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EBWR TEST REPORTS

Compiled by

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TABLE OF CONTENTS

	<u>Page</u>
Numerical Index of Test Reports	7
Contributors.	11
Acknowledgements.	12
Topical Outline with ANL References	13
Introduction	15

SECTION I SYSTEM COMPONENT TESTS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
1	Startup Heater Test, A. Gibson	33
3	Shield Tank Leakage Test, V. C. Hall	39
4A	Emergency Cooler Performance, A. S. Jameson.	40
4B	Emergency Cooler Test, G. E. Robinson	41
5, 5A, 5B, 5C, 5D	Condenser Performance, A. Gibson, N. Kershaw, C. A. Hatstat	49
12	Sensitivity of Ground Detector, S. J. Robb	78
14A	Water Losses from EBWR, A. Gibson	81
16	Condenser Evacuation Rate, V. C. Hall	85
17	Steam Bypass Valve - Generator Load, V. C. Hall	86
18	Shield Cooling System, V. C. Hall.	88
19	Test of High Pressure Boric Acid System, E. A. Wimunc	88
21	Condenser Inleakage Detection, H. Y. Cadoff	95
27	Steam Power Relief Valve, V. C. Hall	99
28	Reactor Vessel Operation Data, G. E. Robinson	101
32	Test Heat Exchanger, A. Gibson.	116
36, 53, 53A	Turbine Inspection, H. L. Gann, R. E. Bailey, C. A. Hatstat	117

SECTION I SYSTEM COMPONENT TESTS (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
38	EBWR Pressure Vessel Gasket Closure, J. M. Barrett	122
58	Examination of the Kanigan Plating in the EBWR Main Steam Line, R. E. Bailey	122
74A	Containment Vessel Air Temperature, L. J. Weidner . .	125
78C	Building Shell Pressure Leak Rate Test, D. J. Roy . . .	126
101A	Turbine Seal Leak Tests, D. R. Jacobson, H. M. Sroka .	129
103	Volume of Water in the Reactor, C. E. Carson	132

SECTION II POWER OUTPUT

8	Power Output versus Galvanometer Reading, J. O. Schuyler	137
11	Decay Heating and Power Calibration Tests, A. Gibson .	140
23, 23A, 23B	Power Run with Zero Steam Bypass, W. C. Lipinski . .	156

SECTION III RADIATION AND FLUX

7	Plant Radiation Levels, C. B. Zitek	163
25, 25A	EBWR Flux Plots, R. E. Bailey	169
25B	First Annual Core Inspection, C. V. Pearson	173
34	Prediction of N ¹⁶ Gamma Activity, A. Gibson, N. A. Kershaw	180
34A	Distribution of N ¹⁶ in EBWR, M. H. Theys, R. L. Mittl .	191
35	Flux Distribution in Tangential Instrument Tube, A. Hirsch, W. Lipinski	201
47	Neutron Survey Beneath EBWR, S. Meyers	201
49	Gamma Flux Determination of EBWR Core, C. A. Pesce, J. F. Pohlman	205
49A	Measurement of Gamma Flux in EBWR, C. A. Pesce, J. F. Pohlman	211
56	Radioactive Crud Cleanup in Nozzles Below EBWR Reactor, E. A. Wimunc	219

SECTION III RADIATION AND FLUX (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
71	Determination of Thermal Neutron Flux in EBWR by Cobalt Activation, R. E. Bailey, R. F. S. Robertson . . .	221
102, 102A, 102B	Relative Effectiveness of Sb-Be Source and Core Induced Neutrons, A. Hirsch	227

SECTION IV CONTROL AND CONTROL RODS

2	Graphs of Rod Position versus Time of Operation, J. O. Schuyler	235
9, 9B	Control Rod Positions and Power Calibration, J. A. DeShong	236
31, 42D, 77	Control Rod Data, A. J. Lizasoain, J. A. Thie, J. F. Pohlman	243

SECTION V PHYSICS

41C	EBWR Void Reactivity versus Power, J. A. DeShong, W. C. Lipinski	247
42, 42B, 42C	Reactivity Changes and Irradiation Effects, D. E. McFeron, J. A. Thie	247
100C	Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth, K. K. Almenas	256

SECTION VI DYNAMICS TESTS

26	Some Kinetics Experiments in EBWR, J. A. Thie	265
41, 41A, 41B	Reactor Transfer Function Measurements, A. Hirsch, W. C. Lipinski	270
43A	Analysis of EBWR Control Rod Ejection Experiment, S. Jacobs, J. A. Thie	273
54B	Operation at 62 Mw, E. A. Wimunc	279
54C	Stability Analysis of 60 Mw Run, J. A. Thie	282
72A	Demonstration of the UTB Voidmeter, J. A. Thie, J. Beidelman	284

SECTION VII FLUID CHARACTERISTICS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
39	Preliminary Determination of Recirculation Rates in EBWR, S. A. Bernsen	293
39A	Recirculation Flow Rate Tests, W. A. Sutherland.	295
39B	EBWR Recirculation Rate, A. Hirsch	297
39E	Further Recirculation Flow Rate Tests, W. A. Sutherland.	298
39G	Measurement of Inlet Subcooling for the Determination of Recirculation Flow Rates, B. M. Hoglund.	303
62, 62A, 62B	Water Level Measurements in the Reactor Vessel, R. L. Mittl, M. Theys.	305

SECTION VIII FUEL

51	Evaluation of the Fuel Transfer Coffin, S. Meyers.	323
55	First Defect Test in EBWR, R. F. S. Robertson	325
64	Fuel Transfer Radiation Survey, C. V. Pearson, L. M. Rentschler.	346
75	Probe of Fuel Channels with a 0.050 Inch Clearance Gage, D. J. Roy, E. A. Wimunc	356

SECTION IX WATER CHEMISTRY AND CORROSION

13A	Water Technology of EBWR During Initial Operation, C. B. Zitek	361
15	Ion Exchange Column No. 1 Resin Removal, V. C. Hall.	376
22	Water Decomposition as a Function of Reactor Power, R. E. Bailey.	377
22A	Water Decomposition at Varying Reactor Powers and Pressures, R. E. Bailey, E. W. O'Keefe	379
29	Deposits from a Fuel Element and Dummy Fuel Elements, C. B. Zitek	386
29A	Scale on Fuel Elements in EBWR, R. H. Leyse	389
29B	Test for Scale Removal from Fuel Assembly H-18, I. Charak, R. H. Leyse	392

SECTION IX WATER CHEMISTRY AND CORROSION (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
29C	Post-irradiation Heating of Fuel Element H-18, R. H. Leyse	401
29D	Visual Inspection of Fuel Assemblies During Core Loading, I. Charak.	408
30	Deposits on the Steam Side of the Condenser and Steam Dryer, C. B. Zitek	410
37	Reactor Steam Purity by Radioactive Sodium Tracer, J. Barrett, H. Cadoff	415
66	Effect of H ₂ Addition to Reactor Feedwater on Radiolytic Decomposition, G. E. Goring, A. P. Gavin. .	419
73A	Measurement of Suspended and Dissolved Solids in EBWR Water, N. R. Grant.	427
79	Corrosion of EBWR Steam Line Samples, N. R. Grant .	429
104	Radiochemical Analysis of Samples from the EBWR Shock Shield, R. J. Arami.	438
Bibliography.		441

NUMERICAL INDEX OF TEST REPORTS

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
1	Startup Heater Test.	I	33
2	Graphs of Rod Position versus Time of Operation.	IV	235
3	Shield Tank Leakage Test.	I	39
4A	Emergency Cooler Performance	I	40
4B	Emergency Cooler Test	I	41
5, 5A, 5B, 5C, 5D	Condenser Performance	I	49
7	Plant Radiation Levels	III	163
8	Power Output versus Galvanometer Reading . .	II	137
9, 9B	Control Rod Positions and Power Calibration . .	IV	236
11	Decay Heating and Power Calibration Tests . .	II	140
12	Sensitivity of Ground Detector	I	78
13A	Water Technology of EBWR During Initial Operation.	IX	361
14A	Water Losses from EBWR	I	81
15	Ion Exchange Column No. 1 Resin Removal . . .	IX	376
16	Condenser Evacuation Rate	I	35
17	Steam Bypass Valve - Generator Load	I	86
18	Shield Cooling System	I	88
19	Test of High Pressure Boric Acid System	I	88
21	Condenser Inleakage Detection.	I	95
22	Water Decomposition as a Function of Reactor Power	IX	377
22A	Water Decomposition at Varying Reactor Powers and Pressures	IX	379
23, 23A, 23B	Power Run with Zero Steam Bypass	II	156
25, 25A	EBWR Flux Plots	III	169
25B	First Annual Core Inspection.	III	173

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
26	Some Kinetics Experiments in EBWR	VI	265
27	Steam Power Relief Valve.	I	99
28	Reactor Vessel Operation Data	I	101
29	Deposits from a Fuel Element and Dummy Fuel Elements	IX	386
29A	Scale on Fuel Elements in EBWR.	IX	389
29B	Test for Scale Removal from Fuel Assembly H-18.	IX	392
29C	Post-irradiation Heating of Fuel Element H-18.	IX	401
29D	Visual Inspection of Fuel Assemblies During Core Loading	IX	408
30	Deposits on the Steam Side of the Condenser and Steam Dryer.	IX	410
31, 42D, 77	Control Rod Data.	IV	243
32	Test Heat Exchanger	I	116
34	Prediction of N^{16} Gamma Activity.	III	180
34A	Distribution of N^{16} in EBWR	III	191
35	Flux Distribution in Tangential Instrument Tube	III	201
36, 53, 53A	Turbine Inspection.	I	117
37	Reactor Steam Purity by Radioactive Sodium Tracer	IX	415
38	EBWR Pressure Vessel Gasket Closure	I	122
39	Preliminary Determination of Recirculation Rates in EBWR.	VII	293
39A	Recirculation Flow Rate Tests.	VII	295
39B	EBWR Recirculation Rate	VII	297
39E	Further Recirculation Flow Rate Tests	VII	298
39G	Measurement of Inlet Subcooling for the Determination of Recirculation Flow Rates . . .	VII	303
41, 41A 41B	Reactor Transfer Function Measurements . . .	VI	270
41C	EBWR Void Reactivity versus Power	V	247

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
42, 42B, 42C	Reactivity Changes and Irradiation Effects. . . .	V	247
43A	Analysis of EBWR Rod Ejection Experiment. . .	VI	273
47	Neutron Survey Beneath EBWR	III	201
49	Gamma Flux Determination of EBWR Core . . .	III	205
49A	Measurement of Gamma Flux in EBWR.	III	211
51	Evaluation of the Fuel Transfer Coffin	VIII	323
54B	Operation at 62 Mw	VI	279
54C	Stability Analysis of 60 Mw Run.	VI	282
55	First Defect Test in EBWR.	VIII	325
56	Radioactive Crud Cleanup in Nozzles Below EBWR Reactor	III	219
58	Examination of the Kanigan Plating in the EBWR Main Steam Line	I	122
62, 62A, 62B	Water Measurements in the Reactor Vessel. . .	VII	305
64	Fuel Transfer Irradiation Survey.	VIII	346
66	Effect of H ₂ Addition to Reactor Feedwater on Radiolytic Decomposition	IX	419
71	Determination of Thermal Neutron Flux in EBWR by Cobalt Activation.	III	221
72A	Demonstration of the UTB Voidmeter	VI	284
73A	Measurement of Suspended and Dissolved Solids in EBWR Water	IX	427
74A	Containment Vessel Air Temperature.	I	125
75	Probe of Fuel Channels with a 0.050 Inch Clearance Gage.	VIII	356
78C	Building Shell Pressure Leak Rate Test	I	126
79	Corrosion of EBWR Steam Line Samples.	IX	429
100C	Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth.	V	256
101A	Turbine Seal Leak Tests.	I	129

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
102, 102A, 102B	Relative Effectiveness of Sb-Be Source and Core Induced Neutrons	III	227
103	Volume of Water in the Reactor	I	132
104	Radiochemical Analysis of Samples from the EBWR Shock Shield	IX	438

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TOPICAL OUTLINE WITH ANL REFERENCES

Section I deals primarily with tests conducted to evaluate the performance of various systems and components of the reactor complex. Additional data on tests in this area may be found in ANL-5726, ANL-5941 and ANL-6189.

Section II covers tests conducted for purposes of power output and instrument calibration.

Section III contains information on tests conducted to determine plant radiation levels under various conditions, deals with the N^{16} problem in the reactor and associated units, and includes tests conducted to evaluate the source strength and adequacy of existing instrumentation for reactor startup.

Section IV contains data on control rods, rod calibration and associated tests.

Sections V and VI cover the physics, reactivity, transfer function and oscillation tests conducted. Additional data concerning this area may be found in ANL-5799, ANL-5850, ANL-5798, ANL-5697, ANL-5849, ANL-5711.

Section VII includes testing conducted to establish operating water levels, recirculation rates subcooling and interface phenomena.

Section VIII covers fuel, fuel defect tests, and fuel element handling. Further data on this area are contained in ANL-6022 and ANL-6091.

Section IX deals with the water chemistry of the system. Areas covered are water treatment, water decomposition, steam purity, corrosion and crud. A survey of this area is covered in ANL-6136.

INTRODUCTION

This collection of Test Reports presents the results of various experiments and tests conducted on the Experimental Boiling Water Reactor (EBWR) at Argonne National Laboratory. The reports selected for publication here represent only a part of the over-all experimental work with EBWR. They were selected on the basis of general value to those interested in nuclear science and engineering.

It must be remembered that these reports were often written to distribute the data and results promptly on problems of immediate interest and are not finished documents. However, they are valuable as a continuing record of experiments and tests of the reactor complex.

It should be emphasized that these test reports and data apply only to the complex under investigation. Any extrapolation of these data to other systems or installations should be carefully reviewed. Opinions and conclusions expressed are those of the test report author and do not necessarily reflect those of the Laboratory.

