$$

The calculation of α is quite straightforward, consisting merely of solving

$$c \frac{d}{dt} (\Theta_0 + \theta) - K \nabla^2 (\Theta_0 + \theta) = \text{heat production per unit volume per second}$$

with appropriate boundary conditions within the fuel element, and with the surface conditions:

$$-K(\nabla \Theta_0)_\perp = A(\Theta_0 - T_c)^\eta = H_0$$

$$-K(\nabla \theta)_\perp = A\eta(\Theta_0 - T_c)^{\eta-1} \theta = h .$$

Here Θ and θ are equilibrium and deviations from equilibrium fuel temperatures; c and K are the specific heat and thermal conductivity of the fuel; $A(\Theta_0 - T_c)^\eta$ is a phenomenological boiling heat transfer expression; and T_c is the coolant temperature. Equation (3) is the result, α being identified with the coefficient of $H_0 n / N_0$.

⁵P. A. Lottes, W. S. Flinn, "A Method of Analysis of Natural Circulation Boiling Systems," Paper delivered at a meeting of the American Nuclear Society, June, 1956.

For uniform heat production in an infinite slab of half-thickness δ ,

$$\begin{aligned} \frac{1}{\alpha} &= \tau \left\{ \frac{\Theta_0 - T_c}{\delta \eta \left(-\frac{d\Theta_0}{dx} \right)_{\text{edge}}} + \frac{\sqrt{\tau p/l} \cotanh \sqrt{\tau p/l} - 1}{\tau p/l} \right\} \\ &= \tau \left\{ \frac{\Theta_0 - T_c}{\delta \eta \left(-\frac{d\Theta_0}{dx} \right)_{\text{edge}}} + \frac{1}{3} + \text{terms of order } \tau p/l \right\}, \end{aligned} \quad (6)$$

where $\tau = c \delta^2 / K$. On the other hand, for delta-function heat production at the central plane of an infinite slab,

$$\begin{aligned} \frac{1}{\alpha} &= \tau \left\{ \frac{\Theta_0 - T_c}{\delta \eta \left(-\frac{d\Theta_0}{dx} \right)_{\text{edge}}} + \frac{\sinh \sqrt{\tau p/l}}{\sqrt{\tau p/l}} + \frac{\cosh \sqrt{\tau p/l} - 1}{\tau p/l} \right\} \\ &= \tau \left\{ \frac{\Theta_0 - T_c}{\delta \eta \left(-\frac{d\Theta_0}{dx} \right)_{\text{edge}}} + \frac{1}{2} + \text{terms of order } \tau p/l \right\}. \end{aligned} \quad (7)$$

3. Lower Order Theories and Results

When the heat transfer time constant is not important, such as in the homogeneous limit, the solution of Eqs. (1), (2), and (4), with $h = H_0 n / N_0$, yields

$$\begin{vmatrix} -\beta - p & \lambda l & (1 - \beta) \\ \beta & -\lambda l - p & \beta \\ \gamma l K_0 & 0 & -\gamma l - p \end{vmatrix} = 0. \quad (8)$$

If, in addition, delayed neutron decay is ignored, the system is second order,

$$\begin{vmatrix} -p - \beta & 1 - \beta \\ \gamma l K_0 & -\gamma l - p \end{vmatrix} = 0. \quad (9)$$

On the other hand, ignoring delayed neutron decay in a non-homogeneous system yields

$$\begin{vmatrix} -\beta - p & 0 & (1 - \beta) \\ \alpha l & -\alpha l - p & 0 \\ 0 & \gamma l K_0 & -\gamma l - p \end{vmatrix} = 0. \quad (10)$$

35

If a root of a critical determinant is expressed as

$$p = (a + i2 \pi f) \ell ,$$

then all functions will have a time dependence

$$\exp (a + i2 \pi f)t.$$

Thus oscillatory conditions exist if any roots (p) have imaginary components. For example: the roots of Eq. (9) are

$$p = -\frac{1}{2}(\beta + \gamma \ell) \pm \sqrt{\frac{1}{4}(\beta - \gamma \ell)^2 + (1 - \beta)\gamma \ell K_0},$$

which are oscillatory if

$$K_0 < -\frac{(\beta - \gamma \ell)^2}{4(1 - \beta)\gamma \ell}. \quad (11)$$

While an extensive numerical exploration of Eq. (5) is needed to give the full ramifications of the theory, a limited investigation of the simpler Eqs. (8), (9), and (10) has been made for reactors of the BORAX type, with the following results:

(1) Delayed neutron decay is not important in the homogeneous limit and Eq. (11) gives the oscillatory criterion. This means the tendency to oscillate is enhanced by large negative reactivities in bubbles, large lifetimes, and small bubble-residence times (when the lifetime to residence time ratio is less than β). Because the slip ratio increases faster than the natural circulation water velocity decreases, at lower reactor pressures, this means the lower pressures have larger γ 's and greater oscillation tendencies.

(2) Once the negative reactivity in bubbles has attained the oscillation threshold the frequency of oscillation increases as the power level rises.

(3) The effect of heterogeneity - taking into account the thickness of the BORAX plates - is had by comparing the solutions of Eqs. (9) and (10). The heat transport time lag leads to a greater tendency to oscillate.

(4) Oscillatory solutions are damped, without recourse to an insertion of flow friction in the theory.

4. Conclusions

A theory of bubbling reactors, which a preliminary numerical investigation shows to qualitatively verify all oscillatory trends found in preliminary analysis of BORAX and SPERT⁶ has been developed: an increasing tendency to oscillate with higher power densities, lower pressures, and longer lifetimes; and higher oscillation frequencies at higher power densities. In addition, both theory and experiment give frequencies in the 1 to 3 cps range for typical conditions.

The parameters which affect stability within the framework of this theory are lifetime, heat transfer time, bubble residence time, and reactivity in bubbles.

One of the most important criteria of bubbling reactor design is that of stability, for it influences both the maximum attainable power and operational safety. With the proper theory supported by appropriate parameter determination experiments, this factor need not be left to chance.

B. Two-Phase Pressure Drop - P. A. Lottes, W. S. Flinn

A basic flow model for boiling water passing upward through heated rectangular channels has been selected; this predicts two-phase friction pressure drop rather closely. Preliminary experimental data fall within 30% of the calculated values of two-phase friction multiplier. The development of the calculation method follows.

From a mass balance at any flow cross section in a boiling channel

$$\frac{V_z}{V_f} = \left(\frac{x}{1-x} \right) \left(\frac{1-\alpha}{\alpha} \right) \left(\frac{\rho_f}{\rho_g} \right), \quad (1)$$

where V is velocity; x is steam weight flow rate divided by total weight flow rate; α is steam volume fraction; ρ is density; and the subscripts g and f refer to steam phase and water phase, respectively.

Present experiments indicate that the ratio of steam to water velocities is practically independent of boiling length, steam voids, power density or steam quality (x).

⁶Special Power Excursion Reactor Tests. W. E. Nyer, et al, "Experimental Investigations of Reactor Transients," IDO-16285, April 20, 1956.

The inlet velocity can be related to the local liquid velocity at any flow cross section by the equation

$$\frac{V_f}{V_{in}} = \frac{1 - x}{1 - \alpha} \quad (2)$$

From Eqs. (1) and (2) it can be shown that

$$\frac{V_f}{V_{in}} = 1 + \left(\frac{\rho_f V_f}{\rho_g V_g} - 1 \right) x \quad (3)$$

Since in the laboratory tests heat is added uniformly over the channel length, the value of x is a linear function of length; therefore, the ratio of local liquid velocity to inlet liquid velocity is a linear function of length.

In the flow model most of the liquid phase is in contact with the channel walls and most of the steam is flowing down a central core. It is proposed that the friction of the mixture may then be found by calculating the frictional drag of water phase along the channel wall.

$$\Delta P_{tpf} = \frac{f\rho}{2gD} \int_0^L V_f^2 dL = \frac{f\rho V_{in}^2}{2gD} \int_0^L \left(\frac{1-x}{1-\alpha} \right)^2 dL \quad (4)$$

We may define the friction factor multiplier as follows:

$$\bar{R} = \frac{\Delta P_{tpf}}{\Delta P_0} \quad (5)$$

where $\bar{R}$ is now the ratio of friction drop over the entire boiling channel to the friction drop over the same channel at the same mass flow rate with no boiling. Under these conditions, Eq. (4) and (5) may be combined to give

$$\bar{R} = \frac{1}{L} \int_0^L \left(\frac{1-x}{1-\alpha} \right)^2 dL \quad (6)$$

Integrating Eq. (6),

$$\bar{R} = \frac{1}{3} \left[1 + \left(\frac{1-x}{1-\alpha} \right)_e + \left(\frac{1-x}{1-\alpha} \right)_e^2 \right] \quad (7)$$

where subscript e refers to exit conditions. Another form of $\bar{R}$ is

$$\bar{R} = \frac{1}{3} \left[1 + \frac{1}{(1-\alpha_e) \left(1 - \frac{\rho_g V_g}{\rho_f V_f} \right)} + \left\{ \frac{1}{(1-\alpha_e) \left(1 - \frac{\rho_g V_g}{\rho_f V_f} \right)} \right\}^2 \right] \quad (8)$$

For small values of x , the expression reduces to:

$$\bar{R} = \frac{1}{3} \left[1 + \frac{1}{1 - \alpha_e} + \left(\frac{1}{1 - \alpha_e} \right)^2 \right] \quad (9)$$

C. Effect of Oxygen Addition on Water Decomposition - R. E. Bailey

In the past it has been generally thought that an impurity in water such as dissolved oxygen would enhance the decomposition of the water in the presence of high-energy irradiation.

At the present time the question arose as to whether the decomposition would gradually increase with increase in oxygen contamination or whether it would slowly increase at first and then reach a point where the rate of decomposition would rise rather rapidly.

The purpose of this investigation was to observe the rate of decomposition as a function of oxygen addition in the CP-5 reactor water. The reactor contains a D_2O volume of 6,020 liters ($pD = 6.5$). The operating temperature is 45C with a power output of one megawatt.

Bottled oxygen was passed through a Dri-Rite trap to remove moisture, a cold trap (acetone and dry ice) for complete drying, a rotameter, and finally through a check valve and into the reactor feed-water line. The oxygen was added to the suction side of the feed pump to ensure thorough mixing.

The oxygen was believed to have the following purity:

99.5% O_2
0.3% A
0.2% N_2 .

While the oxygen was being added, the temperature of the gas from the vapor space in the reactor was recorded before and after it entered the catalytic recombiner. The carrier gas was helium at 26C and 14.772 psia. The gas flow through the recombiner was constant at 30 scfm.

The average ΔT across the recombiner, with the pile running and with the pile down, was observed over a period of 15 days prior to initiation of the experiment. The "pile down" ΔT was included since the catalytic bed is internally heated by an electric heater. Any increase in ΔT was considered to be an increase in decomposition, since the temperature rise is caused by the heat of formation or recombination of a quantity of D_2 and O_2 per unit of time. The average ΔT was 2C.

39

An analysis of the heavy water in the reactor, before addition of oxygen, indicated an oxygen concentration of 0.162 ppm. It should be noted that this determination, along with subsequent oxygen determinations, were slightly high owing to the method imposed for taking samples. The values of ΔT are in error by approximately $\pm 20\%$ due to fluctuations of $\pm 0.4^\circ\text{F}$ in the inlet gas temperature.

In the first run, 0.0152 cc O_2 /liter of D_2O was added in increments to the reactor system. A 0.5°C rise in ΔT was noted.

In the second run, a total of 1.722 cc O_2 /liter was added in increments not exceeding 0.250 cc O_2 /liter D_2O , and only after steady-state conditions had been achieved. The over-all increase in ΔT was approximately 1.5°C .

In the third run (Table III) the oxygen concentration was varied from 0.760 ppm to 2.21 ppm O_2 by the addition of a total of 1.50 cc O_2 /liter D_2O . A corresponding increase of 2°C in ΔT was observed, indicating a possible 100% increase in decomposition.

Based upon the observations made during the course of the study, it is concluded that:

(1) The presence of oxygen in the reactor water increases the rate of decomposition.

(2) Up to a concentration of 2.21 ppm O_2 there was no indication of a particular minimum concentration at which decomposition would suddenly increase many-fold.

(3) In a boiling-type reactor it would be most advisable to have completely deaerated feed water.

The conclusions are confirmed by the results of similar experiments performed by Dolin and Ershler⁷ who shut off the recombiner in a system such as CP-5 and observed the increase in decomposition as a function of time.

⁷P. I. Dolin, B. U. Ershler, (USSR), "Radiolysis of Water in the Presence of H_2 and O_2 due to Reactor Radiation, Fission Fragments, and X-Radiation." Proceedings of the International Conference on the Peaceful Uses of Atomic Energy held in Geneva, August 8 - August 20, 1955, Vol. 7, p. 564 (1956).

Table III

SUMMARY OF WATER DECOMPOSITION EXPERIMENT - RUN No. 3

 ΔT (Normal Reactor Operation); $\Delta T_U = 24^\circ\text{C}$ ΔT (Pile Down); $\Delta T_D = 22^\circ\text{C}$ $\Delta T_U - \Delta T_D = 2^\circ\text{C}$

Time, min	O ₂ add., cc/liter	ΔT , C	$\Delta T_U - \Delta T_D$, C	Remarks
0	-	21.0	0	Reactor down. O ₂ conc.: 0.760 ppm.
90	-	23.0	2	Reactor up - at steady state.
93-104	0.250	-	-	
105	-	23.5	2.5	
120	-	23.9	2.9	
124-35	0.250	23.9	2.9	
180	-	24.0	3.0	
195	-	-	-	Reactor down.
300-11	0.250	23.0	2.0	Reactor up - at steady state.
330	-	23.1	2.1	
360-71	0.250	24.1	3.1	
390-401	0.250	24.3	3.3	
405	-	24.3	3.3	
420-31	0.250	24.2	3.2	
435	-	24.3	3.3	
450	-	24.5	3.5	O ₂ conc.: 2.21 ppm.
465	-	24.6	3.6	
480	-	24.8	3.8	
495	-	25.0	4.0	
510	-	25.0	4.0	

Analysis of Carrier Gas at Start of Run:

He	99.7%
D ₂	0.03%
N ₂	0.1%
O ₂	0.16%

41

D. Fuel and Cladding Materials

1. Effect of Fused Salt Treatment on Corrosion Resistance of Zircaloy 2 - C. R. Breden, J. H. Monaweck

A molten 50-50 mixture of NaNO_3 - NaNO_2 was proposed as a bath for thermal cycling EBWR fuel assemblies. After preliminary tests had indicated that heating Zircaloy 2 in such a bath did not affect its subsequent corrosion resistance in high-temperature water, a more comprehensive study was made. The sample treatments and test results are summarized in Table IV.

Table IV

EFFECT OF FUSED SALT TREATMENT ON CORROSION OF ZIRCALOY 2 IN STATIC DEGASSED WATER AT 680F

Test Time: 4 periods totaling 1424 hr

Expt. No.	Sample Treatment	Over-all wt. change, mg/(cm ²) (mo)	Sample Appearance
44	Cold rolled	-0.038	Adherent grey coating with black edges.
45		-0.0098	
40	Cold rolled; heated in fused salt*	-0.062	Adherent grey coating with black edges.
41		-0.07	
42	Cold rolled; pickled	+0.15	Adherent semi-lustrous black coating.
43		+0.15	
38	Cold rolled; pickled; heated in fused salt*	+0.127	Adherent semi-lustrous black coating.

*24 hr in 50-50 NaNO_3 - NaNO_2 mixture at 525F.

It is apparent that exposure to the fused salt mixture has not affected markedly the corrosion behavior of the Zircaloy 2. The tests are being continued.

2. Corrosion of Tests on Zircaloy 2 Ingot Samples - C. R. Breden

Test specimens prepared from Zircaloy 2 ingots received by the Argonne Metallurgy Division from various sources were tested for 2349 hr in static degassed water at 550F. The results are summarized in Table V.

Table V

CORROSION OF ZIRCALOY 2 INGOT SAMPLES IN STATIC
DEGASSED WATER AT 550F.

Sample Description	Appearance
Bureau of Mines; Ingot 710D960; as cast.	Adherent lustrous black coating after 328 hr. Small white steak after a total of 657 hr. No further change after 2349 hr.
Allegheny Ludlum; Ingot 5y663; as forged.	Adherent lustrous black coating after 328 hr and 657 hr.
Allegheny Ludlum; Ingot 5y525; as forged.	Adherent lustrous black film; one small white spot after 328 hr. No further change after 2349 hr.
Perth Sterling; as forged.	Adherent lustrous black coating; small white area at one edge after 328 hr. No further change after 2349 hr.

Samples machined from Allegheny Ludlum Ingots 5y967, 5y971, 5y972, and 5y977 were pickled and tested in 750F steam at 1500 psi with the following results:

All four samples showed an adherent black coating after the first test period of 174 hr. There was no further change after the second period (336 hr). After the third period (267 hr), the sample from ingot 5y967 was slightly grey in color; the remaining specimens were covered with tiny white spots. Tests are continuing.

3. Aluminum Alloys

a. Dynamic Corrosion Tests - N. R. Grant

Loop No. 1, Run No. 5 (500F, pH 5-6):

Samples of aluminum-nickel alloy (M-388) are currently in long term test. The following observations were made upon completion of the first two months of testing.

43

Test time, hr	Water Velocity, fps	Wt. Loss, mg/cm ²
785	18	11.0
	7	7.6
1511	18	14.2
	7	11.0

The samples had a high initial weight loss but the rate after the two periods was only 4.5 mils/yr for both velocities. The appearance of the cleaned metal surface has been very good.

Loop No. 3, Run No. 4 (500F, pH 6.8 - 7.0):

Samples of aluminum-nickel alloy (M-388) which exhibited a low weight loss after two weeks of corrosion with natural hydrogen gas build-up were divided into two groups. The first was run for an additional four weeks in degassed water; the second group for an additional four weeks with hydrogen gas added to the water. The results are tabulated below:

Total Hours	Loop Condition	Weight Loss, mg/cm ²	
		18 fps	7 fps
317	Natural H ₂ build-up	4.55	3.01
654	Degassed	6.48	4.13
990	Degassed	9.15	5.44
656	200 cc H ₂ /liter added to loop	6.96	4.31
992	200 cc H ₂ /liter added to loop	10.30	5.77

The results are also plotted in Fig. 6. In both cases there was only a slight change in appearance after the first two weeks, but after four weeks the scale on the high-velocity samples started to powder and break (Fig. 7).

Although the samples have a high initial weight loss, there seems to be a good possibility that the long-term weight loss may be reasonably low. The smooth surface appearance is also encouraging. From the relatively short term test it appears that even after the M-338 alloy has been treated under favorable conditions, a sudden change in these conditions will cause an increase in the corrosion.

NOTE: PARENTHEZIZED NUMBERS INDICATE TEST SEQUENCE

FIG. 6
EFFECT OF CHANGING TEST ENVIRONMENT ON
CORROSION OF ALUMINUM-NICKEL ALLOY (M-388)
(LOOP NO. 3, RUN NO. 4; 500 F, pH 6.8-7.0)

b. Boiling Corrosion Tests

The second of a series of boiling corrosion tests on an aluminum-nickel clad, stainless steel tube was completed.⁸ The tube was exposed for 661 hr in water at 422F (300 psig). The average heat flux was 25,000 Btu/(hr)(sq ft).

The weight gain was 1.7 mg/cm². Upon removal of the scale, the weight loss was 1.26 mg/cm². The latter value, combined with the weight loss of the first tube, gives a corrosion rate of 2.8 mils/yr.

Figure 8 shows magnified views of pits which developed from inclusions noted in the clad at the start of the test. One pit occurred at the edge of the welded area in the steam phase of the test

⁸Reactor Engineering Division Quarterly Report - Section I, ANL-5571, (July, 1956) p. 96.