The period covered is December 1956, to July 1959. Reactor operation during that time is summarized as:

Generator	8,745 hr
Electric Energy	35,006 Mw hr
Reactor	11,164 hr
Energy	211,509 Mw hr

Figure I-1 shows the core layout for load No. 46 which was the initial loading after criticality tests. The core consisted of 114 fuel elements with four different types of fuel plate and enrichment arrangements. The central section contained 54 thin fuel elements which were enriched 1.44 percent. Two thin elements with natural uranium fuel were installed in the actual center to serve as flux depressors. Fifty thick fuel elements, enriched 1.44 percent, were placed to surround the central section in a concentric form. The circular four foot diameter core geometry was completed by the addition of six natural fuel thick plate elements and two additional enriched fuel thin elements. The neutron source used was an antimony-beryllium element located at the outside of the core.

Changes made in the core fuel elements are shown in the core loading charts for loads No. 47 through 56 and in Figs. I-2 through I-11. Load No. 46 was in service from December 1956, to January 1958. Figures I-12, I-13 and I-14 show reactor and generator operating levels by months for 1957, 1958 and 1959. Significant activities are listed on the 1958 and 1959 charts.

Fig. I-1. Core Loading No. 46

Core loading #46, the final loading after criticality testing, was used from December 1956 to January 1958. Core changes thereafter were directed by the following persons.

Loading No. 47 by W. J. Kann
Loading No. 48 by C. V. Pearson
Loading No. 49 by C. V. Pearson
Loading No. 50 by C. V. Pearson
Loading No. 51 by C. V. Pearson

Loading No. 52 by C. V. Pearson
Loading No. 53 by C. V. Pearson
Loading No. 54 by W. J. Kann
Loading No. 55 by V. M. Kolba
Loading No. 56 by V. M. Kolba

Element			Element		
<u>No</u>	<u>From</u>	<u>To</u>	<u>No</u>	<u>From</u>	<u>To</u>
Dummy	Core Index 0-278	Storage Pit	H-18	Core Index 27-255	Core Index 27-219
Dummy	Core Index 0-384	Storage Pit	H-11	NEW(Instrumented)	Core Index 27-255
T-23	Core Index 129-53	Storage Pit	H-13	NEW(Instrumented)	Core Index 27-43
T-21	Core Index 129-264	Storage Pit	Dummy	NEW(Instrumented)	Core Index 0-67
ET-46	Core Index 23-82	Core Index 129-264	Dummy	NEW(Instrumented)	Core Index 0-173
ET-18	Core Index 27-219	Core Index 129-53	Dummy	NEW(Instrumented)	Core Index 0-278
H-15	Core Index 27-43	Core Index 23-82	Dummy	NEW(Instrumented)	Core Index 0-384

Fig. I-3. Core Loading No. 48
April 10, 1958

Element No.	From	To	Element No.	From	To
H-11	Core Index 27-255	Storage Pit	Dummy	Core Index 0-67	Storage Pit
H-13	Core Index 27-43	Storage Pit	Dummy	Core Index 0-173	Storage Pit
ET-40	Core Index 77-276	Core Index 27-255	Dummy	Core Index 0-278	Storage Pit
T-7	NEW(Fission Product Specimen)	Core Index 77-276	Dummy	Core Index 0-384	Storage Pit
T-13	NEW	Core Index 27-43	Dummy	NEW	Core Index 0-67, 0-173, 0-278 and 0-384

Fission product release sampling tubes were installed in

Element No.	Core Index Location
T-7	77-276
ET-20	77-254
ET-14	77-64

Fig. I-4 Core Loading No. 49
July 16, 1958

<u>Element No.</u>	<u>From</u>	<u>To</u>
T-7	Core Index 77-276	Storage Pit
T-15	NEW(Fission Product Specimen)	Core Index 77-276

Fission product release sampling tubes installed in

<u>Element No.</u>	<u>Core Index Location</u>
T-15	77-276
ET-20	72-254
ET-14	77-64
ET-49	77-191

Fig. I-5. Core Loading No. 50
November 10-14, 1958

Element No.	From	To
T-15	Core Index 77-276	Storage Pit
T-13	Core Index 27-43	Storage Pit
ET-40	Core Index 27-255	Core Index 77-276
H-7	NEW	Core Index 27-255
H-9	NEW	Core Index 27-43

Water level measuring tubes and instrumentation were attached to fuel elements as follows

Element No.	Core Index Location	Element No.	Core Index Location
ET-49	77-191	H-10	0-301
ET-46	129-264	ET-9	77-297

Fig. I-6. Core Loading No. 51
December 11-12, 1958

Element No	From	To	Element No	From	To
H-7	Core Index 27-255	Storage Pit	T-9	NEW(Fission product specimens and sampling tubes)	Core Index 72-254
H-15	Core Index 23-82	Storage Pit			
EH-17	Core Index 63-101	Core Index 23-82	Dummies	Core Index 0-227, 0-53, 0-240, 0-407	Storage Pit
ET-6	Core Index 72-43	Core Index 63-101			
EH-21	Core Index 63-312	Core Index 27-255	Dummy	NEW(Corrosion samples)	Core Index 0-53
ET-20	Core Index 72-254	Core Index 63-312		NEW(Water sampling tube)	Core Index 0-227
ACF	NEW	Core Index 72-43		NEW(Thermocouple)	Core Index 0-407
				NEW	Core Index 0-240

Water level sample tubes were installed in elements EH-25 at core index location 59-381

Fig. I-7. Core Loading No 52
February 21, 1959

Element <u>No</u>	<u>From</u>	<u>To</u>	Element <u>No</u>	<u>From</u>	<u>To</u>
T-9	Core Index 72-254	Storage Pit	EH-17	Core Index 23-82	Core Index 63-101
Dummy	Core Index 0-407	Storage Pit	T-13	Core Index 72-254	Core Index 27-255
T-13	Storage Pit	Core Index 72-254	ET-20	Core Index 63-312	Core Index 72-254
ACF	Core Index 72-43	Core Index 23-82	EH-21	Core Index 27-255	Core Index 63-312
ET-6	Core Index 63-101	Core Index 72-43	Dummy	NEW	Core Index 0-407

Instrumentation for flux measurement and steam sampling was installed in the following elements

<u>Element No</u>	<u>Location (Core Index)</u>	<u>Element No</u>	<u>Location (Core Index)</u>
ET-5	129-370	ET-15	101-160
ET-51	101-266	ET-25	101-371
ET-31	80-237	EH-42	54-390
EH-44	17-203	ET-38	77-402
ET-43	101-54

Fig. I-9. Core Loading No. 54
June 9, 1959

<u>Element No.</u>	<u>From</u>	<u>To</u>
Dummy	Core Index 0-227	Storage Pit
Dummy	NEW	Core Index 0-277

Instrumentation for water sampling was installed in:

<u>Element No.</u>	<u>Location</u>
ET-49	77-191
ET-53	89-158
ET-5	129-370
ET-38	77-402
Dummy	0-227

Fig. I-10. Core Loading No. 55
 February 29 - March 3, 1960
 (V. M. Kolba)

The core was reloaded into the reactor from the storage pit after completion of vessel modification

Fig. I-11 Core Loading No. 56
April 12, 1960
(V. M. Kolba)

Element No	From	To	Element No	From	To
Dummy	Core Index 0-159	Storage Pit	ET-18	Core Index 129-53	Core Index 63-312
Dummy	Core Index 0-265	Storage Pit	S-2	NEW	Core Index 129-53
Dummy	Core Index 0-371	Storage Pit	EH-17	Core Index 63-101	Core Index 0-39
Dummy	Core Index 0-39	Storage Pit	ET-46	Core Index 129-264	Core Index 63-101
EH-15	Core Index 63-206	Core Index 0-159	S-3	NEW	Core Index 129-264
EH-5	Core Index 129-370	Core Index 63-206	EH-18	Core Index 63-418	Core Index 0-371
S-1	NEW	Core Index 129-370	ET-54	Core Index 129-159	Core Index 63-418
EH-21	Core Index 63-312	Core Index 0-265	S-4	NEW	Core Index 129-159

Fig. I-12. Operating levels in 1957

Fig. I-13. Operating levels in 1958

Fig. I-14. Operating levels in 1959

SECTION I SYSTEM COMPONENT TESTS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
1	Startup Heater Test, A. Gibson	33
3	Shield Tank Leakage Test, V. C. Hall.	39
4A	Emergency Cooler Performance, A. S. Jameson	40
4B	Emergency Cooler Test, G. E. Robinson.	41
5, 5A, 5B, 5C, 5D	Condenser Performance, A. Gibson, N. Kershaw, C. A. Hatstat.	49
12	Sensitivity of Ground Detector, S. J. Robb.	78
14A	Water Losses from EBWR, A. Gibson	81
16	Condenser Evacuation Rate, V. C. Hall.	85
17	Steam Bypass Valve-Generator Load, V. C. Hall	86
18	Shield Cooling System, V. C. Hall	88
19	Test of High Pressure Boric Acid System, E. A. Wimunc	88
21	Condenser Inleakage Detection, H. Cadoff.	95
27	Steam Power Relief Valve, V. C. Hall	99
28	Reactor Vessel Operation Data, G. E. Robinson.	101
32	Test Heat Exchanger, A. Gibson	116
36, 53, 53A	Turbine Inspection, H. L. Gann, R. E. Bailey, C. A. Hatstat.	117
38	EBWR Pressure Vessel Gasket Closure, J. M. Barrett.	122
58	Examination of the Kanigan Plating in the EBWR Main Steam Line, R. E. Bailey.	122
74A	Containment Vessel Air Temperature, L. J. Weidner .	125
78C	Building Shell Pressure Leak Rate Test, D. J. Roy. . .	126
101A	Turbine Seal Leak Tests, D. R. Jacobson, H. M. Sroka.	129
103	Volume of Water in the Reactor, C. E. Carson.	132

STARTUP HEATER TEST

(Test Report No. 1, by A. Gibson)

Procedures. The test was originally intended to be a check of the startup heating system beginning with a stabilized low reactor temperature. However, it was determined that the stabilized reactor temperature was too high (170°F) to commence heating immediately. Therefore, both reactor water purification systems were placed in service to drop the temperature. The ion exchange columns and regenerative heaters were bypassed and the coolers and pumps were operated at maximum capacity. It is estimated that approximately 120 gpm total was circulated through the cooling system and back to the reactor during this period.

In order to insure that no avoidable heat losses occurred (aside from the ion exchange systems) during the test, the reactor vent was closed, and the air ejector steam (and trap) and the safety valve trap lines were closed. The 4 in. drain valve under the steam dryer was also closed.

After the reactor water temperature had dropped to 100°F (as measured on thermocouple T, Fig. 1-1), the reactor purification system was turned off and the startup heater was placed in operation. This was accomplished by first starting one circulating water pump (in order to provide a cooling sump for the heater condensate), then opening the air operated steam inlet and condensate outlet valves from the control room, and finally, by opening wide the inlet steam stop valve. Some water hammer was observed, but nothing serious.

Immediately after startup, the observed water temperature jumped from 100°F to 160°F (see Fig. 1-1). This was unexpected although it was anticipated that faulty readings would be obtained at this point until adequate circulation was obtained. For this reason, the reactor water purification system had been re-valved (after the cooling cycle) to permit use of the ion exchange pump to draw sufficient water past the thermocouple to yield temperatures accurately reflecting the condition of the water in the reactor. Only one ion exchange unit was used, and the cooling water flow was as low as possible to remove a minimum of heat. The regenerative heater was in service, and the ion exchange column was bypassed. Control was effected from the control room. The unit was turned on only long enough to obtain temperature readings (see Fig. 1-1). After sufficient circulation was established, it made little difference as far as the thermocouple was concerned whether the ion exchange pump was turned on or off.

As the reactor pressure rose, a second means of obtaining temperature was available by determining the saturation temperature corresponding to the pressure reading on the gages. After completion of the test (when 100 psig was obtained) the heater was secured and all valves, pumps, etc., returned to their original condition. Fifteen minute readings were taken throughout the test as indicated on the data sheets.

Fig. 1-1. Reactor temperature vs time for startup heating and for cooling

Discussion. As indicated above, the test was originally intended to check the startup heater system, but it was necessary to operate the cooling system to bring the reactor temperature down. The cooling data has been included in this report because of the interesting temperature gradients obtained on the vessel.

Figure 1 verifies the interesting nature of these temperature gradients. As the cooling water system is placed in operation it is shown that the bottom of the vessel is heated first as the hot water is drawn down through the core. After approximately $\frac{3}{4}$ of an hour the cooling effect is felt in the bottom of the vessel and the temperature closely follows the exit water temperature. Below the water level, but in the upper portions of the core, the vessel wall temperatures immediately respond to the cooling water and drop accordingly. At and above the water level the vessel wall temperatures are barely affected by the cooling water.

As the startup heater is placed in operation, the vessel wall temperatures below the water level follow the hot entering (not cold exit temperatures measured by T) fluid temperatures fairly closely. The upper wall temperatures above the water level do not rise until steam is present in the vessel in sufficient quantities to begin raising the reactor pressure. The bottom of the vessel temperature lags considerably behind the system temperature rise.

A maximum difference of 90°F at a given instant was observed between thermocouples 6 and 7 located adjacent to each other. The maximum rate of rise of temperature was observed on thermocouple No. 3 with a change of approximately 60°F in one fifteen minute period. In one hour this thermocouple recorded a 160°F rise which slightly exceeded the maximum allowable vessel rise of 150°F per hour.

As previously indicated, it was necessary to put the ion exchange pump in service shortly after starting the heating cycle in order to obtain a good reading on thermocouple T. The reason for the rapid rise of temperature at this point immediately after starting the heater is difficult to explain. The thermocouple is physically located a considerable distance below the heater, but apparently there was some back flow of hot water from the heater. Approximately 45 minutes were required to establish good thermal circulation in the system as indicated by the effect the ion exchange pump had on the system and by the slope of the water temperature curve. The reason for the dip near the top of the curve is not known except that the thermal siphon does not appear to be very stable.

Calculations indicate that the startup heater operates at less than its predicted design capacity by approximately 17 percent. Since the design temperature rise (100°F to 330°F) over a 2 5 hours operating period was approximately duplicated in the test, and since the saturation temperature in

the heater (as indicated by the pressure readings) was 381°F, which compares well with the 378°F design temperature, the probable reasons for the less than predicted service are as given in Conclusions.

The correlation of the calculated design steam flow rates at the beginning and end of the test compared with the test results at the beginning and end were poor:

	<u>Design</u>	<u>Test</u>
Beginning of test	7700 lb/hr	4700 lb/hr
End of test	1970 lb/hr	2200 lb/hr

This is probably due in part to inaccuracies in the flow meter.