45

WATER VELOCITY: 7 fps

18 fps

DEGASSED WATER

WATER VELOCITY: 7 fps

WATER VELOCITY: 18 fps

200 ccH₂/liter ADDED TO LOOP WATER

FIG. 7
EFFECT OF CHANGE IN GAS CONDITION ON CORROSION OF ALUMINUM-
NICKEL ALLOY (M-388) EXPOSED FOR 990 hr IN 500 F WATER

FIG. 8
PITS ORIGINATED BY INCLUSIONS IN
ALUMINUM-NICKEL CLAD. (X50)

section [Fig. 8(a)]. The second pit [Fig. 8(b)] occurred at the steam-water junction, but not near the welded area. The tube was welded lengthwise with two parallel beads. Some cracks were observed at various points between the beads. Aside from these areas, the over-all surface appearance of the tube was good.

A third tube is currently in test at the same conditions.

c. Corrosion of Spot-welded 2S Aluminum - C. R. Breden

Samples of 2S aluminum, spot-welded to simulate BORAX-III fuel assembly welds, have accumulated 3343 hr exposure in degassed water at 420F. No accelerated corrosion attack in the weld areas has been observed.

d. Corrosion - Irradiation Tests - N. R. Grant

The corrosion-irradiation of aluminum-nickel alloy (M-388) in the Argonne Water Loop at MTR has been completed. Partial results were reported in ANL-5571.⁹

During operation of the loop, the temperature varied as a result of refueling and operational difficulties.

Temp, F	Time, hr		
	Period 1	Period 2	Period 3
>200	24	120	218
330	-	-	16
400	-	20	30
485-500	200	340	300

The results are summarized in Table VI and plotted in Fig. 9.

There was no noticeable increase in corrosion with increase in distance from the core as was observed in MTR-W-6 test.¹⁰ Only the samples in cartridge section 3 show this effect; however, the highest weight loss may be attributed, in part, to erosion (first sample at inlet of test section).

⁹Reactor Engineering Division Quarterly Report - Section I, (July, 1956) p. 100.

¹⁰Reactor Engineering Division Quarterly Report - Section I, ANL-5561, (April, 1956) p. 97.

Table VI

SUMMARY OF CORROSION-IRRADIATION TESTS ON ALUMINUM-NICKEL ALLOY (M-388)

(MTR W 7)
Water Velocity-15 fps

Cartridge Section	Hr at 485 - 500F	Water Conditions		Total Neutron Exposure		% of Exposure at 485 - 500F	Weight Loss, mg/cm ²	
		pH	Resistivity, megohm-cm	Fast, nvt x 10 ¹⁹	Thermal, nvt x 10 ²⁰		In-pile	Out-of-pile
1	200	6.2 - 7.8	0.5 to 3.0	2.0	2.0	80	1.82; 3.22 3.14; 3.98	6.86; 7.98 9.40; 11.40
4 (Replacement)	340	7.4 - 8.5	Very Low	3.0	3.0	75	8.56; 7.48 7.52; 7.78	11.39; 10.73 11.43; 11.25
2	540	-	-	5.0	5.0	77	4.64; 4.95 4.56	13.63; 13.36 13.15; 12.76
5 - 6 (Replacement)	300	7.3 - 8.5	0.25	4.8	4.8	50	6.96; 6.59 6.46; 6.82 7.30; 6.70 6.56; 5.78	14.1 ; 13.7 13.9 ; 14.0 13.7 ; 13.7 14.0 ; 14.1
3	840	-	-	9.8	9.8	64	7.86; 8.54 10.9	17.8

FIG. 9
 CORROSION-IRRADIATION OF ALUMINUM-NICKEL
 ALLOY (M-388) IN ARGONNE WATER LOCP AT MTR.
 WATER VELOCITY: 16 fps; TEMPERATURE: 485-500 F

49

Due to operational errors, the ion exchange column was not placed in operation during the second and third test periods. The effect of the resultant low-purity water was more pronounced on the in-pile replacement samples [Fig. 10 (c)]. The original in-pile samples immersed in high-purity water during the first period [Fig. 10 (a)] were not affected greatly during the second period. With continued exposure to low-purity water in the third period, however the samples may have lost some of the protective scale and corroded at a higher rate. If only the first two original in-pile samples are used for a determination, the corrosion rate is 6.4 mils/yr.

The in-pile samples showed some corrosion attack around the clamp area. The attack was more pronounced on the out-of-pile samples [Fig. 10 (d)]. Appearance-wise, the surface roughness of the original out-of-pile samples increases with time, while the surface of the in-pile samples remains about the same.

In general, the results were not as good as had been expected. A new test of longer duration is being planned.

4. Urania - Thoria Fuel Materials

a. Solution - Irradiation Tests - R. E. Bailey

Run No. 22 of a series of irradiation tests on urania-thoria fuel pins in the CP-5 Autoclave Facility has been completed. The fuel pin consisted of four ceramic pellets (approximately 0.250 in. dia. and 0.48 in. long), encapsulated in an aluminum-1% nickel can, with the annular space (~0.005 in.) filled with lead to facilitate heat transfer. The pellets were composed of a mixture of 90% ThO₂ and 10% UO₂ which was 93.22% enriched. The pellets were fired, in a gas-fired furnace, to a temperature of 3327F. The average density of the fired pellets was 9.8 gm/cm³; the porosity was 1.3%.¹¹

For this particular run a hole (1/16 in.) was drilled 0.50 in. from the top of the pin through the aluminum-nickel cladding and the lead bond. This was done in order to observe any deleterious effects of the water on the urania and thoria, and to release any buildup of internal pressure in the capsule.

The capsule was irradiated for 20 weeks in demineralized water at 450-500F at 500 psig, with an exposure of 114 MWD of power. This corresponds to a calculated burnup of 4.93% of the uranium atoms, and 0.164% of total core atoms.

¹¹ERRATUM: The density and porosity of specimens irradiated in Runs No. 18, 20, and 21, reported in Reactor Engineering Quarterly Reports: ANL-5511, 5561, and 5571, respectively, should be corrected accordingly.

FIG. 10
REPRESENTATIVE AREAS OF ALUMINUM - 1% NICKEL
SAMPLES EXPOSED IN ANL LOOP AT MTR
(MTR-W-7)

5/

After irradiation the capsule was slightly distorted and was covered with a brownish-black coating of aluminum oxides. The distortion was noted as a slight increase in diameter near the bottom of the capsule with a gradual tapering off near the top. A dimensional check revealed a slight increase (0.25 in.) in length.

Following removal of the oxide scale, the capsule was weighed and photographed. The weight loss was 0.37 gram, corresponding to a corrosion rate of 7.7 mils/yr (assuming a linear corrosion rate and using an average area), in a test run of 2737 hr duration at full CP-5 power. Magnified views of the capsule are shown in Fig. 11.

The heat generated by the pin at full CP-5 power was calculated to be 71,400 Btu/(hr)(sq ft). It is believed that boiling occurred on the surface of the pin.

The temperature gradient through the fuel pin was calculated in the following manner:

The self-shielding value (F)¹² of the fuel pin was calculated and found to be 1.019; hence it was assumed that the heat produced per unit volume was directly proportional to the mass of the material. Gamma-ray heating and end effects were neglected. The thermal conductivity of thoria was used. A scale factor (corrosion film on the pin surface) was included. The unit elongation (E_z) in the longitudinal direction was assumed to be zero.

The resultant high temperature gradient (Fig. 12) would indicate the presence of considerable thermal stresses.

The thermal stresses were calculated in the following manner:

The temperature drop (ΔT) was obtained from the equation of Carter:¹³

$$\frac{\partial^2 T}{\partial r^2} + \frac{1}{r} \frac{\partial T}{\partial r} - \frac{q}{k} = 0 \quad , \quad (1)$$

where

r = radius

q = heat per unit volume

k = thermal conductivity .

¹²S. Glasstone and M. C. Edlund, "The Elements of Nuclear Reactor Theory," (New York: D. Van Nostrand Co., Inc., 1952), pp. 268-272.

¹³J. C. Carter, "Temperature and Stress Distribution in Spheres, Rods, Tubes, and Plates in which the Heat Source is within the Boundaries of the Solids," ANL-4690 (September 7, 1951).

52

(X2)

(X6)

FIG. 11
MAGNIFIED VIEWS OF Al-Ni ENCAPSULED URANIA-THORIA
FUEL PIN IRRADIATED IN CP-5 AUTOCLAVE FACILITY
(RUN NO. 22)

FIG. 12
TEMPERATURE GRADIENT THROUGH URANIA-THORIA FUEL PIN.
(CP-5 IRRADIATION - RUN NO. 22)

The stress equations for long solid cylinders are:¹⁴

$$\sigma_r = \frac{\alpha E}{1 - \nu} \left(\frac{1}{b^2} \int_0^b T r \, dr - \frac{1}{r^2} \int_0^r T r \, dr \right) \quad (2)$$

$$\sigma_t = \frac{\alpha E}{1 - \nu} \left(-T + \frac{1}{b^2} \int_0^b T r \, dr + \frac{1}{r^2} \int_0^r T r \, dr \right) \quad (3)$$

$$\sigma_z = \frac{\alpha E}{1 - \nu} \left(-T + \frac{2\nu}{b^2} \int_0^b T r \, dr \right) + E \epsilon_z, \quad (4)$$

¹⁴S. Timoshenko, "Theory of Elasticity," (New York: McGraw-Hill Book Co., Inc., 1934), pp. 366-368.

where

$\sigma_r, \sigma_t, \sigma_z$ = radial, tangential and longitudinal stress, respectively.

α = coefficient of thermal expansion.

E = modulus of elasticity.

ν = Poisson's ratio.

b = radius of outer surface.

T = temperature.

Substituting the value of ΔT in Eqs. (2), (3), and (4):

$$\sigma_r = \frac{E \alpha q}{16 k(1 - \nu)} (r^2 - b^2) \quad (5)$$

$$\sigma_t = \frac{E \alpha q}{16 k(1 - \nu)} (3r^2 - b^2) \quad (6)$$

$$\sigma_z = \frac{E \alpha q}{16 k(1 - \nu)} \left[4r^2 - 2b^2 (2 - \nu) \right] + E\epsilon_z \quad (7)$$

The data used to calculate the stresses were:

$E = 19.6 \times 10^6$ psi (value at $\rho = 8.82$ and a porosity of 17.6%).

$\nu = 0.084$ (average between 0 and 0.167).

$\alpha = (3.44 \times 10^{-6})/F.^{15}$

$k = 0.358$ Btu/(hr) (m) (F) (value at theoretical density of 10.06 gm/cc).¹⁶

Due to differences in density, porosity, and heat treatment between the materials used to obtain the above data and the materials used in the fuel pins, there is some error in the stress gradients plotted in Fig. 13.

The greatest tensile stress is at the surface of the pin (6.53×10^3 psi). The greatest compressive stress is at the center (13.7×10^3 psi). The modulus of rupture for similar ceramic is 1.3×10^3 psi and the compressive strength is 210×10^3 psi. Since the modulus of rupture is between the compressive strength and the tensile strength of the material, it

¹⁵W. A. Lambertson, F. H. Gunzel, Jr., "Refractory Oxide Melting Points," AECD-3465 (Oct. 22, 1952).

¹⁶"Homogeneous Reactor Project Quarterly." Progress Report for Period Ending July 31, 1953, ORNL-1605 (Oct. 20, 1953).

FIG. 13
STRESS DISTRIBUTION IN URANIA-THORIA FUEL PIN
IRRADIATED IN CP-5 AUTOCLAVE FACILITY
(RUN NO. 22)

would seem logical that the thermal stresses under steady-state conditions would be sufficient to cause cracking. The thermal stresses can be reduced by increasing the thermal conductivity and by decreasing the diameter of the pellet since the stress is inversely proportional to the diameter squared.

One other possibility for the cause of the cracking would be the formation of fission gas which, due to the low porosity of the ceramic, would be trapped and hence would create additional stresses.

Once the ceramic is cracked the fission gases can escape. It is conceivable that this released gas is responsible for the deformation of the 1% Ni-Al jacket, since the deformation increases with distance from the drilled hole.

After the capsule was weighed, the can was removed with a 20% NaOH solution. It was discovered that the top pellet, the one directly above the hole, was completely encapsulated in lead (Fig. 14).

56

TOP PELLET BEFORE REMOVAL OF LEAD

CROSS SECTION OF LARGE SEGMENT OF TOP PELLET

PIECES FROM BROKEN TOP PELLET

PIECES FROM BROKEN PELLET

FIG. 14
URANIA-THORIA PELLETS IRRADIATED IN
CP-5 AUTOCLAVE FACILITY (RUN NO. 22)

57

Upon removal of the lead (by melting), the pellet was found intact. No dimensional changes had occurred; hence, the elongation of the capsule was not caused by the growth of the pellets. While enroute to a high-level activity cave, the top pellet broke apart, as did the others (Fig. 14).

Examination of the broken pellet revealed no evidence of ill effects from the water which might have entered the capsule. There were some reddish spots on the surface of the pellet which was completely encapsulated in lead, and these might have come from an iron oxide deposit.

In conclusion it can be stated that the ceramic material possibly cracked under thermal stresses and/or fission gas buildup.

b. Fabrication of BORAX-IV Fuel Elements - C. K. Soppet

The tube plate extrusion die has been received and approximately 500 ft of 2S aluminum extrusion (trial run) has been fabricated. The extrusion is satisfactory in all respects. Due to the extensive porosity encountered in casting the M-388 alloy, the fuel assembly and fittings were cast in 356 aluminum alloy at Rock Island Arsenal and delivered to Argonne for finish machining. Side plate stock has been rolled into strip which is currently being heat treated and straightened preparatory to machining into side plates. The 2 wt-% boron-stainless steel rod, to be used as removable poison, has been received.

A trial batch of 40 kg of thoria has been fabricated by the Metallurgy Division. The pressing was successful, indicating that the equipment can be used without further modification. Full-scale production is expected to begin about July 15.

The equipment for lead bonding the fuel pellets into the tube plate has been fabricated and assembled. Some difficulty was encountered in loading the pellets (0.240 in. OD) into the tubes (0.256 in. ID) of the tube plate. Consequently the pellet diameter has been reduced to 0.225 in. Some unbonded areas have also been observed. It is thought that these are caused by a pellet moving along one side of the tube during the insertion of the pellets into the lead-filled tube plate thereby reducing the annulus and restricting the flow of lead around the pellet. Methods of keeping the pellet centered in the tube during insertion are being investigated.

E. Control and Thermal Shield Materials

1. Corrosion of Boron-Stainless Steel - N. R. Grant

The static and dynamic testing of various boron-stainless steels has continued. The results to date are shown in Tables VII, VIII, and IX. Almost all of the alloys show favorable corrosion resistance. All of the tests are being continued.

Table VII
DYNAMIC CORROSION OF BORON-STAINLESS STEEL IN DEGASSED WATER AT 550F
Water Velocity: 22 fps

Source	Sample No.	Type of Weld	Nominal Composition, %		Analyzed % B	Time, hr	Rate of Weight Change, (mg)/(cm ²)/(mo)	Comments
			Cr	Ni				
KAPL	A-4403a b	-	18	14	NA*	5850	-0.0433, -0.01 -0.0391, -0.0485	Adherent black coating.
Electric Steel Foundry Co.	A-4463	Welded with 308 SS rod. Cross section of weld.	18	8	1.5, 1.49	4585	-0.0205, -0.0066	Adherent lustrous black tarnish. Loose black coating on edges. Weld same as sample.
Superior Steel Co.	A-4480	-	18	11	0.96, 1.01	3831	-0.0059, -0.0111	Thin loose black coating, heavier on edges, over adherent black tarnish.
Universal Cyclops	A-4483a	Welded with 308 SS rod. Cross section of weld.	18	15	0.68, 0.63	3831	-0.0162, -0.0149	Adherent brown-black tarnish. Purple tarnish on weld. Loose black coating on edges.
	A-4483d	Sample cut from welded slab out of heat zone, parallel to rolling direction.				3831	-0.0046, -0.0031	Adherent brown-black tarnish. Loose black coating on edges.
	A-4538c,d	-	18.1	14.8	1.16	2196	-0.0275, -0.0365	Adherent black tarnish. Loose black coating on edges.
Crucible Steel Co. (Heated to 2050F, forged at 1850F, and air cooled.)	A-4494b	-	18.8	8.7	1.85	3831	-0.0155, -0.0074	Adherent black tarnish. Loose black coating on edges.
(Same as above, then heat treated at 2000F for 1 hr, furnace cooled.)	A-4494i,j	-	18.8	8.7	1.96	2196	-0.0262, -0.0157	Adherent blue-black tarnish. Loose black coating on edges.
	A-4494x,y	Welded with Arcos Chromend K rod, lime coated. Cross section of weld.	18.8	8.7	1.85	2919	-0.0293, -0.0245	Adherent blue-black tarnish. Brown-black tarnish on weld. Loose black coating on edges.

*Not analyzed.

Table VIII
STATIC CORROSION OF BORON-STAINLESS STEEL

Source	Sample No.	Nominal Composition, %		Analyzed % B	Temp., F	Condition	Time, hr	Rate of Weight Change, (mg)/(cm ²)/(mo)	Comments
		Cr	Ni						
Electric Steel Foundry Co.	A-495	12	8	1	600	Degassed Air satd. Degassed	3538	+0.0103 +0.0161 +0.0231	Adherent black coating. Adherent brown-black tarnish. Pickled before test in 40% HNO ₃ -4% HF. Adherent black coating.
	A-496	12	8	2	600	Degassed Air satd. Degassed	3538	+0.0102 -0.0387 +0.0052	Adherent black coating. Adherent grey-black coating. Pickled before test in 40% HNO ₃ -4% HF. Adherent black coating.
	A-497	12	8	3	600	Degassed Air satd. Degassed	3538	-0.0026 +0.0232 -0.2040	Adherent black coating. Adherent black coating. Pickled before test in 40% HNO ₃ -4% HF. Adherent blue-black coating.
	A-4473b-1* b-2* a-1** a-2**	18	14	2.8 2.8 2.2 2.2	550	Degassed	3161	+0.0285 +0.0182 -0.0340 -0.0010	Adherent dull black coating.
	A-4561	19.7	12.2	4	550	Degassed Air satd.	633	-0.2960 -0.2310	Slightly loose, uneven, brown-black coating. Adherent black coating.
	A-4480	18	11	1	550	Degassed Air satd.	2517	-0.0179 -0.0273	Adherent brown-black coating. Adherent black coating.
	A-4494a	18.8	8.7	2	550	Degassed Air satd.	1908	+0.0474 +0.0808	Adherent black coating. Area where samples are in contact is noticeable. Adherent brown and black coating. Loose black coating where samples are in contact. Dark tarnish spot at one end.
	A-4494g	18.8	8.7	2	550	Degassed Air satd.	1032	+0.0928 +0.0792	Adherent black coating. Adherent blue-black to brown-black tarnish.
	A-4538	18.1	14.8	1	550	Degassed Air satd.	663	+0.0352 +0.0177	Adherent brown, red-brown, purple, and black tarnish. Adherent brown to black tarnish.
	A-4528a**** A-4528b A-4528c 1/8 in. radius 3/16 in. radius 1/4 in. radius 3/8 in. radius	18	8	2	550 550 550 550	Degassed Degassed Degassed Degassed	996 996 996 996	+0.0061 -0.0082 -0.0056 -0.0490 -0.0138 -0.0218	Sample cracked before test. Adherent, uneven, green and black coating. Sample cracked before test. Adherent, uneven, dark brown and black coating. Adherent, uneven grey-black coating. Adherent, uneven dark reddish-brown coating. Adherent, uneven grey & grey brown coating. Adherent, uneven grey-black coating.
	A-4528d 1/8 in. radius 3/16 in. radius 1/4 in. radius 3/8 in. radius				550	Degassed	996	-0.0236 -0.0025 -0.0044 -0.0056	Sample broken along bend before test. Adherent, uneven, grey and black coating. Adherent, uneven, grey-black coating. Adherent, uneven, grey-black coating. Adherent, uneven grey coating.