Conclusions. It is concluded from this experiment that within the accuracies obtainable with current heat transfer and fluid flow calculation procedures, the unit is operating satisfactorily. It is operating at lower rates than anticipated by approximately 17 percent, but this deviation could be explained by a lower than expected thermal siphon rate (quite possible in view of the erratic nature of the circulation), by the difficulties encountered in calculating heat transfer coefficients (although it is noted that a scale coefficient was included in the design and the unit is still fairly new and should be relatively unfouled), and by the inaccuracies in running the experiment.

Design Conditions From Graph No. RE-7-14581-A (in Startup Heater file), dated 11-23-54, by A. Lindsay, the average reactor temperature (including equivalent metal heat) will be raised from 100°F to 327°F in 2.5 hours with heater operating at capacity.

Wt. steel in vessel	= 129,500 lb
Flange	= 9,300 lb
Shielding	= 22,000 lb
Structural core	= <u>44,200 lb</u>

Total 205,000 lb

205,000 lb metal at specific heat 0.11 Btu/lb = 22,600
 Equiv lb water wt. of reactor water (288 ft³) = 18,000 lb
 at 1.0 Btu/lb = 18,000 lb

Total equivalent weight = 40,600 lb

(a) Total heat added by startup heater

$$Q = W C_p \Delta t = 40,600 \times 1 \times (327-100) \\ = 9.22 \times 10^6 \text{ Btu}$$

- (b) The average heating rate over 2.5 hour period is:

$$q_{\text{avg}} = \frac{9.22 \times 10^6}{2.5} = 3.69 \times 10^6 \text{ Btu/hr}$$

- (c) Estimated steam consumption at beginning and end of test:

From Curve RE-7-14581-A

Slope at beginning = $40^\circ\text{F}/0.25 \text{ hours}$

$$\frac{dQ}{d\theta} = WC \frac{dT}{d\theta} = 40,600 \times 1 \times \frac{40}{0.25} = 6.5 \times 10^6 \text{ Btu/hr}$$

$$\text{Steam rate} = \frac{6.5 \times 10^6 \text{ Btu/hr}}{844 \text{ Btu/hr}} = 7700 \text{ lb/hr}$$

Slope at end = $25/0.6$

$$\frac{dQ}{d\theta} = 1.66 \times 10^6 \text{ Btu/hr steam rate} = 1970 \text{ lb/hr}$$

Test Conditions. From the plot of temperature vs time (Fig. 1) it is obvious that portions of the reactor vessel were already hot and did not have to be heated through the complete cycle by the startup heater.

From Fig. 1-1 it appears that all metal above thermocouple 7 was only heated from approximately 200°F to 330°F . Exclusive of the flange, shielding and core, this is approximately $\frac{1}{2}$ of the steel in the vessel. One-half of the shielding is also assumed to have followed this heating cycle:

- (a) Heat added to upper portion of vessel (metal)

$$\text{Vessel steel} = \frac{129,500}{2} = 64,750 \text{ lb}$$

$$\text{Shielding} = \frac{22,000}{2} = 11,000 \text{ lb}$$

Total 75,750 lb

$$75,750 \times 0.11 = 8330 \text{ lb equivalent water weight}$$

$$Q_{\text{upper vessel}} = W C_p \Delta t = 8330 \times 1 \times (330 - 200) = 1.08 \times 10^6 \text{ Btu.}$$

(b) Heat added to flange (metal)

From data sheet, thermocouple 13 recorded a temperature change from 200°F to 308°F.

$$Q_{\text{flange}} = 9300 \times 0.11 \times (308-200) = 0.11 \times 10^6 \text{ Btu}$$

(c) Heat added to bottom of $\frac{1}{2}$ of vessel and core (metal)

Vessel steel	=	64,750 lb
Shielding	=	11,000 lb
Structural core	=	44,200 lb
		<hr/>
		119,950 lb

From Fig. 1 and the data sheet it is apparent that the majority of the vessel was heated from approximately 100°F to 325°F. The exception to this is the extreme bottom which was only heated to approximately average 250°F.

Estimate average heating cycle from 100°F to 310°F

$$Q_{\text{bottom}} = 119,950 \times 0.11 \times (310-100) = 2.10 \times 10^6 \text{ Btu}$$

(d) Heat added to reactor water

From Fig. 1, the water temperature measured by thermocouple T was raised from 100°F to 322°F, and as measured by pressure gage (sat. temp.) was raised to 338°F. The upper temperature limit probably represents something between these limits for average water temperature in the vessel, or approximately 330°F.

$$Q_{\text{water}} = 18,000 \times 1 \times (330-100) = 4.14 \times 10^6 \text{ Btu}$$

(e) Estimated heating loss in IX system

From Fig. 1 and data sheets, the IX system was on for approximately one hour. It is estimated that the water flow rate thru the unit was approximately 10 gpm. The average temperature rise was 100 to 150°F.

$$Q_{\text{IX}} = 1 \text{ (hr)} \times 1 \text{ (Btu/ft-°F)} \times 10 \times 5000 \text{ (lb/hr)} \times (150-100) \text{ (°F)}$$

$$Q_{\text{IX}} = 0.25 \times 10^6 \text{ Btu}$$

(f) Total heat added by startup heater

$$\begin{aligned}
 Q_{\text{upper vessel}} &= 1.08 \times 10^6 \text{ Btu} \\
 Q_{\text{flange}} &= 0.11 \times 10^6 \\
 Q_{\text{bottom}} &= 2.10 \times 10^6 \\
 Q_{\text{water}} &= 4.14 \times 10^6 \\
 Q_{\text{IX}} &= 0.25 \times 10^6 \\
 \hline
 Q_{\text{Total}} &= 7.68 \times 10^6 \text{ Btu}
 \end{aligned}$$

(g) Average heating rate of heater

$$q_{\text{avg}} = \frac{7.68 \times 10^6 \text{ Btu}}{2.5 \text{ hrs}} = 3.07 \times 10^6 \text{ Btu/hr}$$

(h) Comparison between design and test conditions

$$\text{Performance ratio} = \frac{3.06 \times 10^6}{3.69 \times 10^6} = 0.83$$

$$100 - 83 = 17\%$$

The average heating rate (over the same range as the predicted design curve) of the actual startup heater is 17 percent less than design predictions.

REACTOR SHIELD TANK LEAKAGE TEST

(Test Report No. 3, by V. C. Hall)

A low leak rate of the reactor shield tank is required to prevent activation of argon in the air surrounding the reactor vessel. A test was conducted on the reactor shield tank to determine the leakage rate. The normal operating pressure of the unit is 1 inch water vacuum.

The following leakage rates were observed:

<u>Pressure,</u> <u>inches water vacuum</u>	<u>Leakage rate</u> <u>cubic ft/minute (STP)</u>
1	1.5
2	2.5
3	3.0
4	3.5
5	4.5
6	5.5

To date, no argon activity has been detected in any air samples in the stack discharge, therefore this leakage rate is considered satisfactory.

EBWR EMERGENCY COOLER PERFORMANCE

(Test Report No. 4A, by A. S. Jameson)

Data obtained on the cooling performance of the emergency cooler is shown in Table I and Fig. 4A-1. The test was conducted at a starting pressure of 370 psig; however, another test at 600 psig will be conducted later.

Table I - Emergency Cooler Operation.
Pressure Decrease with Time

Seconds	Psig	Seconds	Psig
0	370	390	294
10	365	420	290
20	364	450	285
30	361	480	280
40	360	510	274
50	359	540	269
60	356	570	263
75	351	600	259
90	350	630	255
105	345	660	250
120	341	690	246
135	339	720	242
150	338	750	238
165	332	780	233
180	330	810	229
195	329	840	225
210	325	870	220
225	321	900	217
240	320	930	213
270	316	960	209
300	310	990	207
330	304	1020	202
360	298

Fig. 4A-1. Emergency cooler test

EBWR EMERGENCY COOLER TEST

(Test Report No. 4B, by G. E. Robinson)

A test was run with the emergency cooler to determine the rate of pressure drop in the reactor and the rate of rise of the inlet cooling water temperature.

Test Procedure

Prior to shutting down reactor:

- (a) Reactor should be at 20 Mw, 60,000 lb/hr through by-pass valve, turbine trip and admission valves closed
- (b) Reactor pressure 600 psig, on auto control, stop position zero.
- (c) Reactor level 11 ft-11 in sight glass. Feedwater control automatic
- (d) Overhead storage tank water supply open.
- (e) Supply water valve to emergency cooler open.
- (f) Record laboratory water flow integrator reading in service building basement Laboratory water only to overhead storage tank.

Secure reactor at console:

- (a) Turn control power key switch off Log time of shutdown.
- (b) Turn feedwater control to "manual" and close the feedwater regulating valve with the manual control switch.
- (c) Set the by-pass control to "manual."
- (d) Turn off ejector steam, fans, and one circulating pump.
- (e) Restart feedwater pump No. 2, if required.

In plant action required after the reactor is shut down:

- (a) Restart feedwater pump No. 2, if required.
- (b) Open vacuum breaker, leave open.
- (c) Open the 4-in. dryer drain valve by bleeding air from the cylinder. Observe that the valve is open.

- (d) Shut the 1 in. drain underneath the dryer.
- (e) Close all drains, i.e., air ejector, relief line, and main steam line.

In control room:

- (a) Stop ion exchange system.
- (b) Close main steam stop and gland seal steam valves after turbine vacuum is zero.

In control room:

- (a) Initiate emergency cooler operation from Panel No. 5. Log time the switch is turned to "open" position.

In plant:

- (a) Station a man with a telephone to advise what happened to the cooler when emergency cooling is initiated.
- (b) Observe equipment performance in the attic after flow is quiet.
- (c) Periodically observe the behavior of dryer, storage tank, and piping during the test.

In control room:

- (a) Take data initially every $\frac{1}{2}$ min., i.e., pressure and temperature.

Reactor Pressure. Immediately upon shutting down the reactor, the pressure slowly began to drop. Seven minutes after shutdown the emergency cooler inlet valve was opened. At that time the reactor pressure was 580 psig. A plot of reactor pressure versus time after cooling valve opened is given in Fig. 4B-1. Ten minutes after cooling had started the pressure had dropped 115 psi. At the end of one hour the reactor pressure had dropped to 150 psig.

Fig. 4B-1. Emergency cooler test

Cooling Water Temperatures. Thermocouples to measure the inlet and the outlet temperature of the cooling water were located approximately three feet from the emergency

cooler on the cooling water inlet and outlet pipes. The thermocouple on the inlet was between the emergency cooler and the air-operated emergency cooler valve. Prior to shutting down the reactor, the temperature of the empty cooling water inlet line was 91°F (time 0113, Table I). After shutting down the reactor, this temperature went up to 211°F (time 0126, Table I). This corresponds to the approximate time of opening the hand valve upstream of the emergency cooler air-operated valve and indicates that a slight amount of water probably leaked passed the air-operated valve.

Table I - Emergency Cooler Test

Reactor power	20 Mw	Main stream stop valve	Closed
Pressure	600 psig	Feedwater pump	Running
Level	11'-11"	Feedwater regulating	Closed
Reactor shutdown time	0122:20	Feedwater stop	Open
Emerg. cooler water valve opening time	0130	Cooling water	
IX Condition	Secured	Inlet temp. prior to inject.	91½°F
Vent	Secured	Exit temp. prior to inject.	251°F
4 in. drier drain	Open	Level controller	On
1 in. drier drain	Closed	Approximate level	3 in. below overflow pipe

Time	Reactor pressure, (TV) psig	Cooling water inlet, °F	outlet, °F	Reactor level, (TV)	Time	Reactor pressure, (TV) psig	Cooling water inlet, °F	outlet, °F	Reactor level, (TV)
0113	600	91½	251	11'-11"	0142	445	68	220	11'-11"
0126	598	211	280	11'-11"	30	440			
0130	580	80	221	11'-11"	0143	440			
30*	575				30	435			
0131	570	69	216	11'-11"	0144	430	68	235	11'-11"
30	560				30	425			
0132	550	68	215	11'-11"	0145	420			
30	540				30	415			
0133	540	68	215	11'-11"	0146	410	68	215	11'-11"
30	535				30	410			
0134	530	68	215	11'-11"	0147	405			
30	520				30	400			
0135	520				0148	395			
30	510				30	390			
0136	510	68	218-280	11'-11"	0149	390			
			varying		30	380			
30	500				0150	380	68	258	12'-1"
					30	370			
0137	500				0151	365			
30	495				30	360			
0138	490	68	218-278	11'-11"	0152	355			
			varying		30	355			
30	480				0153	350			
					30	345			
0139	475				0154	340			
30	470				30	340			
0140	465	68	225	11'-11"	0155	330	68	219	12'-1"
30	460				30	330			
0141	455				0156	325			
30	450				30	320			

*30 seconds

Table I (cont'd.)

Time	Reactor pressure, (TV) psig	Cooling water inlet, °F	Cooling water outlet, °F	Reactor level, (TV)	Time	Reactor pressure, (TV) psig	Cooling water inlet, °F	Cooling water outlet, °F	Reactor level, (TV)
0157	315				0220	190	72	225	12'-1"
30	310				0221	180			
0158	310				0222	180	73	240	12'-1"
30	300				0223	174			
0159	300				0225	168	75	234	12'-1"
30	300				0227	160	80	233	12'-1"
0200	295	68	221	12'-1"	0230	149	84	231	12'-1"
30	290				0232	142	80	220	12'-0"
0201	290				0234	132	81	222	12'-0"
30	285				0236	125	82	224	12'-0"
0202	280				0238	118	82	224	12'-0"
30	280				0240	109	84	265	12'-0"
0203	280				0242	102			
30	275				0244	98			
0204	270				0246	92	91	240	12'-0"
30	265				0248	88	91	237	
0205	260	68	220	12'-1"	0250	83	89	228	12'-0"
30	260				0252	81	96	220	12'-0"
0206	255				0254	80			
30	250				0256	78	95	227	12'-0"
0207	250				0258	72			
30	245				0300	70	98	252	12'-0"
0208	240				0305	62	104	233	12'-0"
30	240				0310	58	101	253	12'-0"
0209	240				0315	52	105	228	12'-0"
30	235				0320	45	108	230	12'-0"
0210	230	68	235	12'-1"	0325	42	108	248	12'-0"
30	230				0330	40	111	246	12'-0"
0211	230				0345	35	118	241	12'-0"
30	225				0400	30	121	243	11'-11"
0212	220				0415	30	118	268	11'-11"
30	220				0430	32	119	258	11'-11"
0213	215				0445	32	126	214	11'-11"
30	215				0500	32	121	254	11'-11"
0214	210				0515	30	122	261	11'-10"
30	210				0530	30	123	246	11'-10"
0215	210	69	226	12'-1"	0545	30	123	257	11'-9"
30	205				0600	30	127	233	11'-9"
0216	200				0615	30	124	249	11'-9"
0217	200				0630	30	125	255	11'-9"
0218	195				0645	30	124	230	11'-9"
0219	190				0700	30	125	246	11'-9"

Reactor pressure versus inlet temperature of cooling water was plotted in Fig. 4B-2. The temperature of the cooling water did not increase until a pressure of 210 psig was reached. At 100 psig the cooling water inlet reached a temperature of 84°F. At the end of the test (5½ hours after the emergency cooler valve was opened) the reactor pressure was holding constant at 30 psig while the cooling water temperature was increasing and had reached a value of 127°F.

Fig. 4B-2. Reactor pressure vs inlet cooling water temperature

The outlet temperature of the cooling coils was constantly varying. At the same time, the emergency cooling water flow alarm was going on and off. A reasonable explanation would be that slugs of water flashing into steam momentarily build up a pressure equal to or greater than the 65 ft head of the storage tank, thus causing the variance in outlet temperature and stoppage of flow. This would also explain the pulsating steam flow noticed at the 15,000 gallon storage tank.

Reactor Vessel Temperatures.

Figure 4B-3 shows a temperature profile at hourly intervals as the reactor vessel was cooled down. At no time did the cooling rate exceed the manufacturer's recommendation of 185°F maximum. The maximum rate measured was at thermocouple no. 8 between 20 min. and 30 min. after shutdown. For this ten minute period the cooling rate was 26°F, or 156°F per hour. (See also Table II).

Fig. 4B-3. Reactor cooling rate

Make-up Water to the Storage Tank. Approximately 30 gallons of water was added to the storage tank during the test. However, there is some doubt as to the accuracy of this figure. It is, therefore, recommended that a check on this figure be made the next time the emergency cooler is put in operation.

Table II - Reactor Vessel Temperatures (Corrected)

Time	No. 1	No. 2	No. 3	No. 4	No. 5	No. 6	No. 7	No. 8	No. 9	No. 10	No. 11	No. 12	No. 13
0130	478	480	477	473	478	479	478	477	480	480	480	480	477
0140	474	474	471	462	465	465	465	464	476	476	476	476	474
0150	462	459	455	441	445	445	443	443	469	471	473	474	471
0200	441	436	434	416	420	420	418	417	460	464	465	470	469
0210	425	418	417	397	402	402	400	398	452	459	460	466	466
0220	406	397	396	378	381	381	380	378	447	451	454	464	466
0232	380	372	371	356	358	359	356	356	434	442	445	460	466
0240	368	360	359	344	347	347	345	345	427	436	440	457	462
0250	351	343	341	329	331	329	331	331	417	429	434	452	460
0305	328	321	321	312	312	312	311	318	403	418	424	447	457
0310	321	315	315	306	306	306	302	315	398	416	421	445	457
0320	312	306	306	297	297	296	294	312	391	-	416	441	455
0330	297	293	293	286	286	285	283	305	381	-	407	436	451
0340	291	286	286	279	279	279	278	300	376	-	403	434	448
0350	285	282	282	275	276	276	275	298	370	-	401	433	448
0400	280	278	278	272	273	273	270	295	366	-	397	430	447
0415	276	273	273	269	270	270	269	293	360	-	392	426	444
0430	273	270	269	265	267	267	266	289	352	-	390	422	441
0445	272	269	267	265	266	266	265	286	346	-	385	421	439
0500	270	267	265	263	264	264	264	284	340	-	381	416	437
0515	267	266	264	261	263	263	263	280	337	-	380	415	435
0530	266	264	263	260	261	261	260	278	331	-	379	411	432
0545	266	264	262	259	261	261	260	277	328	-	375	410	431
0600	264	263	261	258	260	260	259	274	324	-	371	408	429
0630	262	261	258	257	258	258	258	271	314	-	366	400	423
0700	261	259	257	254	255	256	256	268	310	-	363	398	420

Heat Removal. The amount of heat removed for the first hour of operation was approximately 1.6 mw excluding decay heat. This is slightly high when compared with the manufacturer's design figure of 1 mw. The accuracy of this figure depends largely on the amount of water in the reactor. This figure was taken as 25,000 lbs for a level of 12 ft (see Heat Removal Calculations).

Summary

1. The reactor pressure can be reduced from 580 psig to 30 psig in $3\frac{1}{2}$ hours with an initial cooling water temperature of 68°F.
2. The rate of cooling of the reactor vessel is well within the manufacturer's maximum of 185°F per hour.
3. The heat removed from the reactor for the first hour of operation was approximately 1.6 mw excluding decay heat.

Heat Removal Calculations

Vessel and shield weights

Weight steel in vessel	129,500 lb carbon steel
Cover plate	9,300 lb carbon steel
Thermal shield	22,000 lb stainless steel
Structural core	2,500 lb eq. wt. of water
Weight of water in reactor	25,000 lb

First Half Hour

Assume vessel can be divided in half for weight and temperature purposes and that removed from the cover plate is negligible.

Upper Half

Weight = 64,750 lb

Average temperature difference = $479^{\circ}\text{F} - 458^{\circ}\text{F} = 21^{\circ}\text{F}$

$Q = W C_p \Delta T$

$$= 64,750 \text{ lb} \cdot 21^{\circ}\text{F} \cdot \frac{0.12 \text{ Btu}}{\text{lb } ^{\circ}\text{F}} = 1.63 \times 10^5 \text{ Btu}$$

Lower Half

Weight = 64,750 lb

Average temperature difference = $476^{\circ}\text{F} - 428^{\circ}\text{F} = 48^{\circ}\text{F}$

$Q = W C_p \Delta T$

$$= 64,750 \text{ lb} \cdot 48^{\circ}\text{F} \cdot \frac{0.12 \text{ Btu}}{\text{lb } ^{\circ}\text{F}} = 3.72 \times 10^5 \text{ Btu}$$

Water

$Q = W C_p \Delta T$

Between 488°F and $420^{\circ}\text{F} = 68^{\circ}\text{F}$

$$= 25,000 \text{ lb} \cdot 68^{\circ}\text{F} \cdot \frac{1.03 \text{ Btu}}{\text{lb } ^{\circ}\text{F}} = 1.75 \times 10^6 \text{ Btu}$$

Thermal Shield

$Q = W C_p \Delta T$

$$= 22,000 \text{ lb} \cdot 68^{\circ}\text{F} \cdot \frac{1.03 \text{ Btu}}{\text{lb } ^{\circ}\text{F}} = 1.79 \times 10^5 \text{ Btu}$$

Condensed Steam

Steam volume $\approx$ 450 cu ft Press. from 585 psig to 295 psig

$$\frac{450 \text{ ft}^3}{.7765 \text{ ft}^3/\text{lb}} - \frac{450 \text{ ft}^3}{1.4944 \text{ ft}^3/\text{lb}} = 580 \text{ lb} - 301 \text{ lb} = 279 \text{ lb}$$

$$279 \text{ lb} \cdot \frac{1203 - 395 \text{ Btu}}{\text{lb}} = 2.25 \times 10^5 \text{ Btu}$$

Structural Core

Temperature varies same as water

$$Q = 2,500 \text{ lb} \cdot \frac{1 \text{ Btu}}{^\circ\text{F}} \cdot 68^\circ\text{F} = 1.7 \times 10^5 \text{ Btu}$$

Total for first half hour = $2.86 \times 10^6 \text{ Btu}$

Second Half HourUpper Half of Vessel

Average temperature difference = $458^\circ\text{F} - 428^\circ\text{F} = 30^\circ\text{F}$

$$Q = W C_p \Delta T$$

$$= 64,750 \text{ lb} \cdot 30^\circ\text{F} \cdot \frac{0.12 \text{ Btu}}{\text{lb } ^\circ\text{F}} = 2.33 \times 10^5 \text{ Btu}$$

Lower Half of Vessel

Average temperature difference = $428^\circ\text{F} - 365^\circ\text{F} = 63^\circ\text{F}$

$$Q = W C_p \Delta T$$

$$= 64,750 \text{ lb} \cdot 63^\circ\text{F} \cdot \frac{0.12 \text{ Btu}}{\text{lb } ^\circ\text{F}} = 4.8 \times 10^5 \text{ Btu}$$

Water

$$Q = W C_p \Delta T$$

Average temperature difference $420^\circ\text{F} - 365^\circ\text{F} = 55^\circ\text{F}$

$$= 25,000 \text{ lb} \cdot 55^\circ\text{F} \cdot \frac{1.03 \text{ Btu}}{\text{lb } ^\circ\text{F}} = 1.42 \times 10^6 \text{ Btu}$$

Condensed Steam

Steam vol. $\approx$ 540 cu ft

Press. from 295 psig to 149 psig

$$\frac{450 \text{ ft}^3}{1.4944 \text{ ft}^3/\text{lb}} - \frac{450 \text{ ft}^3}{2.768 \text{ ft}^3/\text{lb}} = 301 \text{ lb} - 162 \text{ lb} = 139 \text{ lb}$$

$$139 \text{ lb} \cdot \frac{1203 - 338 \text{ Btu}}{\text{lb}} = 1.2 \times 10^5 \text{ Btu}$$

Thermal Shield

$$Q = W C_p \Delta T$$

$$= 22,000 \text{ lb} \cdot \frac{0.12 \text{ Btu}}{\text{lb}^\circ\text{F}} \cdot 55^\circ\text{F} = 1.45 \times 10^5 \text{ Btu}$$

Structural Core (Temp. varies same as water)

$$Q = 2500 \text{ lb} \cdot \frac{1 \text{ Btu}}{\text{lb}^\circ\text{F}} \cdot 55^\circ\text{F} = 1.38 \times 10^5 \text{ Btu}$$

$$\text{Total for second half hour} = 2.54 \times 10^6 \text{ Btu}$$

$$\text{Total for hour} = 5.4 \times 10^6 \text{ Btu}.$$

CONDENSER PERFORMANCE

(Test Report Nos. 5, 5A, 5B, 5C, 5D, by A. Gibson,
N. Kershaw, and C. A. Hatstat)

Tests were made to compare the condenser performance with design predictions and to estimate condenser performance under 40 megawatt reactor operation. The procedure consisted of comparing condenser operating data with a performance chart which was prepared by Allis-Chalmers in accordance with the standard practice of the Heat Exchange Institute for computing condenser operation. Westinghouse Electric Corporation has prepared a slide rule called the Steam Condenser Calculator, which is based upon the same Heat Exchange Institute standards, and this slide rule was used for:

1. Checking the Allis-Chalmers chart (see Calculations)
2. Determining the actual condenser performance (see Calculations)
3. Predicting condenser performance under 40 Mw reactor load (see Performance Chart, Fig. 5-1).

Actual condenser performance was obtained by utilizing data taken: (a) during the normal course of reactor operation and, (b) during a condenser performance test run. Condenser performance was evaluated with and without the turbine in operation, and at varying loads and circulating water inlet temperatures.

Fig. 5-1. Performance chart

Test Factors. Condenser duty is calculated for different back pressures in all cases by utilizing steam flows and enthalpies. An attempt was made to correlate water flows and temperatures, but the data taken from the plant instruments was not accurate enough.

In determining condenser duty with the turbine in operation, it is necessary to know the exhaust enthalpy of the turbine steam as well as the flow rate.

Under Calculations the manufacturer's predicted turbine steam rate (adjusted for varying conditions of back pressure, load, and throttle pressure) is compared with the observed turbine steam rates. In each case the observed steam rate is considerably greater than the predicted rate. The accuracy of the bypass steam flow meter is also examined. It is noted that for the low bypass flow rates (e.g., high turbine loads) the bypass flow meter is only accurate within approximately $\pm 5,000$ lb per hr. Since this is considerably greater than one would expect the turbine steam flow to deviate from the manufacturer's predictions, the manufacturer's curve is utilized in determining steam flow rates and exhaust enthalpies for given conditions of load, throttle pressure and back pressure.

In order to determine the circulating water flow rates through the condenser, readings of the circulating water pump suction and discharge pressure gages were taken. The total dynamic head calculated from these readings was compared with the manufacturer's test curve of the pumps and the corresponding flow rate determined (Calculations). Since the gage could not be read with extreme accuracy, this method of determining flow is subject to some error which is reflected to a certain extent in the condenser performance calculations.

As indicated above, a check of the Allis-Chalmers performance curves using Heat Exchange Institute methods was made with good results (Calculations).

Fig. 5-2. Test data

From the Calculations, and from the performance curve, it is apparent that the condenser performance cannot be determined accurately with the present instrumentation. Observation of the test data in Calculations and Fig. 5-2 indicates that there is considerable disagreement between the apparent condenser pressure as obtained on the remote pressure gage (Hg absolute), and that obtained by reading the hotwell temperature and

comparing the corresponding saturation pressure. The temperature readings in every case indicate a much poorer vacuum than the pressure gage shows. If the pressure gage is correct, the condenser is operating at a cleanliness factor of approximately 90 to 100 percent, and if the thermocouple is correct, the unit is operating at a cleanliness factor of approximately 60 to 70 percent. There is evidence to indicate that the thermocouple readings are more nearly correct. There is a better correlation of data at various loads with the thermocouple than that obtained with the pressure gage. The pressure gage has been observed at low condenser loads to indicate a perfect vacuum.

If the condenser is actually operating with approximately a 65 percent cleanliness factor, the vacuum will, of course, be much poorer than expected at 40 Mw reactor load. With full bypass flow, for instance, and with a 60°F inlet circulating water temperature, the absolute pressure will be approximately 4.5 in. Hg instead of the 3.0 in. Hg indicated by the Allis-Chalmers Performance Chart for an 85 percent cleanliness factor.