*As cast

**Heat treated at 2256F for 3 hr.

***NA = Not analyzed.

****All A-4528 samples pickled in 40% HNO₃-4% HF before test.

Table IX
STATIC CORROSION OF WELDED BORON-STAINLESS STEEL

Source	Sample No.	Origin of Sample	Type of Weld	Nominal Composition, %			Test Conditions		Rate of Weight Change mg/cm ² /mo	Comments
				Cr	Ni	B	Temp., F	Water Condition		
Universal Cyclops (vacuum melted)	A-4483b	Cross section of weld, then cut in half through weld.					600	Degassed	-0.0219, -0.0140	Adherent dark brown, red, and blue tarnish. Grey-black tarnish on weld. Blue-black coating on contact area.
							550	Degassed	-0.0179, +0.0080	Adherent brown-black tarnish. Grey-black tarnish on weld. Red-black coating on contact area.
							550	Air satd.	-0.0219, -0.0260	Adherent dark brown tarnish. Grey-black tarnish on weld. Black coating on contact area.
	A-4483c	Cut from welded slab out of heat zone.	308 SS rod	18	15	1.2	0.68, 0.63	Degassed	+0.0018, -0.0036	Adherent reddish-brown tarnish, blue-black on contact area.
							550	Degassed	+0.0018, -0.0199	Adherent brown-black tarnish, black on contact area.
	A-4483d	Cut from welded slab out of heat zone, parallel to rolling direction.					550	Air satd.	-0.0307, -0.0181	Adherent blue tarnish, brown around edges. Black tarnish on contact area.
							550	Degassed	-0.0039, -0.0065	Thin loose red-brown coating over adherent brown-to-black tarnish.
Superior Steel Co.	A-4507-1,2		Heliarc, bare 304 SS-1% B rod, single bead, 1% B to 304 SS.					Degassed Air satd.	+0.0078 -0.0118	Adherent black coating. Adherent brown, blue, and purple tarnish, and loose brown-black coating on holder contact area.
	A-4507-3,4		Heliarc, bare 304 SS-1% B rod, double bead, 1% B to 1% B.					Degassed Air satd.	-0.0003 +0.0118	Adherent black coating. Adherent brown and black tarnish. Loose black coating on holder contact area.
	A-4507-5,6		Heliarc, bare 304 SS-1% B rod, single bead, 304 SS to 1% B.					Degassed Air satd.	-0.0078 +0.0098	Slightly loose black coating. Two small brown spots on weld. Adherent dark red to blue-black tarnish. Loose black coating on holder contact area.
	A-4508-7,8	Cross section of weld.	308 SS coated rod, single bead, 1% B to 304 SS.	18	8	1	550	Degassed Air satd.	+0.0178 +0.0078	Adherent black coating. Adherent grey-brown and blue-tarnish, brown on weld. Loose brown coating on holder contact area.
	A-4508-9,10		308 SS coated rod, double bead, 1% B to 1% B.					Degassed Air satd.	+0.0569 -0.0255	Adherent dull black coating. Adherent reddish-brown to black tarnish. Loose black coating on holder contact area.
Crucible Steel Co.	A-4508-11,12		308 SS coated rod, single bead, 304 SS to 1% B.					Degassed Air satd.	+0.0235 0.0000	Adherent black coating. Adherent dark brown-to-black tarnish. Black tarnish on weld.
	A-4494c, d, e, f		Arcos Chromend K rod, lime coated.	18.8	8.7	2	1.85	Degassed	+0.0210, +0.0265	Adherent black coating, dark brown on weld. A few small brown spots appearing on metal.
								Air satd.	-0.1274, -0.1462	Adherent brown, blue, and green-black tarnish. Loose black coating on contact holder area. Sudden large loss in weight.
	A-4534a	Cross section of weld.	Inco A rod, 1.5% B welded to 304 SS.	18.8	8.7	1.5	550	Degassed	-0.0045	Adherent green-black tarnish on boron-stainless steel. Adherent blue, brown, and purple tarnish on 304 SS. Adherent blue and purple tarnish on weld.
	A-4534b		Arcos Chromend K rod, lime coated, 1.5% B welded to 304 SS.					Air satd.	+0.0543	Adherent dark reddish-brown-to-black tarnish.
								Degassed	+0.0495	Adherent green-black tarnish on boron-stainless steel. Adherent dark reddish-brown on rest of sample.
								Air satd.	+0.0495	Adherent blue, straw, and brown tarnish. Also spots of blue-black tarnish.

61

2. Corrosion of Miscellaneous Control Materials - C.R. Breden

a. Rolled Hafnium

Samples of Allegheny Ludlum rolled hafnium (typical of EBWR control rod material) are still satisfactory after 2004 hr and 1294 hr in static, degassed water at 600F and 680F, respectively. Similar specimens are showing low weight changes and satisfactory tarnish films after 2919 hr in dynamic water at 525F.

b. Boron Carbide Solution Tests

Two samples of pressed boron carbide (Norton) have been continued in test in 420F and in 500F degassed, demineralized water.¹⁷ After 328 hr both samples exhibit an adherent grey coating, slightly roughened surfaces and no measurable weight change.

The boron content of the water is 0.14 $\mu\text{gm/liter}$ at 420F, and 0.10 $\mu\text{gm/liter}$ at 500F.

c. Aluminum-Nickel Jacketed Cadmium

Corrosion tests were made on a simulated control rod consisting of a cadmium sheet enclosed in a picture frame cover plate box of 2S aluminum - 1% nickel alloy. The cover plates were fastened together by spot welds around the frame area. No attempt was made to make the box watertight. The fabrication concept was based on earlier tests which indicated that the corrosion rate of cadmium in high-temperature water, while appreciable, was not excessive or catastrophic in nature.

During two successive two-week tests in 420F air-saturated water the over-all weight change rate was +4.21 and +1.3 $\text{mg}/(\text{cm}^2)(\text{mo})$, respectively. The cadmium content in the water was 0.005 $\mu\text{gm/ml}$ and 0.1 $\mu\text{gm/ml}$, respectively. Some corrosion attack is visible in the spot weld areas which appear to have a black scale. In addition, there is a small lump of extruded corrosion product over the crevice between one cover plate and the picture frame, adjacent to a small mound which appears to be a blister in the edge of the picture frame.

F. Structural Material Corrosion Tests - N. R. Grant

Samples of various structural materials have been continued in static and dynamic corrosion tests. All materials, except Thermonol in dynamic test, appear to be fairly corrosion resistant (Tables X and XI).

¹⁷Reactor Engineering Division Quarterly Report - Section I, ANL-5571, (July, 1956) p. 121.

Table X
STATIC CORROSION OF MISCELLANEOUS STRUCTURAL MATERIALS

Material	Source	Sample No.	Test Conditions			Rate of Weight Change, mg/(cm ²)(mo)	Comments
			Temp., F	Water Condition	Time, hr		
304 Stainless Steel (Cladding for EBWR pressure vessel)	Babcock & Wilcox	A-4462	550	Degassed	2541	+0.0135, +0.0030, +0.0539, +0.0389	Adherent dark brown coating. Area where the samples are in contact is noticeable. Loose brown-black coating where samples are in contact.
Stainless Steel	Universal Cyclops (SS A-202)	A-4526	500	Degassed	1062	+0.0085, +0.0085, +0.0128, -0.0169	Adherent straw and blue tarnish. Adherent blue to red-brown tarnish. Adherent red, brown and blue-green tarnish. Adherent blue and straw tarnish with streaks of brown.
Stainless Steel	Allegheny Ludlum	A-4569	500	Degassed	656	+0.0934, +0.0385, +0.0857	Slightly loose black coating. Adherent brown, purple, and blue tarnish. Thin slightly loose black coating over adherent brown-black tarnish.
Aluminum (welded)	Argonne	A-4395	420	Air satd.	2682	+0.0399, +2.280, +1.570, 0.641	Adherent brown tarnish. Adherent medium grey coating. Surface slightly blistered. Adherent medium grey coating.
Thermonol	Glenn L. Martin	A-4540	500	Degassed	1032	-0.0161, -0.0284	Slightly loose black coating over adherent black tarnish. Thin slightly loose black coating over adherent black tarnish.
Thermonol	Universal Cyclops	A-4525	500	Degassed	658	+0.0141, 0.0000, -0.0564, +0.0282	Adherent black coating. Adherent black tarnish with areas of brown. Thin loose black coating over adherent black tarnish. Thin loose black coating over adherent green-brown tarnish.
			680	Degassed	330	+0.1689, +0.0702	Thin loose black coating over adherent black tarnish. Adherent spotted red-brown coating.

Table XI
DYNAMIC CORROSION OF MISCELLANEOUS STRUCTURAL MATERIALS
Water Velocity: 22 fps

Material	Source	Sample No.	Test Conditions			Rate of Weight Change, mg/(cm ²)(mo)	Comments
			Temp., F	Water Condition	Time, hr		
Stainless Steel	Universal Cyclops (SS A-202)	A-4526e	550	Degassed	2196	-0.0147, -0.0158	Adherent lustrous dark blue and brown tarnish. Loose black coating on edges.
Thermonol	Glenn L. Martin	A-4540c	500	200 cc/1 H ₂	656	-7.58	Loose black coating.
Thermonol	Universal Cyclops	A-4525	550	Degassed	725	-0.908	Loose black coating.

63

G. Control Rod Drive Mechanisms

1. Hydraulic - W. J. Kann

Testing of the hydraulic control rod drive system has been discontinued.

The concluding tests merely reproduced earlier results. Visual inspection of the major components revealed no signs of wear or damage. The tapered bore cylinder had only a few burnish marks, but there was no scoring or galling of the bore.

2. Magnetic Jack - J. N. Young

The prototype model magnetic jack unit (No. 20)¹⁸ for the EBWR is currently in lifetime test in a facility designed to simulate the anticipated EBWR operating conditions: stagnant water at 600 psi, with 488F saturated steam at the top, and 100F water at the jack.