Referring to Fig. 5-2 again, it is noted that for a given pressure, the hotwell temperature is higher when the condenser is operating with full bypass steam than with the turbine in service. This is to be expected, since the location of the bypass steam line is just above the hotwell, and the entering steam tends to heat the water more, resulting in less subcooling than with the turbine in operation. This difference therefore seems to indicate the existence of some subcooling during turbine operation. However, it is also noted that the saturation temperature corresponding to a given pressure reading indicates a lower temperature than the effluent condensate. This does not seem possible, since it is obviously not possible to get superheat in the effluent condensate. This, again, seems to indicate inaccurate readings on the pressure gage.

Various other phenomena have been observed in the operation of the condenser, such as a raising level with increasing vacuum, and with increasing bypass steam flow. Fluctuation of level is also observed. These phenomena can partially be explained by the position of the bypass line which permits the incoming steam to disturb the water level, but there also seems to be a correlation between level and vacuum.

Conclusions. The following is concluded from the condenser tests conducted to date:

1. Better instrumentation is required if an accurate determination of condenser performance is desired.

2. From the data available, it appears that the condenser is operating at approximately 60 to 70 percent cleanliness factor. If this is actually the case, it could be due to fouling of the tubes, or improper design, probably the former since Allis-Chalmers is a member of the Heat Exchange Institute and uses their standard methods for sizing units. This was verified to a certain extent by an actual check of their performance curve.

The possibility of design error should not be ruled out however, since in the computation of condenser performance it is necessary to choose a material and gage factor, c_m which is related to the size and thermal conductivity of the tubes. The values for most conventional tube material are standardized and published by the Heat Exchange Institute but the aluminum alloy tubes in this condenser are not included. Allis-Chalmers used a value of 0.975 for this factor (see Calculations) and this appears to be a reasonable value based upon the corresponding thermal conductivities of the alloys mentioned by the Heat Exchange Institute, and aluminum.

3. Assuming that 60 to 70 percent cleanliness factor is correct, the unit will operate at 40 Mw reactor load with approximately $1\frac{1}{2}$ in. less vacuum than that indicated on the performance curve for 85 percent cleanliness factor.
4. Operation with excess bypass steam results in less subcooling of the effluent condensate than when most of the steam first passes through the turbine.
5. Level fluctuations are caused to a certain extent by the bypass steam flow.

Recommendations. The following steps are recommended if a more accurate determination of condenser performance is desired:

1. An absolute pressure manometer (or vacuum manometer and barometer) be installed at the condenser vacuum breaker to more accurately indicate the pressure in the condenser and to serve in calibrating the existing gage.
2. The thermocouple on the condenser effluent be calibrated and the limits of accuracy determined.
3. Install a flow measuring device in the circulating water lines.
4. Additional operating data be obtained under controlled conditions.

5. The turbine steam rate be accurately determined.
6. Add a new or additional bypass steam flow meter to the bypass steam line for low steam flows since the present meter is too inaccurate at low flows and is of little value when it is most needed.

If the above steps indicate that the condenser is operating at 65 per-cent cleanliness factor and better performance is desired, the following steps are recommended:

1. A water box manhole be removed and the actual condition of the tubes be observed.
2. A water treating expert be contacted to determine if the water treatment is adequate.

Cooling Tower. The above discussion in no way reflects the effect of the cooling tower performance which may or may not be performing in accordance with design predictions.

Calculations.

1. Condenser loading at 20 Mw reactor power - all steam through bypass valve

Back press., in. Hg Abs.	1.0	2.0	2.5	3.0	4.0	5.0
Vacuum, in. Hg	29.0	28.0	27.5	27.0	26.0	25.0
Sat. temp., °F	79.03	101.1	108.7	115.0	125.4	133.8
h_{liq} , Btu/lb	47.1	69.1	76.7	83.0	93.3	101.7
h_{stm} , Btu/lb	1203	1203	1203	1203	1203	1203
Δh , Btu/lb	1156	1134	1126	1120	1110	1101
w_{stm} , lb/hr	60,000	60,000	60,000	60,000	60,000	60,000
Q_{cond} , 10^6 Btu	69.4	68.0	67.5	67.2	66.6	66.1

2. Condenser loading at 20 Mw reactor power with turbine on line at 4200 kw.

a. Turbine Steam Rate:

From data taken on 2-20-57 at 10:00

$kw = 4200$, $W_s = 60,000$, $W_{bypass} = 5,000$

$P_{cond} = 0.51$ in. Hg Abs.

$P_{reactor} = 600$ psig

$T_{cw\ in} = 53^\circ F$

$P_{throttle} = 575$ psig (590 psia)

From Turbine Design Data:

Generator cooler design for 164 kw heat pickup with generator at 5000 kw

$$\frac{164}{5000} = 0.0328 \text{ or } 3.28 \text{ percent}$$

Assuming the same ratio holds for 4200 kw gen. losses =
 $0.0328 \times 4200 = 138 \text{ kw}$

Turbine shaft kw = $4200 + 138 = 4338 \text{ kw}$

1) From flow data taken on 2-20-57

$$W_{\text{turb}} = 60,000 - 5,000 = 55,000 \text{ lb/hr}$$

for 575 psig at throttle and 0.51 in. Hg Abs.

$$T_{\text{SR}} = \frac{55,000}{4,200} = 13.1 \text{ lb/kw-hr}$$

2) From Design Data

At 4200 kw, $T_{\text{SR}} = 12.66 \text{ lb/kw-hr}$

$W_{\text{turb}} = 4200 \times 12.66 = 53,200 \text{ lb/hr}$ at 560 psig and
 2.5 in. Hg Abs.

At throttle: $P = 575 \text{ psia}$ $T = 481.6$ $h = 1203.6$ $\phi = 1.446$

Exhaust at const $P = 2\frac{1}{2} \text{ in. Hg}$ $h = 818.0$ $\phi = 1.446$

$$\phi \text{ Available Energy} = \Delta h_A = 385.6 \text{ Btu/lb}$$

From (2) shaft kw = 4338

$$4338 \times 3413 \text{ Btu/kw-hr} = 53,200 \text{ lb/hr} \times \Delta h_{\text{utilized}} \text{ Btu/lb}$$

$$\text{Utilized Energy} = \Delta h_u = \frac{4338 \times 3413}{53,200} = 278 \text{ Btu/lb}$$

$$\text{Design Turbine Efficiency (includes Exhaust Losses)} = \frac{278}{385.6} \times 100 = 72.1\% \text{ at } 4200 \text{ kw}$$

At 0.51 in. Hg Back Press , assuming 72.1% Eff. Constant,

$$\text{Throttle } P = 590 \text{ psia } \quad h = 1203.4 \quad \phi = 1.446$$

$$\text{Exhaust } P = 0.51 \text{ in Hg } \quad h = 750.0 \quad \phi = 1.446$$

$$\Delta h_A = 453.4$$

$$\text{Utilized Energy} = 0.721 \times 453.4 = 327 \text{ Btu/lb}$$

$$W_{\text{turb}} = \frac{4338 \text{ kw} \times 3413 \text{ Btu/kw-hr}}{327 \text{ Btu/lb}} = 45,300 \text{ lb/hr} = \text{calculated steam flow}$$

From (1) above measured steam flow = 55,000 lb/hr

$$\text{Exhaust Enthalpy} = 1203 - 327 = 876 \text{ Btu/lb}$$

3) From data taken on 1-31-57 at 0100.

$$\text{kw} = 4200 \quad W_s = 62,000 \quad W_{\text{bypass}} = 12,000$$

$$P_{\text{cond}} = 0.75 \text{ in. Hg Abs} \quad P_{\text{turb}} = 575 \text{ psig (590 psia)}$$

$$T_{\text{circ water in}} = 53^\circ\text{F}$$

At 0.75 in. Hg, 72.1 percent Eff.:

$$\text{Throttle } P = 590 \quad h = 1203.4 \quad \phi = 1.446$$

$$\text{Exhaust } P = 0.75 \text{ in Hg } \quad h = 765.0 \quad \phi = 1.446$$

$$\Delta h_A = 438.4$$

$$\text{Utilized Energy} = 0.721 \times 438.4 = 316$$

$$W_{\text{turb}} = \frac{4338 \times 3413}{316} = 46,800 \text{ lb/hr} = \text{calculated steam flow}$$

$$\text{Measured steam flow} = 62,000 - 12,000 = 50,000 \text{ lb/hr}$$

$$\text{Exhaust Enthalpy} = 1203 - 316 = 887 \text{ Btu/lb}$$

4) Accuracy of Bypass Steam Flow Meter

$$\text{From M.M.\&M. data, Repeatability} = \frac{1}{4} \text{ percent}$$

$$150 \text{ in water column full scale at } 100,000 \text{ lb/hr}$$

$$0.0025 \times 150 = 0.375 \text{ in.}$$

if

$$1.5 \text{ in.} = 10,000 \text{ lb/hr}$$

$$1.5 - 0.375 \cong 1.1$$

$$1.1 \text{ in.} = \left(\frac{1.1}{1.5}\right)^2 \times 10,000 = 5,370 \text{ lb/hr}$$

$$1.5 + 0.375 \cong 1.9$$

$$1.9 \text{ in.} = \left(\frac{1.9}{1.5}\right)^2 \times 10,000 = 16,000 \text{ lb/hr}$$

$\therefore$ At 10,000 lb/hr flow meter only reads within $\pm 5,000$ lb/hr

5) Actual Turbine Steam Rate:

	Calc. turbine steam rate, lb/hr	Measured turbine steam rate, lb/hr
From (2) above:	45,300	55,000
From (3) above:	50,000	46,800

Since the flow meter reads within only $\pm 5,000$ lb/hr it is not justified to indicate that the turbine is using more than the design steam flow rate for a given set of conditions. This appears to be the case, but it is not certain. Therefore, the calculated turbine steam rate shall be used for all following computations.

6) Condenser Loading at 4200 kw:

At 0.51 in. Hg Back Press. (29.49 in. Hg Vac. - referred to 30 in.)

$$Q_{\text{cond-turb}} = 45,300 (876-27) = 38.5 \times 10^6 \text{ Btu/hr}$$

At 0.75 in. Hg Back Press. (29.25 in. Hg Vac.)

$$Q_{\text{cond-turb}} = 46,800 (887-38) = 39.7 \times 10^6 \text{ Btu/hr}$$

Note: Above loadings due to turbine duty only - does not include bypass steam.

7) Condenser Loading at 4200 kw - 20 Mw on Reactor

Tabular Data for Various Back Pressures

Gen. load, kw	4200	4200	4200	4200	4200	4200	4200
Turb. shaft load, kw	4338	4338	4338	4338	4338	4338	4338
Back press., in. Hg Abs	0.5	0.75	1.0	1.5	2.0	3.0	4.0

Vacuum, in. Hg	29.5	29.25	29.0	28.5	28.0	27.0	26.0
$P_{throttle}$, psia	590	590	590	590	590	590	590
$h_{throttle}$, Btu/lb	1203	1203	1203	1203	1203	1203	1203
$\phi_{throttle}$, Btu/lb	1.446	1.446	1.446	1.446	1.446	1.446	1.446
$h_{exhaust}$, @ const ϕ , Btu/lb	750	765	776	794	806	825	838
$\Delta h_{available}$, Btu/lb	453	438	427	409	397	378	365
Est. turb. η , %	72.1	72.1	72.1	72.1	72.1	72.1	72.1
Utilized energy = Δh_{a-xy} Btu/lb	327	316	308	295	286	273	263
W_{turb} , lb/hr	45,300	46,800	48,100	50,200	51,800	54,200	56,300
$h_{steam\ condenser}$, Btu/hr	876	887	895	908	917	930	940
$h_{liq\ condenser}$, Btu/lb	27	38	47	60	69	83	93
$\Delta h_{condenser}$, Btu/lb	849	849	848	848	848	847	847
$Q_{cond-turb}$, 10^6 Btu/hr	38.5	39.7	40.8	42.6	43.9	45.9	47.7
W_{bypass} , lb/hr	14,700	13,200	11,900	9,800	8,200	5,800	3,700
Δh_{bypass} , Btu/lb	1176	1165	1156	1143	1134	1120	1110
$Q_{cond-bypass}$, 10^6 Btu/hr	17.3	15.4	13.8	11.2	9.3	5.9	3.7
$Q_{cond-total}$, 10^6 Btu/hr	55.8	55.1	54.6	53.8	53.2	51.8	51.4

3. Reactor Steam thru Bypass Valve - 40 Mw - Condenser Loading

Back press., in. Hg Abs.	2.0	3.0	4.0	5.0	6.0	6.5
Vacuum, in. Hg	28.0	27.0	26.0	25.0	24.0	23.5
h_{steam} , Btu/lb	1203	1203	1203	1203	1203	1203
h_{liq} , Btu/lb	69	83	93	102	109	112
$\Delta h_{cond.}$, Btu/lb	1134	1120	1110	1101	1094	1091

W_{steam} , lb/hr	120,000	120,000	120,000	120,000	120,000	120,000
Q_{cond} , 10^6 Btu/hr	136	134	133	132	131	131

4. Condenser Loading at 5000 kw on Turb., 40 Mw on Reactor

Gen. load, kw	5000	5000	5000	5000	5000
Turb. shaft load, kw	5164	5164	5164	5164	5164
Back press., in. Hg Abs.	1.5	2.0	3.0	4.0	5.0
Vacuum, in. Hg	28.5	28.0	27.0	26.0	25.0
P_{throttle} , psia	590	590	590	590	590
h_{throttle} , Btu/lb	1203	1203	1203	1203	1203
ϕ_{throttle} , Btu/lb	1.446	1.446	1.446	1.446	1.446
$h_{\text{exhaust @ const } \phi}$, Btu/lb	794	806	825	838	847
$\Delta h_{\text{available}}$, Btu/lb	409	397	378	365	356
Est. turb. η , percent	72.8	72.8	72.8	72.8	72.8
Utilized energy, Btu/lb	298	289	275	266	259
$W_{\text{turb.}}$, lb/hr	59,200	61,000	64,200	66,300	68,100
$h_{\text{steam condenser}}$, Btu/lb	905	914	928	937	944
$h_{\text{liq. cond.}}$, Btu/lb	60	69	83	93	102
$\Delta h_{\text{condenser}}$, Btu/lb	845	845	845	844	842
$Q_{\text{cond-turb}}$, 10^6 Btu/hr	50.0	51.6	54.2	56.0	57.4
W_{bypass} , lb/hr	60,800	59,000	55,800	53,700	51,900
Δh_{bypass} , Btu/lb	1143	1134	1120	1110	1101
$Q_{\text{cond-bypass}}$, 10^6 Btu/hr	69.4	66.9	62.5	59.6	57.2
$Q_{\text{cond-total}}$, 10^6 Btu/hr	119.4	118.5	116.7	115.6	114.6