H. Argonne Water Loop at MTR

1. Operation - E. L. Martinec, G. A. Freund

Cartridge MTR-W-8 (an EBWR prototype fuel plate) has been under irradiation since the start of MTR cycle 66 and had accumulated a total exposure time of 3905 MWD at the end of cycle 71 on July 5, 1956. Essentially all of this exposure time was above 400F with approximately 75% at 485F. The maximum burnup is estimated to be 0.35% of the uranium atoms.

The EBWR prototype test plate contains uranium - 5% zirconium - 1.5% niobium with 1.44% U^{235} and is clad with Zircaloy 2. An identical natural uranium alloy plate is being exposed in the out-of-pile zone of the loop.

The loop operated satisfactorily from cycle 67 to cycle 70 with a few exceptions:

(1) The pressure drop across two of the main flow filters exceeded 5 psi shortly after the start-up of cycle 67. The filters were new and had been installed a few days prior to start-up. The high pressure drop caused the loop water flow to drop steadily until the low flow necessitated lowering the temperature from 485F to 450F. The filters were removed at the end of the cycle and were not replaced.

¹⁸Reactor Engineering Division Quarterly Report - Section I, ANL-5571, (July, 1956) Fig. 45, p. 125.

(2) After the start-up of cycle 70, the third filter indicated a high pressure drop and caused a low flow condition to exist. This filter is the discharge filter for "A" pump and was bypassed by switching to "B" pump. The filter will not be replaced until there is a shutdown for maintenance other than filter replacement.

(3) "A" pump developed trouble during cycle 67 and automatic switching to "B" pump took place. A test of "A" pump indicated the power to the pump was in excess of twice the normal power consumption of 12.5 kw, and the loop flow had almost ceased. The pump was removed and replaced with a reconditioned pump. The reconditioned pump has been functioning satisfactorily.

(4) During cycle 70 the top assembly of the relief valve became loose and relieved the system at a pressure lower than the normal relief setting. Although there was a considerable loss of water, it appears that the cartridge did not become dry. The relief valve leaked excessively on a pressure test and was plugged to permit continued operation. The system is protected by another relief valve on the top of the surge tank.

2. Examination of Second Irradiated Tube Tip - R. E. Bailey

The second in-pile tube tip from the Argonne Water Loop at MTR was examined to evaluate corrosion and irradiation effects upon the stainless steel Type 316 tube tip and the weld joining the tube tip to the Type 347 main tube.

The tube tip was in MTR for a total of 18 cycles and was exposed to 8902 MWD of power. At the tip of the tube this corresponds to a maximum fast neutron flux of 4.8×10^{20} and a maximum slow neutron flux of 4.8×10^{21} . For about 25% of the time while in service the tip was exposed to high temperature and pressure, the maximum conditions being 500F and 1500 psig.

Approximately 8.5 in. of the tube tip were removed at MTR and shipped to Argonne. The tube was sectioned longitudinally to facilitate examination of the inner surface. A fine, slightly adherent coating of Fe_3O_4 was observed. Removal of the coating (by brushing) disclosed a more adherent film. There was no corrosion attack at the weld (see Fig. 15).

Two horseshoes were cut from the tube with adjacent faces lying on the center line of the tip. One horseshoe was given to the Metallurgy Division for hardness measurements and metallographic examination along the length of the cut section. The other horseshoe was used for bend tests to compare the ductility of the irradiated tip, the weld metal and the weld-stainless steel interface with a similar unirradiated tip section.

FIG. 15
TYPE 316 IN-REACTOR TUBE TIP NO. 2
IRRADIATED IN ARGONNE WATER LOOP AT MTR

65

To compare the irradiation effects on the weld metal, the weld was cut from one of the two remaining tube halves. This semi-circular section was then bent in until it was completely closed. A similar test was performed on the unirradiated control.

In each bend test the metal was bent in 180 degrees except at the tube tip. Here the metal was bent in until closed. The welds on the unirradiated specimen were made more distinguishable by means of a slight etch, hence the darkness of the weld metal in Fig. 16 (c).

From all indications the irradiated material was still as ductile as the unirradiated control. This can be readily observed by noting the absence of cracks in the bent irradiated test sections shown in Fig. 16.

The metallographic examination of the irradiated metal showed no peculiarities when compared with the unirradiated control.

Hardness values obtained were as follows:

<u>Irradiated Specimen</u>	<u>Brinell</u>
Tube tip	255
Between tip and weld	270
On weld metal	255
On base metal adjacent to weld metal	265
Below weld	249
<u>Unirradiated Control</u>	
Average hardness	216

These values are an average of ten measurements made with a Knoop micro-hardness tester. These data show that the weld metal is as hard as the base metal, that the hardness tends to decrease as the neutron flux decreases along the length of the tube, and that the maximum increase in Brinell hardness due to irradiation was 54 BHN. It should be noted that the first irradiated tube tip from MTR showed an increase of 28 BHN (the final hardness was 234 BHN) after being exposed to a fast neutron flux of 3×10^{20} .

L. W. Fromm¹⁹ indicates that the Type 300 series stainless steels reach a saturation point where the effect of atomic dislocations becomes constant and, hence, the hardness reaches a constant value. In his plot of BHN versus integrated fast neutron flux a saturation hardness of 230 BHN is indicated. Data from both irradiated tube tips show values higher than this. They are, however, not too far out of line for two reasons.

¹⁹L. W. Fromm, "Design Evaluation of In-Reactor Tube for Argonne Water Loop at MTR (ANL-2)," ANL-5403 (March 1955).

UNIRRADIATED CONTROL

UNIRRADIATED CONTROLS

UNIRRADIATED CONTROL

(a) BENDS AT TUBE TIP

(b) BENDS AT WELDED SECTION OF TUBE TIP

(c) BENDS IN WELD RING

FIG. 16
BEND TESTS OF SPECIMENS FROM TYPE 316 TUBE TIP
IRRADIATED IN ARGONNE WATER LOOP AT MTR

One is that environmental conditions at the time of irradiation have an effect in the final hardness, and the second is that a hardness determination made on a rough as-cut surface is lower than that obtained on a polished surface. The hardness values on the first tube tip were made on the as-cut surface. The values obtained on the second tip were made on a polished surface after the metal test section had been ground down to eliminate any effects due to the slight cold working of the stainless by the cutting operation.

From the above it can be concluded that up to 8902 MWD of operation of the Argonne Water Loop at MTR there seemed to be no deleterious effects caused by irradiation or corrosion.

PART B. FAST POWER BREEDER REACTOR PROGRAM

SUMMARY

L. J. Koch

The preliminary design requirements for the EBR-II Plant (Architect-Engineer Prospectus) have been completed. An Architect-Engineer Contract Board has been established, and invitations have been extended to qualified firms to submit proposals to carry out the A-E work for the EBR-II Plant.

The preliminary plant design described in the A-E prospectus consists of three major plants plus supporting facilities. The reactor and primary coolant system, plus the fuel disassembly cell and associated equipment, are contained in a gastight steel structure (the Reactor Plant). Connected to this structure by appropriate air locks are: The Power Plant containing the control room for the entire power cycle, and the turbine generator and electrical systems; and the Process Plant containing the fuel process cell and associated laboratories. Two additional buildings are employed, one housing the secondary sodium pumping, purification, and storage equipment, and the other the steam generator. These units are interconnected with the Reactor Plant and Power Plant by appropriate piping. The plant arrangement is shown in Fig. 17. Figure 18 is an artists' conception of the over-all plant.

FIG. 17
EBR-II PLANT ARRANGEMENT

FIG. 18
EBR-II FACILITY

71

A change has been made in the basic reactor core geometry which involves eliminating the central blanket. This has been done to simplify analysis, particularly the consequences associated with a reactor melt-down. The central blanket creates a "hole" in the center of the core which might give rise to an "implosion-type" accident. Although this is a very improbable event, it is not amenable to analysis. Preliminary analyses have been completed on the modified core geometry, including critical mass estimates and the effect of control rods at various locations.

Performance tests have been conducted on a full-scale prototype of the EBR-II control rod drive mechanism. Latch release times were determined to be between 0.003 and 0.007 sec. Displacement versus time curves were also determined for the mechanism, with loads approximating the weight of the entire control rod system.

The irradiation of two EBR-II fuel elements containing "fissium" alloy pins was completed. These elements were irradiated to an estimated burnup of approximately 1% and were in excellent condition. The dimensional change of the pins was less than the clearance provided by the 0.006-inch static sodium bond in the fuel element. Some surface roughening occurred on the fuel pins, but there was no indication of warpage or appreciable growth.

The EBR-II Working Model continued to operate satisfactorily. The d-c electromagnetic pump has operated approximately 2,000 hr with no difficulty.

The Liquid-Metal Mock-Up Building (D-308) was completed and occupied. The construction and installation of three large (12 in. diameter) pipe loops were initiated. The large rotating plug seal test is also being installed. The two 5,000-gpm sodium pumps are scheduled for delivery the latter part of the summer, and the 10,000-gpm, d-c electromagnetic pump is being assembled. The test program on these pumps will be initiated upon the completion of the installation work.

EXPERIMENTAL BREEDER REACTOR-II (EBR-II)

A. Physics Calculations on the EBR-II Design Study No. 2 - W. B. Loewenstein

The nuclear properties of the EBR-II Design Study No. 2 have been evaluated on the basis of one-dimensional diffusion theory calculations. This analysis differs principally from that pertaining to Design Study No. 1 of EBR-II²⁰ in that the central blanket containing 55% depleted uranium has been replaced by fuel assemblies, thus enlarging the core by seven sub-assemblies. Most of the calculations were based on a core height of 14 in.; some information has been obtained concerning a core with a height of 17 in.

The current analysis is based on an upper and lower blanket containing 38% depleted uranium by volume. The design has since been modified so that this region contains 32% uranium by volume. Previous comparisons of a blanket containing 38% depleted uranium with one containing 55% depleted uranium indicated that on criticality the 6% difference is not serious.

To determine the critical mass of the system, the self-consistent analysis described in the calculations on Design Study No. 1²⁰ was applied. In essence, the method is such that the criticality calculation is considered complete when a radially bare slab and an axially bare cylindrical analysis yield the same critical core enrichment. The analysis further requires that at criticality

$$K^2 = b_r^2 + b_z^2 \quad ,$$

where

K^2 is the material buckling for the critical core enrichment,

b_r^2 is the radial buckling for the slab calculation,

and

b_z^2 is the axial buckling for the cylindrical calculation.

This analysis yields a critical mass of 192 kg of U^{235} for the core with a 14-in. height. The corresponding enrichment indicates that the U^{235}/U^{238} ratio = 1.1. The analysis further suggests an axial reflector saving of 10.4 cm averaged over both ends.

²⁰ Reactor Engineering Division Quarterly Report, ANL-5461
(April 15, 1955), p. 119

The analysis is based on a set of ten-energy group cross sections. This set has been used in the analysis of ZPR-III assemblies. In general, it is found that predicted critical masses, using one-dimensional methods, are higher than those experimentally observed. For example, a spherical calculation of a cylindrical assembly similar to EBR-II overpredicted the critical mass by 3.4%. Current work also indicates that the self-consistent criticality analyses for cylindrical assemblies predict critical masses which are too high by at least 5%, but probably not more than 10%. This sort of uncertainty must be noted when the calculated critical masses of this study are considered.

The above calculations are based on the principle that eight of the twelve control rods are in the core. The control rods pertinent to these calculations are those containing about 26% fuel alloy by volume.

If the control rods are located in the fifth row of subassemblies, as indicated by Design Study No. 1, the worth per rod is calculated to be $\delta k/\text{rod} \sim 0.06$. Placing rods in the fourth row of subassemblies gives $\delta k/\text{rod} \sim 0.009$. With rods in the third subassembly row, $\delta k/\text{rod} \sim 0.012$. Hence, it is seen that $\delta k \sim 0.003$ is gained per rod per row as it is moved inward from the edge of the core. Actual rod worths must be considered less than calculated worths. Simulated rod removal experiments on a ZPR-III assembly similar to EBR-II indicate that such effects are usually calculated to be greater than observed. Disagreement between theory and experiment is attributed, in part, to the inability to perform an exact calculation of such effects. The calculated effects may be as much as 20% greater than those observed, though, in general, the difference between theory and experiment is not as great.

If one accidentally replaces the fuel alloy in the central core subassembly with fully enriched material, calculation indicates that such a change gives $\delta k \sim +0.014$.

Replacing a fuel subassembly with an inner blanket subassembly in the fifth row gives $\delta k/\text{assembly} \sim -0.0041$.

Replacing a blanket subassembly with a fuel subassembly in the sixth row (first row of the blanket) gives $\delta k \sim +0.0027$.

If one considers a control rod at the center of the reactor where the fuel is replaced by a section containing 36% by volume of about 70% enriched B^{10} in boron carbide, the associated reactivity effect is $\delta k/\text{rod} \sim 0.034$.

Similar boron control with the normal control rods in the fifth row gives $\delta k/\text{rod} \sim 0.010$.

Calculated sodium density coefficients for the reactor give $\delta k/k \sim 0.06 \delta \rho/\rho$. This number probably will be slightly higher when one takes full account of the sodium in the structural gaps above and below the core. The sodium density coefficient for the core alone appears to be such that $\delta k/k \sim 0.04 \delta \rho/\rho$.

If one considers a volumetric expansion of the whole system, without changing the sodium density but conserving all other constituents, it is found that $\delta k/k \sim -0.4 \delta V/V$.

Assuming that the core radius contracts slightly, again conserving fuel and structural materials without changing the sodium density, it appears that $\delta k/k \sim -0.3 \delta r/r$. In this calculation the core contracts toward the center, thus introducing a sodium gap between the core and inner blanket.

A number of reasonable assumptions concerning the manner of a slow expansion of the system gives the following isothermal temperature coefficient:

$$\frac{\delta k}{\delta T} \sim -3.6 \times 10^{-5}/^{\circ}\text{C}.$$

This temperature coefficient is to be considered typical of average coolant temperatures of about 400C and does not include any possible contribution due to the Doppler effects.

To gain further insight, a number of possible core sizes were investigated along with the one described.

No.	Core Height, in.	Number of Core Assemblies (including control rods)	Critical Mass, kg U^{235}	$\frac{\text{U}^{235}}{\text{U}^{238}}$	Axial Reflector Saving, cm
1	17.5	91	262	0.62	10.4
2	14.2	91	239	0.75	11.0
3	17.5	61	206	0.86	10.0
4	14.2	61	192	1.1	10.4

A control rod calculation for cores such as No. 1 and No. 2 has also been completed. Here the rods are not at the edge of the core, but in the same location called for in the design. Such a calculation gives $\delta k/\text{rod} \sim 0.0058$. This is less than the worth of a control rod in the reference core and will be further reduced if one considers a rod with a 14-inch stroke in a core with a height greater than 14 in.

75

A calculation of the neutron lifetime for the reference assembly gives $\ell \sim 0.8 \times 10^{-7}$ sec.

A two-dimensional analysis of Design Study No. 1²¹ yields results different from those determined on the basis of the self-consistent one-dimensional analysis. A similar two-dimensional analysis of Design Study No. 2, to be performed in the near future, will give further insight concerning the parameters reported here.

B. Irradiation of EBR-II-Type Fuel Elements - J. H. Monaweck,
E. S. Sowa

Two fuel elements, tests Nos. CP-5-7²² and CP-5-8,²³ were removed from the CP-5 reactor for examination. Both elements contained "fissium" alloy. CP-5-7 contained 4% of the alloying constituents, while CP-5-8 contained 3%. The total irradiation time for each element corresponds to an estimated 1% burnup of total atoms. The thermal cycles accumulated were: 352 for CP-5-7 and 271 for CP-5-8.

The fuel elements were removed with great ease from their finned tube capsules. Figure 19 shows magnified views of both fuel elements after removal of surface NaK. The photographs are representative of the entire fuel element with the exception of the top end of CP-5-7 which was dented by a physical mishap during handling. The mishap resulted in loss of the gas sample within the tube; however, it was possible to measure the gas pressure in CP-5-8. This pressure was found to be close to atmospheric, which leads to the conclusion that only insignificant amounts of fission product gases escaped from the uranium into the gas space.

The extreme brittleness of the fuel alloy after irradiation prevented easy decanning of the 8-mil clad; consequently, the fuel was broken into many small pieces and it was impossible to get a total length measurement. No welding of the uranium to the clad was observed. The condition of the fuel is shown in Fig. 20. Some surface roughening occurred, particularly in the higher temperature regions. During the decanning process, examination of the upper section of the fuel element revealed the restrainer to be in good condition; in addition, the uranium was not in contact with the bottom of the restrainer. Since in both cases the growth gap was 0.284 in., the uranium growth was less than this amount.

²¹ Reactor Engineering Division Quarterly Report, ANL-5571 (July, 1956), p. 131.

²² Reactor Engineering Division Quarterly Report, ANL-5561 (April, 1956), p. 141.

²³ Ibid, ANL-5571 (July, 1956), p. 135.

76

TOP

8 in. FROM BOTTOM

BOTTOM

CP-5-7: 4% FISSION ALLOY

TOP

14 in. FROM BOTTOM

BOTTOM

CP-5-8: 3% FISSION ALLOY

FIG. 19
SURFACE APPEARANCE OF EBR-II TYPE FUEL ELEMENTS
IRRADIATED IN CP-5. (BEFORE DECANNING)

1 in. FROM TOP

8 in. FROM TOP

BOTTOM

CP-5-7: 96% U - 0.16 Zr - 2 Mo - 1.2 Ru - 0.24 Ph - 0.40 Pd

1/2 in. FROM TOP

8 1/2 in. FROM TOP

12 1/4 in. FROM TOP

CP-5-8: 96.91 U - 0.09 Zr - 0.01 Nb - 1.54 Mo - 1.16 Ru - 0.19 Rh - 0.10 Pd

FIG. 20
SURFACE OF FUEL AFTER REMOVAL OF CLADDING.
CENTRAL METAL TEMPERATURE: 750 F

Irradiation of CP-5-9²⁴ was suspended pending completion of alteration work on CP-5.

The sodium bond taken from test CP-5-5 was analyzed for the presence of fission products. CP-5-5 was a fuel element with a 0.060-inch hole through the top cap. This hole served as a channel connecting the sodium thermal bond within the fuel element to the sodium bond in the outer annulus between the jacket of the fuel and the finned tube wall; thus the fission product contaminants could pass freely through this hole from the inner to the outer bond.

The fuel, which was 2% Zr-98% U enriched to 20%, was irradiated to an estimated burnup of 0.77 total atom-per cent. During the irradiation period this sample was exposed to 219 thermal cycles. The maximum central metal operating temperature was approximately 800F, while the average central metal temperature was 760F.

Four representative products were selected to determine bond contamination: barium-140, strontium-89, ruthenium-160, and cerium-144. Both the inner and the outer bond were analyzed.

For any element examined, less than 1% of the inner bond contamination was found in the outer bond. Also, in each case the contamination in either bond was very much smaller than the fission yield of these elements for this burnup.

Element	μg of Element Formed in Fuel	μg of Element in Total Bond		Ratio of Outer Bond/ Inner Bond ($\times 10^{-2}$)
		Inner Bond	Outer Bond	
Ba-140	38,200	0.385	0.00373	0.968
Sr-89	18,300	11.7	.0314	.268
Ru-160	2,280	4.62	.00226	.0439
Ce-144	34,100	37.0	0.0992	0.269

C. Iron-Uranium Compound Formation at Sodium Wetted Uranium-Stainless Steel Interfaces - R. A. Stella

Since irradiation growth of the fuel pin in an EBR-II fuel element would result in contact, under pressure, of the sodium-wetted uranium and the stainless steel clad, a series of tests are in progress to investigate diffusion and compound formation at such an interface and to develop a means of inhibiting the same.

²⁴ Reactor Engineering Division Quarterly Report - Section I, ANL-5571, (July, 1956) p. 135.

29

An initial test showed surface treatments which are general practice, or arise from manufacturing processes, have no inhibiting effect on compound formation at $1300\text{F} \pm 15\text{F}$. A second test indicated chrome plate is an effective barrier to compound formation at this temperature.