5. Condenser Circulating Water Flow Rate

Circ. pump suct. press gage reading 7.5 in. Hg Vac. 3.7 psi

Circ. pump discharge press gage reading 21 psig

Differential Press = 24.7 psi

TDH = $2.31 \times 24.7 = 57.0$ feet

From pump mfr's test curve,

Flow rate of pump = 6,650 gpm

$\times 2$

Total condenser flow = 13,300 gpm

Condenser design flow rate = 13,650 gpm

6. Test Data

Time	Date	Gen. kw	Rctr. stm. flow, lb/hr	Cond. circ. wtr. temp , °F		Cond. press. in. Hg Abs.	Cond. press. corres. to sat. press. of hotwell	Hotwell temp., °F	Throttle press., psia
				In	Out				
a) With Turbine on Line									
2100	1-10-57	4200	61,000	50	62	0.83	1.10	82	590
0100	1-31-57	4200	62,000	53	64	0.75	1.10	82	590
1000	2-20-57	4200	60,000	53	63	0.51	1.00	79	590
1300	3-8-57	4250	59,500	56	67	0.85	1.18	84	590
1200	3-9-57	4200	60,000	58	67	0.92	1.25	86	590
2200	2-14-57	4200	62,000	59	70	0.95	1.21	85	590
0100	4-6-57	4300	60,000	62	72	1.20	1.42	90	585
0600	4-8-57	4200	59,000	58	68	1.00	1.29	87	580
b) With Turbine Inoperative - all steam bypassed to condenser									Rctr. Press.
2000	1-7-57	0	59,000	50	62	1.2	1.51	92	595
0500	1-10-57	0	60,000	57	68	1.5	1.82	98	600
1400	2-19-57	0	58,000	54	67	1.0	1.42	90	560
1400	4-5-57	0	60,000	64	--	1.91	2.05	102	600
2000	4-9-57	0	60,000	63	76	1.50	1.87	99	600
2100	4-9-57	0	60,000	61	74	1.45	1.71	96	600
2350	4-9-57	0	60,000	60.8	73	1.35	1.66	95	600
0900	4-10-57	0	60,000	60	74	1.4	1.71	96	600
1700	4-12-57	0	60,000	61	74	1.45	1.76	97	600

7. Check on Predicted Performance Curve Using Westinghouse Condenser Slide Rule Based upon Heat Exchange Institute Data

For $\frac{7}{8}$ in. x 17 BWG 20-0 in. single pass alum. tubes, 13,650 gpm, 85 percent cleanliness factor, 5300 ft².

Tube constant K = 0.1624 (slide rule)

Gage factor cm = 0.975 (estimated from slide rule)

$$V = \frac{G \times K \times L}{A} = \frac{13,650 \times 0.1624 \times 20}{5300} = 8.37 \text{ ft/sec}$$

Choose inlet water temperature = 60°F, from slide rule:

$$\alpha = 0.357 = \frac{R}{\theta} = \frac{\text{water temp rise}}{t_v - t} \quad \begin{array}{l} t_v = \text{condensing temperature} \\ t = 60^\circ\text{F} = \text{inlet water temperature} \end{array}$$

Choose condenser duty = 60×10^6 Btu/hr

$$60.0 \times 10^6 = W \times C_p \times R = 13,650 \times 500 \times 1 \times R$$

$$R = \frac{60.0 \times 10^6}{500 \times 13,650} = 8.79^\circ\text{F}$$

$$\theta = \frac{R}{\alpha} = \frac{8.79}{0.357} = 24.6^{\circ}\text{F} = t_v - t = t_v - 60$$

$$t_v = 24.6 + 60 = 84.6^{\circ}\text{F} \quad P = 1.20 \text{ in. Hg Abs.}$$

On 30 in. scale

$$\text{Vacuum} = 30.00 - 1.20 = 28.8 \text{ in. Hg}$$

From Allis Condenser performance chart,

$$\text{Vacuum} = 28.8 \text{ in. Hg}$$

At 80°F inlet water temp

$$\alpha = 0.395$$

$$Q = 100 \times 10^6 \text{ Btu/hr}$$

$$R = \frac{100 \times 10^6}{500 \times 13,650} = 14.65$$

$$\theta = \frac{14.65}{0.395} = 37.1$$

$$t_v = \frac{80}{117.1} \rightarrow 3.20 \text{ in.}$$

$$30 - 3.2 = 26.8 \text{ in. Vacuum}$$

From Allis chart Vacuum = 26.8

8. Comparison of Predicted and Test Performance of Condenser

a) With turbine in operation

From estimated average inlet water temperature vs back pressure test data (see performance chart Fig. 5-1)

Average inlet temperature = 58°F at 29.06 in. Hg Vac, (0.94 in. Abs) and 54.7×10^6 Btu/hr

From (5) of Calculations,

Condenser circ. water $\approx 13,000$ gpm

$$V = \frac{G \times K \times L}{A} = \frac{13,000 \times 0.1624 \times 20}{5300} = 7.96 \text{ ft/sec}$$

$$R = \frac{Q}{W C_p} = \frac{54.7 \times 10^6}{500 \times 13,000} = 8.42^{\circ}\text{F}$$

Assume 100% cleanliness factor

From slide rule, $\alpha = 0.416$

$$\theta = \frac{R}{\alpha} = \frac{8.42}{0.416} = 20.2^{\circ}\text{F}$$

$$t_v = 20.2 + 58 = 78.2^\circ\text{F or } 0.97 \text{ in. Hg Abs.} \\ 29.02 \text{ in. Hg Vac.}$$

From test data above, vacuum = 29.06 in. Hg Vac.

$\therefore$ Condenser cleanliness factor $\cong 100$ percent

(Assuming press. gage in in. Hg Abs. is correct)

If thermocouple is correct, $t_v = 86^\circ\text{F}$, $P \cong 1.25 \text{ in. Hg Abs.}$
28.75 in. vacuum

$$\theta = 86 - 58 = 28$$

$$\alpha = R/\theta = \frac{8.42}{28} = 0.301$$

From Westinghouse slide rule

Cleanliness factor 67 percent

(Assuming thermocouple is correct)

b) With turbine inoperative - bypass system on

From estimated average inlet water temperature vs back pressure test data (see performance chart, Fig. 5-1)

Average inlet temperature = 60°F at 28.65 in. Hg Vac. (1.35 in. Hg Abs.)
and $70 \times 10^6 \text{ Btu/hr}$

Assuming pressure gage is correct:

$$V = 7.96 \text{ ft/sec [from (8)]}$$

$$R = \frac{70 \times 10^6}{500 \times 13,000} = 10.8$$

$$t_v = 88.36 \text{ for } 1.35 \text{ in. Hg}$$

$$\theta = -60 + 88.36 = 28.36$$

$$\alpha = \frac{10.8}{28.36} = 0.380$$

From slide rule, cleanliness factor $\cong 88$ percent

Assuming thermocouple is correct:

$$t_v = 95^\circ\text{F} \quad P = 1.66 \text{ in. Hg} \\ (28.34 \text{ in. Vac.})$$

$$\theta = 95 - 60 = 35$$

$$\alpha = \frac{10.8}{35} = 0.308$$

Cleanliness factor = 67 percent

From Test Data on 4-11-57 - Bypass Control

Steam flow, lb/hr	50,000		40,000		30,000	
Meas. method of back press.	Press. Ga.	Thermo.	Press. Ga.	Thermo.	Press. Ga.	Thermo.
Back press., in. Hg Abs.	1.10	1.47	0.74	1.18	0.44	0.82
Vacuum, in. Hg.	28.90	28.53	29.26	28.82	29.56	29.18
h_{steam} , Btu/lb	1203	1203	1203	1203	1203	1203
h_{liquid} , Btu/lb	50	59	38	52	23	41
Δh , Btu/lb	1153	1144	1165	1151	1180	1162
Q_{cond} , 10^6 Btu/hr	57.6	57.2	46.6	46.1	35.4	34.9
V , ft/sec	7.96	7.96	7.96	7.96	7.96	7.96
R , °F	8.86	8.80	7.16	7.09	5.45	5.37
t_v , °F	82	91	70	84	55	73
Inlet temp, °F (circ. W)	59	59	57	57	52	52
θ , °F = $t_v - t_{\text{inlet}}$	23	32	13	27	3	21
$\alpha = R/\theta$	0.386	0.275	0.551	0.263	1.82	0.255
TTD = $\theta - R$, °F (Not less than 5°)	14	13	8	20	5	16
Cleanliness factor, percent	90	60	100	57	--	59
Hotwell level, inches	6.0	6.0	6.8	6.8	7.3	7.3

Instrument Check. A Zimmerli absolute pressure gage, a vacuum gage and barometer were installed locally at the vacuum breaker of the main condenser to check the condenser instrumentation and operation (see Recommendations). Periodic readings were taken as indicated in Table I.

Discussion and Conclusions. Examination of the data indicates that the most consistent results were obtained with the absolute pressure Zimmerli gage. Readings obtained on this gage consistently showed that the hotwell condensate was very slightly subcooled (as measured by the remote indicating thermocouple).

It is also apparent from the data that the absolute pressure as recorded in the control room reads from 0.3 to 0.4 in. of mercury lower than it should.

A check of the above performance data against the condenser performance graph (Fig. 5-1) again shows that the condenser appears to be operating at between 60 and 70 percent cleanliness factor.

Table I

	Date Time	5/27/57 1420	5/30/57 0900	6/3/57 0920	6/4/57 0900	6/4/57 1150
Local cond. press.	Zimmerli	1.64" (94.6°)	2.10" (102.8°)	1.75" (96.7°)	1.91" (99.6°)	2.13" (103.3°)
	Vacuum	27.9"	27.5"	27.85"	27.65"	27.40"
	Barometer	-	29.5"	29.44"	29.39"	29.37"
	Barom. vac.	-	2.00" (101.1°)*	1.59" (93.6°)*	1.74" (96.5°)*	1.97" (100.6°)*
Remote cond. press.	Abs. press.	1.33" (87.9°)	1.78" (97.3°)	1.33" (87.9°)	1.50" (91.7°)	1.80" (97.7°)
	Vacuum	27.8"	27.4"	27.8"	27.6"	27.4"
Hotwell temp.		94°	102°	97°	99°	103°
Circ. water temp.	Inlet	68	74	71	73	76
	Outlet	78	85	81	84	87
Steam flow	Reactor	60,000	60,000	60,000	59,500	54,000
	Bypass	6,000	8,000	6,000	5,000	25,000
Turbine load, kw		4,100	4,000	4,050	4,050	2,050

Notes: All pressure and vacuum readings are shown in inches of mercury.

(*)Indicates the saturation temperature corresponding to the pressure shown:

Future Condenser Performance Checks.

1. Pitot tubes have been ordered for installation in the circ. water lines to check the water flow rates.
2. During the next plant shutdown the waterbox manhole will be removed and the condition of the tubes will be observed.
3. The absolute pressure gage in the control room for measuring condenser pressure will be re-calibrated.

Circulating Water Flow. To gain additional information on the condenser performance of EBWR, a pitot traverse of the circulating water inlet pipe was taken. The traverse was run at two locations, 90 degrees apart, at one inch intervals across the inlet pipe (Fig. 5-3). The data of velocity as a function of diameter is shown in Table II and Fig. 5-4. The average velocity based on equal areas was found to be 5.70 ft per sec corresponding to 11,940 gal per min. The condenser cleanliness factor based on this circulating water flow is approximately 53 percent versus the design 85 percent cleanliness factor. Operation under this reduced cleanliness factor causes an increase of 0.65 in. Hg abs. in condenser back-pressure at the present maximum reactor heat loading of 20 Mw.

R	Position	Area	Velocity ft/sec	
			West	East
1	0.55 in.	33.024 in. ²	4.42	4.67
2	2.36		4.93	5.17
3	4.24		5.42	5.62
4	6.55		5.86	6.02
5	9.92		6.36	6.43
6	19.08		6.66	6.46
7	22.45		6.44	6.18
8	24.76		6.25	5.94
9	26.64		6.03	5.69
10	28.45		5.79	5.43

Avg. Velocity = 5.79 ft/sec

$$5.79 \text{ ft/sec} \times \frac{660.5}{144} \text{ ft}^2 \times 7.48 \text{ gal/ft}^3 \times 60 \text{ sec/min} \\ = 11,940 \text{ gal/min}$$

Total Area = 660.5 in.²*

*Equal areas per National Assoc. of Fan Man. Bulletin #110
First Edition, 1950

Fig. 5-3. Pitot traverse locations

Table II

Point No.	East stop cock			West stop cock		
	ΔP^{**} , in. TBM	ΔP , in. H ₂ O	Flow,* ft/sec	ΔP^{**} , in. TBM	ΔP , in. H ₂ O	Flow,* ft/sec
28	0	-	-	0	-	-
27	3.40	6.63	4.79	3.00	5.86	4.49
26	3.80	7.32	5.02	3.40	6.64	4.78
25	4.05	7.90	5.22	4.50	8.79	5.53
24	4.25	8.29	5.34	4.35	8.49	5.41
23	4.90	9.56	5.74	4.65	9.07	5.59
22	5.10	9.95	5.85	4.80	9.30	5.66
21	5.50	10.72	6.08	5.10	9.95	5.86
20	5.95	11.60	6.32	5.75	11.22	6.21
19	6.15	12.00	6.42	6.45	12.59	6.58
18	6.10	11.90	6.40	5.65	11.03	6.16
17	6.10	11.90	6.40	6.35	12.40	6.53
16	6.15	12.00	6.42	5.90	11.50	6.29
15	6.25	12.20	6.48	6.15	12.00	6.42
14	6.40	12.50	6.56	6.65	12.97	6.69
13	6.45	12.57	6.58	6.50	12.68	6.60
12	6.65	12.96	6.68	6.75	13.19	6.69
11	6.20	12.10	6.45	6.65	12.97	6.68
10	5.90	11.50	6.29	6.50	12.68	6.60
9	6.00	11.70	6.35	6.50	12.68	6.60
8	5.85	11.41	6.27	6.20	12.10	6.45
7	5.80	11.41	6.27	6.35	12.39	6.53
6	5.70	11.12	6.19	5.55	10.82	6.10
5	5.70	11.12	6.19	5.95	11.60	6.32
4	5.50	10.72	6.07	6.10	11.90	6.40
3	5.20	10.15	6.01	5.90	11.50	6.29
2	5.00	9.75	5.79	5.80	11.41	6.27
1	4.80	9.37	5.68	5.40	10.54	6.02

*Water temp. = 51.5°F at time of readings both circ.
water pumps on - reactor cold.

**Meriam fluid No. 3 - Specific gravity 2.95

$$\text{Flow, ft/sec} = V = .8 \sqrt{2gh} = 1.855 \sqrt{h} \quad \text{where } h \text{ is in inches of H}_2\text{O}$$

Fig. 5-4. Circulating water velocity vs diameter

Alleviation of this poor condenser performance can be realized by cleaning the condenser tubes of the accumulation on the cooling water side; however, at present no solution to the problem of keeping the water side of the aluminum condenser tubes clean has been realized due to the type of material used and present cooling water tower water treatment.

Condenser Cleaning. The EBWR main condenser was chemically cleaned to remove deposits on the water side of the aluminum tubes. The cleaning operation was done in four stages: (a) a cleaning solution was circulated through the condenser for 30 minutes to remove scale and deposits; (b) the condenser was drained and flushed with water; (c) a second cleaning solution was left in the condenser for 24 hours to remove traces of copper; (d) the copper removal solution was drained to the retention tanks, followed by one flushing operation, then the circulating water pumps were started to complete flushing. Total time required was about 48 hours. The solutions used were those recommended by the Alcoa Research Laboratories and Allis-Chalmers. Equipment and solutions were furnished by Solvent Service Inc.

Visual examination of tube interiors following the cleaning operation led to the conclusion that the cleaning operation had been effective in that the tubes appeared to be free of scale and deposit.

Post-Cleaning Performance. The test was conducted for the purpose of determining the effect of the cleaning operation on condenser performance with emphasis on the cleanliness factor. (Table III.) Earlier tests were handicapped by a lack of accurate temperature measuring devices and knowledge of turbine exhaust heat rates at various loads and condenser pressures. In this test, mercury-in-glass thermometers with scale divisions of 0.1°F were used to measure circulating water and hotwell temperatures. In addition, a Zimmerli absolute pressure gauge, a vacuum gauge and barometer, and a mechanical standard absolute pressure gauge were

employed to measure condenser pressure. Studies of turbine-generator steam rates, losses and efficiencies were undertaken in an attempt to determine exhaust steam enthalpy and flow rate at various loads and back pressures.

Table III - Condenser Performance (Assuming Constant C.W. Flow Rate of 11,940 gpm)

Date	5-27-57*	6-3-57*	6-4-57*	3-25-58**	3-25-58**	3-25-58**
Time	1420	0920	1150	0820	0900	1300
C.W. flow rate, gpm	11,940	11,940	11,940	11,940	11,940	11,940
Rise (C.W. out - C.W. in) °F	10	10	11	9.3	9.1	5.5
Heat added to C.W., Btu/lb	9.98	9.98	10.97	9.3	9.1	5.5
Heat added to C.W., Btu/hr	59.3×10^6	59.3×10^6	65.2×10^6	55.3×10^6	54.2×10^6	32.8×10^6
Range (vac. temp. - C.W. in) °F	26.6	25.7	27.3	27.2	27.14	18.53
Terminal difference, °F	16.6	15.7	16.3	17.9	18.04	13.03
Log mean temperature difference, °F	21.3	20.2	21.5	22.2	22.5	15.65
Heat transfer area, ft ²	5300	5300	5300	5280	5280	5280
Heat transfer rate, Btu/hr-ft ² -°F MD	526	555	573	474	458	398
Temperature correc- tion, F_t	.985	1.005	1.033	.905	.905	.855
Tube mat'l and gauge, factor, F_{tm}	.975	.975	.975	.975	.975	.975
Water velocity, f_{ps}	7.3	7.3	7.3	7.3	7.3	7.3
Base heat transfer, HEI	710	710	710	710	710	710
Cleanliness factor, percent	77	79.8	82.2	75.8	73.1	67.2

*Test data using control board instruments prior to cleaning.

**Test data using local calibrated instruments after cleaning.

Generator efficiency was estimated from an empirical formula which has been found to fit NEMA requirements for machines of this size. Turbine mechanical losses were estimated using a formula from the same source. These formulae are given in Post-Cleaning Calculations.

Turbine efficiency studies are tabulated in Turbine Efficiency and Exhaust Heat Flow with curves plotted on Figs. 5-5 through 5-8. These studies are based on steam rate and steam rate correction data, supplied

with the turbine, and are not the results of tests. However, at normal loads, they are within the accuracy of the method of determining circulating water flow, and are adequate for the purpose of this report.

Fig. 5-5. Turbine-condenser heat flow sheet 1

Fig. 5-6. Turbine-condenser heat flow sheet 2

Fig. 5-7. Generator efficiency

Fig. 5-8. Generator efficiency

Two methods were used to calculate the heat transfer rate of the condenser:

1. Compute turbine exhaust heat to the condenser and from this determine the heat transfer rate of the condenser. Circulating

water flow is computed from the measured rise in its temperature and the heat transfer rate.

2. Assume a circulating water flow rate and, from its temperature rise, determine the condenser heat transfer rate.

The former yields circulating water flow rates substantially lower than would appear to be reasonable, in view of previous tests. The latter method indicates higher rates of exhaust heat than would be expected from Allis-Chalmers' data, particularly at low loads. However, the deviations do not amount to more than 10 percent in either case. The use of manufacturer's curves to determine circulating water flow from suction and discharge pressure readings was discarded because of the difficulty in reading the gauges combined with the low slope of the curve.

An attempt was made to check the cleanliness factors determined prior to cleaning by using the method of this report. No reasonable correlation could be found, as the cleanliness factor for the previous tests was consistently higher than that following the cleaning operation. These results are attributable to the following:

1. Uncertainty of the rate of circulating water flow. A 10 percent error in this measurement appears directly in the cleanliness factor, if heat transfer rates are calculated from circulating water temperature rise and flow rate. (Post-Cleaning Calculations.)
2. Inaccurate circulating water temperature measurements. All tests prior to cleaning used control board recording instruments to obtain these data. A check of this measurement was made in the current test and deviations of the order of -10 percent appear in the calculations of circulating water temperature rise. This error appears directly in the cleanliness factor.

Post-Cleaning Test Conclusions.

1. The improvement in cleanliness factor achieved by cleaning the condenser is indeterminate because of instrument errors in tests conducted prior to cleaning combined with lack of data on the circulating water flow rate.
2. The present cleanliness factor lies between 69 and 76 percent.
3. Circulating water flow rate is substantially lower than the design figure of 13,650 gpm.

Post-Cleaning Calculations.

1. Steam rate = guaranteed rate x correction factor = 11.9 lb/kw hr
 Guaranteed rate taken from A-C design data at 560 psig throttle pressure, 2.5 in. Hg abs. back pressure. Correction factor from A-C curve 03-056-003, Fig. 5-7.
2. Steam flow = steam rate (lb/kw hr) x generator load (kw)
 $= 11.9 \times 5000 = 59,300 \text{ lb/hr.}$
3. Heat available = throttle enthalpy - exhaust enthalpy (constants)
 $= 1203 - 766 = 437 \text{ Btu/lb}$
4. Power at coupling = generator load x $\frac{1}{\text{gen. efficiency}}$
 from Fig. 5-8, eff. at 5 Mw = 0.968
 Power at coupling = $5000 \times \frac{1}{.97} \text{ kw} = 5160 \text{ kw}$
5. Power at coupling, Btu/lb = $5160 \text{ kw} \times 3413 \text{ Btu/kw hr} \times \frac{1}{\text{flow, lb/hr}}$
 $= 5160 \times 3413 \times \frac{1}{59,300} = 297 \text{ Btu/lb}$
6. Turbine mech. loss (Kent's Mech. Eng. Handbook, 12th Ed., Power, p. 8-69)
 $= \frac{.04}{\sqrt{\text{rating Mw}}} \times \text{rating (kw)} = \frac{.04 \times 5000}{\sqrt{5}} = 89 \text{ kw}$
7. Turbine mech. loss (Btu/lb) = $89 \text{ kw} \times 3413 \text{ Btu/kw hr} \times \frac{1}{\text{flow rate, lb/hr}}$
 $= 89 \times 3413 \times \frac{1}{59,300} = 5.12 \text{ Btu/lb}$
8. Total heat utilized (Btu/lb) = Power at coupling turbine mech. loss
 $= 297 + 5.12 = 302 \text{ Btu/lb}$
9. Exhaust enthalpy = throttle enthalpy - heat utilized
 $= 1203 - 302 = 901 \text{ Btu/lb}$
10. Engine Efficiency = $\frac{\text{heat out}}{\text{heat available}} = \frac{5000 \text{ kw} \times 3431 \text{ Btu/kw hr}}{437 \text{ Btu/lb} \times 59,300 \text{ lb/hr}}$
 $= 65.8 \text{ percent}$
11. Heat to condenser = exhaust enthalpy x flow rate
 $= 901 \text{ Btu/lb} \times 59,300 \text{ lb/hr} = 53.5 \times 10^6 \text{ Btu/hr}$

$$\begin{aligned}
 12. \text{ Generator efficiency} &= 0.98 - \frac{0.055}{\sqrt[3]{Mw \text{ rating}}} \times \frac{\text{rating (kw)}}{\text{load (kw)}} \\
 &= 0.98 - \frac{0.055}{\sqrt[3]{5}} \times \frac{5000}{5000} \\
 &= 0.98 - 0.032 = 0.948
 \end{aligned}$$

Condenser performance:

1. Heat removed with condensate = condensate enthalpy x feedwater flow
 $= 48.82 \text{ Btu/lb at } 81.8^\circ\text{F} \times 60,000 \text{ lb/hr}$
 $= 2.88 \times 10^6 \text{ Btu/hr}$
2. Heat to cooling water = heat to condenser - heat removed in condensate
 $= 54.8 \times 10^6 - 2.88 \times 10^6 = 51.9 \times 10^6 \text{ Btu/hr}$
3. Heat to cooling water (Btu/lb steam)
 $= \frac{\text{Heat to cooling water (Btu/hr)}}{\text{steam flow (lb/hr)}} = \frac{51.9 \times 10^6}{60,000} = 893 \text{ Btu/lb}$
4. Heat to C.W. (Btu/lb) = C.W. temperature rise ($^\circ\text{F}$) x 1 Btu/ $^\circ\text{F}$ - lb
 $= 9.3 \text{ Btu/lb}$
5. Calc. C.W. flow = heat to C.W. (Btu/hr) x $\frac{1}{\text{heat to C.W. (Btu/lb)}}$
 $\times \frac{1}{500} \text{ gpm/lb hr}$
 $= \frac{51.9 \times 10^6}{500 \times 9.3} = 11.150 \text{ gpm}$
6. Log mean temperature difference = $\frac{\text{rise (}^\circ\text{F)}}{\left(\frac{\text{range}}{\text{temp. diff.}}\right)} = \frac{9.3}{\left(\frac{27.2}{17.9}\right)} = 22.2^\circ\text{F}$
7. Cleanliness factor = $\frac{\text{Heat transfer rate (Btu/hr-ft}^2\text{-}^\circ\text{F MD)}}{\text{base H.T. rate x temp. correction x tube factor}}$
 $= \frac{445}{687 \times 0.905 \times 0.975} = 73.5 \text{ percent}$

Turbine Efficiency and Exhaust Heat Flow.

Assume: Throttle steam at 560 psig, steam rate per Fig. 5-7.