A third test was made to determine the chrome plate thickness necessary to provide an effective barrier between the clad and fuel pin material. Test temperature was $1400\text{F} \pm 15\text{F}$; duration 36 days; interface pressure 1500 and 250 psi. Chrome plate thicknesses were: 0.001, 0.002, 0.003, and 0.004 in. The results showed that (1) no compound formation occurred and (2) the thickness of chrome plate need not exceed 0.001 in.

D. EBR-II Primary Sodium System Piping and Insulation - H. R. Heisler, R. W. Seidensticker

1. Insulation Requirements

The section of the piping investigated connects the upper plenum of the reactor vessel to the heat exchanger and is submerged in static sodium. The pipe is constructed of 16-inch Schedule 30, Type 304 stainless steel and is approximately 15 ft long. For present purposes, the normal (full power) flow through the pipe is taken as 10,000 gpm of sodium at 900F, and the temperature of the bulk sodium surrounding the pipe as 720F. The rate of heat transfer between the sodium being pumped through this section of pipe and the stagnant bulk sodium surrounding the pipe (for the uninsulated case) has been calculated at approximately 225,000 Btu/(hr)(ft). This value represents an unacceptably high thermal stress (about 32,000 psi) in the pipe wall; in addition, the heat loss itself, although largely recoverable, is somewhat higher than desired (about 1.6% of total reactor power).

The basic insulation requirements for this system are: reliability, compatibility with sodium, resistance to radiation damage, and economic feasibility. In order to meet some of the above requirements, it was found that a system employing an inner and outer pipe with the insulation contained in the annulus between the two pipes would have to be used. The outer pipe used in this study (20-inch Schedule 10 stainless steel) forms an annulus of 1-3/4 in. between it and the 16-inch Schedule 30 stainless steel inner pipe.

The relative heat transfer properties and thermal stresses for the various systems considered are summarized in Table XII.

In systems 1 and 4 the annular region between the inner and outer pipes is leaktight. This requirement induces the problems of differential thermal expansion, high thermal stresses, and high fabrication costs.

Table XII

SYSTEM CHARACTERISTICS

No.	Type of Insulation in Pipe Annulus	Average Heat Transferred to Bulk Sodium, Btu/(hr)(ft)	Approximate Maximum Thermal Stress Under Scram Conditions, psi	Degree of Relia- bility	Rela- tive Cost
1	Aerogel (LK-61)	96	28,200 (tension)	Poor	5
2	1-in. dia. hollow SST balls	Not calculated	Not calculated	Good	100
3	1-in. dia. SST tubes; 2 rows deep; strapped to pipe	21,800	Not calculated	Good	20
4	Inert gas	7,000	28,200 (tension)	Poor	4
5	Stagnant sodium	94,200	20,600 (tension)	Excellent	1

In system 3 the outer pipe is not required because the hollow tubes are strapped to the inner pipe. The magnitude of the thermal stresses in this design approaches the value calculated for systems 1 and 4. System 3 eliminates the problem of differential thermal expansion but the cost of fabrication is beyond the range of economic feasibility.

In systems 2 and 5 the annular region between the inner and outer pipes is not leaktight; therefore the problems of differential thermal expansion and leak integrity are eliminated. However, the fabrication cost of system 2 is beyond the range of economic feasibility. Although system 5 allows the most heat to be transferred, it has the lowest thermal stresses and appears to be the most reliable and economical system studied. The thermal stress incurred during steady-state operation is about 11,000 psi, and the total rate of heat transfer through the pipe is about 0.7% of full reactor power.

2. Calculation of Transient Thermal Stresses

When the reactor is scrammed the average temperature of the sodium flowing through the 16-inch Schedule 30 pipe will begin to decrease from approximately 900F and will approach 720F in a relatively short period of time. Although thermal shock of the pipe under these conditions is not considered severe, reasonably high transient thermal stresses will occur. Approximations of the magnitude of these stresses have been determined (Table XII).

81

The method used to calculate the thermal stresses for systems 1 and 4 is described in detail in the Liquid Metals Handbook.²⁵

The transient thermal stresses for system 5 were calculated from the temperature profiles shown on Fig. 21. These temperature profiles were plotted from points found from a numerical solution which consisted, briefly, of calculating heat transfer between pipe inner surface (T_1), pipe (T_2), and annulus (T_4) during each of many successive time intervals. The duration of each time interval satisfied the transient conduction criterion

$$(\Delta x)^2 / \epsilon \Delta t \geq 2$$

Pipe outer surface temperatures were then determined by calculating heat transfer between pipe (T_2), pipe outer surface (T_3), and annulus (T_4). It was assumed that the temperature of the inner surface of the pipe decreased from approximately 900F at a constant rate of 25F/sec (which is probably quite conservative).

The following equation was used to calculate the biaxial thermal stresses occurring at the inner surface of the inner pipe:

$$\sigma = \frac{E \alpha (T_m - T_i)}{1 - \mu},$$

where

E = Modulus of elasticity

α = Coefficient of thermal expansion

T_m = Mean pipe wall temperature

T_i = Temperature at inner surface of pipe

μ = Poisson's ratio.

The maximum value of σ for system 5 occurs when the temperature of the inner surface of the pipe wall is approximately 720F. The average ordinate, T_m , at this time is found from the method of equal areas and is equal to 775.5F. Therefore, the maximum biaxial thermal stress for system 5 is:

$$\sigma_{\max} = \frac{(26)(10)(775.5 - 720)}{1 - 0.3} = 20,600 \text{ psi (tensile).}$$

It is emphasized that all of the thermal stress calculations presented here are based on an elastic analysis and the final results apply only to the regions of the pipe away from the end connections.

²⁵ Na-NaK Supplement, TID-5277 (July 1, 1955) pp. 221-222.

R.W. SEIDENSTICKER: D.O., 12-11-56

FIG. 21
CALCULATED TEMPERATURE PROFILES
TROUGH INNER PIPE

E. EBR-II Control Rod Mechanism - Mark-A - E. Hutter,
R. W. Seidensticker

The temporary test setup of the Mark-A control rod mechanism and the installation of instruments have been completed. Tests have been conducted with 85-psi initial pressure in the air cylinder and lead weights fastened to the shortened tubing (to bring the weight of the scrambled part of the mechanism to about 327 lb). The calculated weight of the actual EBR-II assembly, including the control rod and allowing for sodium displacement, is 316 lb.

Scram release times of the roller latches have been measured to determine the time delay between actuation of the scram button and the start of the vertical scram movement. The release time varies with air cylinder pressure; with 85 psi, the release time varies between 0.003 and 0.007 sec, and increases to 0.011 sec with no pressure.

Figure 22 shows the air cylinder coupled with the hydraulic shock absorber. The 14-inch stroke of the control rod is divided into 10 in. of accelerated movement and 4 in. of deceleration by the shock absorber. The impact of the piston on the shock absorber causes the maximum decelerating force. For the tests conducted thus far, a cylindrical dash pot has been used with the shock absorber piston having a straight 3-7/8 in. taper with a 0.005- to 0.035-in. annulus.

Figure 23 is a plot of velocity versus displacement observed for a constant cylinder pressure of 85 psig. Figure 24 shows the calculated and measured displacement as a function of time for the same condition. The peak deceleration force under this condition was calculated to be 9,800 lb. Figures 25 and 26 show similar curves for limited cylinder air volume (air pressure decreasing from 85 to 23.6 psig from start to end of stroke). The peak deceleration force for this condition was calculated at 5,400 lb.

F. Bellows Seal and Sodium Condensation Test Apparatus -
E. Hutter, O. S. Seim

The test apparatus shown in Fig. 27 has been in operation since May, 1955. Its purpose is to test the performance of welded bellows as seals for vertically moving shafts. The inside of the bellows is exposed to sodium vapor. Both the design of the seal and the temperature distribution during testing are representative of those contemplated for certain applications in the EBR-II. The apparatus also is equipped to measure the sodium condensation rate along a vertical tube in three places varying in distance from the sodium surface.

FIG. 22
AIR CYLINDER
HYDRAULIC SHOCK ABSORBER

RE-7-19373-A

FIG. 24
DISPLACEMENT VS TIME FOR SCRAMMED MARK-A CONTROL
ROD WITH CONSTANT CYLINDER PRESSURE OF 85 psig

FIG. 23
MEASURED SCRAM VELOCITY VS DISPLACEMENT
FOR MARK-A CONTROL ROD WITH CONSTANT
CYLINDER PRESSURE OF 85 psig

85

86

FIG. 26
MEASURED DISPLACEMENT VS TIME FOR SCRAMMED
MARK-A CONTROL ROD WITH CYLINDER PRESSURE
DECREASING FROM 85 psig (START) TO
23.6 psig (END OF STROKE)

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FIG. 25
MEASURED SCRAM VELOCITY VS DISPLACEMENT FOR
MARK-A CONTROL ROD WITH CYLINDER PRESSURE
DECREASING FROM 85 psig (START)
TO 23.6 psig (END OF STROKE)

FIG. 27
BELLOWS SEAL AND SODIUM CONDENSATION
TEST APPARATUS

The test setup employs a shaft (2 in. dia. x 8 ft-6 in. long) which is oscillated vertically by a motorized rack and pinion drive. The upper end of the shaft is sealed to a platform by a stainless steel bellows which operates at room temperature. The lower end of the shaft is submerged in sodium held at 750F. A shaft guide tube has an aluminum bronze bearing also submerged in sodium. Argon blankets the sodium. The argon pressure varies from 15 to 30 in. of water with extension and contraction of the bellows.

The bellows is of welded construction, fabricated of stainless steel Type 304 disks (0.010 in. thick). It expands and compresses from $4\frac{1}{2}$ to $19\frac{1}{2}$ in. at a rate of 2 in./min.

The bellows has completed 20,000 full cycles without incident. Due to the relatively short time of operation (2 months), virtually no sodium vapor or oxide has been deposited on the inner surfaces.

The rate of sodium condensation was measured over a period of 22 days. The sodium buildup at a point one foot above the sodium level was equivalent to 0.09 in./yr; at higher levels the amount was negligible. The tests are continuing.