Determine: Exhaust steam conditions at various back pressures, loads

A. <u>Generator load</u> , kw	5000	5000	5000	5000	5000
Back pressure,					
in. Hg Abs	1.0	1.5	2.0	2.5	3.0
Steam rate, lb/kw hr	11.9	12.08	12.3	12.55	12.85
Steam flow, lb/hr	59,300	60,200	61,200	62,700	64,200

Throttle enthalpy, Btu/lb	1203.5	1203.5	1203.5	1203.5	1203.5
Exhaust enthalpy, Btu/lb	766	794	806	816	825
Heat available, Btu/lb	437	409	397	387	378
Power at coupling, kw	5160	5160	5160	5160	5160
Power at coupling, Btu/lb	297	292	287	281	273
Turbine mech. loss, kw	89	89	89	89	89
Turbine mech. loss, Btu/lb	5.12	5.04	4.96	4.86	4.72
Total heat utilized, Btu/lb	302	297	292	286	278
Exhaust enthalpy, Btu/lb	901	906	911	917	925
Exhaust entropy, Btu/lb	1.676	1.654	1.632	1.623	1.620
Engine efficiency, %	65.8	69.2	70.3	70.3	70.3
Exhaust moisture, %	18.6	18.8	18.9	18.6	18.2
Heat to condenser, Btu/hr	53.5×10^6	54.5×10^6	55.7×10^6	57.5×10^6	59.3×10^6
B. <u>Generator load</u> , kw	3750	3750	3750	3750	3750
Back pressure, in. Hg Abs	1.0	1.5	2.0	2.5	3.0
Steam rate, lb/kw hr	11.95	12.2	12.5	12.8	13.15
Steam flow, lb/hr	44,800	45,600	46,800	47,900	49,200
Throttle enthalpy, Btu/lb	1203	1203	1203	1203	1203
Exhaust enthalpy, Btu/lb	766	794	806	816	825
Heat available, Btu/lb	437	409	397	387	378
Power at coupling, kw	3810	3810	3810	3810	3810
Power at coupling, Btu/lb	290	285	278	272	264
Turbine mech. loss, kw	89	89	89	89	89
Turbine mech. loss, Btu/lb	6.77	6.65	6.48	6.33	6.17
Total heat utilized, Btu/lb	297	292	284	278	270

Exhaust enthalpy, Btu/lb	906	911	919	925	933
Exhaust entropy, Btu/lb	1.686	1.658	1.647	1.636	1.635
Exhaust moisture, %	18.1	18.3	18.0	17.9	17.3
Engine efficiency, %	65.3	68.7	68.7	69.0	68.9
Heat to condenser, Btu/hr	40.6×10^6	41.5×10^6	43×10^6	44.3×10^6	45.8×10^6
C. <u>Generator load</u> , kw	2500	2500	2500	2500	2500
Back press., in. Hg Abs	1.0	1.5	2.0	2.5	3.0
Steam rate, lb/kw hr	12.3	12.62	13.0	13.4	13.8
Steam flow, lb/hr	30,750	31,600	32,500	33,500	34,500
Heat available, Btu/lb	437	409	397	387	378
Power at coupling, kw	2650	2650	2650	2650	2650
Power at coupling, Btu/lb	294	286	278	270	262
Turbine mech. loss, kw	89	89	89	89	89
Turbine mech. loss, Btu/lb	9.8	9.6	9.3	9.0	8.75
Total heat utilized, Btu/lb	304	296	287	279	271
Exhaust enthalpy, Btu/lb	899	908	916	924	932
Exhaust entropy, Btu/lb	1.673	1.652	1.642	1.635	1.632
Exhaust moisture, %	18.8	18.6	18.3	17.9	17.5
Engine efficiency, %	63.5	66	66.2	65.8	65.5
Heat to condenser, Btu/hr	2.76×10^7	2.87×10^7	2.98×10^7	3.09×10^7	3.21×10^7
D. <u>Generator load</u> , kw	1250	1250	1250	1250	1250
Back press., in. Hg Abs	1.0	1.5	2.0	2.5	3.0
Steam rate, lb/kw hr	13.75	14.3	14.85	15.4	16.05
Steam flow, lb/hr	17,200	17,900	18,550	19,250	20,020
Heat available, Btu/lb	437	409	397	387	378
Power at coupling, kw	1378	1378	1378	1378	1378

Turbine mech. loss, kw	89	89	89	89	89
Turbine wheel power, kw	1467	1467	1467	1467	1467
Turbine wheel power, Btu/lb	291	279	270	260	249
Exhaust enthalpy, Btu/lb	912	924	933	943	954
Exhaust entropy, Btu/lb	1.68	1.681	1.671	1.668	1.671
Exhaust moisture, Btu/lb	18.3	17.1	16.8	16.1	15.3
Engine efficiency, %	56.8	58.3	58	57.3	56.4
Heat to condenser, Btu/hr	15.7×10^6	16.5×10^6	17.3×10^6	18.1×10^6	19.1×10^6

Condenser Test Data

Date of test: 3-25-58

Date of Cleaning: 3-5, 6-58

Design Data: Single pass, 1157 tubes, 20 ft effective length, $\frac{7}{8}$ in. #17 BWG,
Al clad 3003 wall, 7072 clad.

Time	0820	0845	0900	1300
------	------	------	------	------

Local Data

Zimmerli gauge, cm Hg, abs	3.1	2.95	2.98	-
Vacuum, in. Hg	28.5	28.5	28.5	-
Barometer, in. Hg	29.55	29.55	29.55	29.55
Absolute press. Ga., m. Hg, abs	1.19	1.183	1.2	0.745
C.W. temperature inlet, °F	57.2	57.2	57.5	51.7
C.W. temperature outlet, °F	66.5	66.6	66.6	57.2
Hotwell temperature, °F	81.8	81.8	82.1	66.2
C.W. pump #1 suct. press., m. Hg	-3	-3	-3.5	-
C.W. pump #1 disch. press., psig	26	24	23.5	-
C.W. pump #2 suct. press., in Hg	-3	-3	-3	-
C.W. pump #2 disch. press., psig	26	24	24	-

Gauge Board Data

Condenser press., in.Hg abs	1.05	1.07	1.08	0.57
Hotwell temperature, °F	82	82	82.5	-
C.W. inlet temp., °F	58	58.5	59	-

C.W. outlet temp., °F	67	67	67.5	-
Steam flow lb/hr	58,000	57,000	57,500	30,000
Turbine load, kw	4200	4150	4180	2200

Condenser Performance after Cleaning

Tests on: 3-25-58		CH-6-16-58		
Time	0820	0845	0900	1300
Generator load, kw	4200	4150	4180	2200
Condenser press., in. Hg abs ¹	1.22	1.16	1.18	0.75
Condenser press., in. Hg abs ²	1.097	1.097	1.104	0.648
Turbine steam flow, lb/hr ³	50,300	49,800	50,100	26,700
Bypass steam flow lb/hr	7,700	7,200	7,400	3,300
Turbine heat to cond., Btu/hr ³	45.5 x 10 ⁶	45 x 10 ⁶	45.3 x 10 ⁶	24 x 10 ⁶
Bypass heat to cond., Btu/hr	9.25 x 10 ⁶	8.65 x 10 ⁶	8.9 x 10 ⁶	3.97 x 10 ⁶
Total heat to cond., Btu/hr	54.8 x 10 ⁶	53.7 x 10 ⁶	54.2 x 10 ⁶	28 x 10 ⁶
Heat removed with cond., Btu/hr	2.88 x 10 ⁶	2.83 x 10 ⁶	2.88 x 10 ⁶	1.02 x 10 ⁶
Heat to cooling water, Btu/hr	51.9 x 10 ⁶	50.9 x 10 ⁶	51.3 x 10 ⁶	27 x 10 ⁶
Heat to cooling water, Btu/lb steam	893	893	893	900
Circ. water temp. rise, °F	9.3	9.4	9.1	5.5
Heat added to C.W., Btu/lb	9.3	9.4	9.1	5.5
Calc. C.W. flow, gpm	11,150	10,830	11,170	9,820
Range, (vac. temp - C.W. in) °F	27.2	27.01	27.14	18.53
Terminal difference, °F	17.9	17.61	18.04	13.03
Log mean temp. diff., °F	22.2	22.1	22.5	15.65
Condensing surface, ft ²	5280	5280	5280	5280

¹ Zimmerli gauge

² Local hotwell temperature

³ From Fig. 5-7, this report, based on Zimmerli press.

Heat trans. rate, Btu/hr-ft ² -°F MD	445	438	433	327
Temperature correction, F_t	0.905	0.905	0.905	0.855
Tube mat'l and ga. factor, F_{tm}	0.975	0.975	0.975	0.975
Water velocity, fps	6.83	6.65	6.8	6.0
Base heat transfer, HEI	687	678	686	645
Cleanliness factor, %	73.5	73.3	71.6	60.8

Condenser Performance Before Cleaning

CH-6-16-58

Date	5-27-57	5-30-57	6-3-57	6-4-57
Time	1420	0900	0920	1150
Generator load, kw	4100	4000	4050	2050
Condenser pressure, in. Hg abs ¹	1.64	2.10	1.75	2.13
Turbine steam flow, lb/hr ²	50,000	49,800	49,800	27,600
Bypass steam flow, lb/hr	10,000	10,200	10,200	26,400
Turbine heat to cond., Btu/hr ²	45.5 x 10 ⁶	45.5 x 10 ⁶	45.2 x 10 ⁶	25.6 x 10 ⁶
Bypass heat to cond., Btu/hr	12.03 x 10 ⁶	12.3 x 10 ⁶	12.3 x 10 ⁶	31.7 x 10 ⁶
Total heat to cond., Btu/hr	57.5 x 10 ⁶	57.8 x 10 ⁶	57.5 x 10 ⁶	57.3 x 10 ⁶
Heat removed with cond., Btu/hr ³	3.7 x 10 ⁶	4.2 x 10 ⁶	3.9 x 10 ⁶	3.8 x 10 ⁶
Heat to cooling water, Btu/hr	53.8 x 10 ⁶	53.6 x 10 ⁶	53.6 x 10 ⁶	53.5 x 10 ⁶
Heat to cooling water, Btu/lb steam	898	892	892	990
C.W. temp. rise, °F	10	11	10	11
Heat added to C.W., Btu/lb	9.98	10.97	9.98	10.97
Calc. C.W. flow, gpm	10,800	9,800	10,750	9,780
Assume constant C.W. flow, gpm	11,940	11,940	11,940	11,940
Rise (C.W. out - C.W. in) °F	10	11	10	11

¹ Zimmerli gauge

² From Fig. 5-7, this report

³ Based on hotwell temperature (local)

Range (vac temp. - C.W. in) °F	26.6	28.8	25.7	27.3
Terminal difference, °F	16.6	17.8	15.7	16.3
Log mean temp. diff., °F	21.3	22.8	20.2	21.5
Condensing surface, ft ²	5300	5300	5300	5300
Heat trans. rate, Btu/hr-ft ² - °F MD	478	443	500	470
Temperature correction F _t	0.985	1.02	1.005	1.033
Tube mat'l and gauge cor., F _{tm}	0.975	0.975	0.975	0.975
Water vel. (calc. C.W. flow), f _{ps}	6.6	6.0	6.55	6.0
Base heat transfer (HEI)	675	645	673	645
Cleanliness factor, %	73.8	69	76	74.8

SENSITIVITY OF GROUND DETECTOR

(Test Report No. 12, by S. J. Robb)

This test was to determine the sensitivity limits of the ground detector used in conjunction with the 120 volt DC bus.

Discussion. The sensitivity curves shown in Fig. 12-1 are calculated from the resistance values of the ground detector. The curve equations were derived from the circuit shown on Fig. 12-1.

1. Ground Detector and Indicating Lights

$$R_g = \left(\frac{120 - V_D}{2V_D - 120} \right) 300$$

where

R_g = insulation resistance to ground

V_D = detector coil voltage

The resistance of the detector system is 300 ohms each side to ground and the bus voltage is 120 volts.

2. Ground Detector Only

$$R_g = \left(\frac{120 - V_D}{2V_D - 120} \right) 1000$$

Fig. 12-1

Ground detector sensitivity curves, 120 v dc bus

The resistance of the detector system is 1000 ohms each side to ground.

3. Relay Alarms on Pick-up

As the setting approaches 60 volts, a point is reached where the alarm contact will not reseat except by hand. A setting just above this is the minimum relay setting. There is also a range of coil voltage just above this minimum setting where the alarm contact vibrates and prevents the horn from being silenced on the annunciator. These characteristics are inherent to a plunger type relay.

Results

1. As found, the insulation resistance to ground of the 120 V DC bus had to be less than 500 ohms before the ground detector relay would initiate an alarm.
2. The resistance of the detector and indicating lights was 300 ohms, positive to ground and negative to ground.
3. Minimum setting of the detector raised the alarm level of insulation resistance to 900 ohms.

4. By removing indicating lights, minimum setting was equivalent to an insulation resistance of 3000 ohms.

Conclusions. The highest resistance coil connected to the DC bus is approximately 1000 ohms (BA-3 coil on High Pressure Boric Acid System). A ground on the right side of this coil would apply approximately 40 volts to this coil. If the indicating lights were present, this voltage would be approximately 50 volts. There is a possibility that this could cause high pressure boric acid injection. The MG-6 auxiliary relays on the 440 volt Automatic Transfer System are also 1000 ohms resistance. A single ground in the right place might cause a power interruption.

Recommendations. A voltmeter type ground detector with higher internal resistance should be used for two reasons.

1. The detector itself would not create a hazard when connected to the DC bus.
2. Insulation resistance alarm setting could be raised so that the alarm would be initiated before there is a possibility of relay mal-operation.

WATER LOSSES FROM EBWR

(Test Report No. 14A. By A. Gibson)

Objective. The test was made to determine the quantity and location of water losses in the plant in order to evaluate changes which must be made prior to heavy water operation. Three possible types of leakages were considered:

1. External leakage from valves, magnetrols, equipment, etc.
2. Losses from the air ejector discharge gases in the form of dissociated water (hydrogen and oxygen), and moisture rejected from the system with the gases.
3. In-leakage contamination.

Operation Procedures. Whenever a leak from the reactor water or steam system is spotted:

1. An attempt should be made to stop the leak by tightening the packing.
2. If the leak is large and cannot be stopped by tightening the packing, the valve or leak point should be marked for re-packing or repair when the plant is down, and a calibrated bucket should be placed under it.
3. Small leaks on valve leak-off connections which cannot be stopped will fill balloons placed on the leak-off connections. As a balloon is filled it should be emptied and the amount of leakage noted, and the balloon replaced.
4. At least two times per shift the plant should be inspected for leaks and the above steps be taken when leaks are discovered.
5. An accurate record should be kept of all water removed for sampling purposes.
6. An accurate record should be kept of make-up water added to the systems.
7. An attempt should be made to estimate the amount of water losses incurred during equipment repairs.

Test Methods. The method consisted of placing balloons over the leak-off connections of all valves having such connections, and measuring the leakage rates. Other leaks from points noted throughout the plant were measured. Data for the second and third portions of the experiment, e.g. air ejector losses and in-leakage contamination were obtained by determining the rate of air leakage into the main condenser and vent condenser of the liquid recovery system. This in-leakage was measured by isolating the condensers while they were under vacuum, closing off all equipment auxiliary to them, and noting the rate of pressure rise. The gas in-leakage and water contamination were then computed.

After determining the in-leakage rate, the air ejector discharge gas flowmeter was calibrated and the quantity of escaping gases measured with the reactor operating at 20 Mw. This flow rate was determined to be 2.0 cfm measured at 14.7 psia and 70°F. Since the total in-leakage had previously been determined to be 0.5 cfm, the net flow rate due to radiolytic gases was assumed to be 1.5 cfm. By gas analysis of the air ejector discharge gases, the composition by volume of the gases was determined to be approximately 50 percent hydrogen, 30 percent oxygen, and 20 percent inert gases.

Discussion. As described above, the external valve leakages were measured by means of balloons and buckets wherever this was possible. There are approximately 200 valves in the plant, approximately 45 of which are equipped with leak-off connections, consisting of a lantern ring with leak-off taps located between packing rings. Forty-four of these valves were checked for the leakage rate over a nine day period and the average rates determined. Of the forty-four, twenty-five leaked at measurable rates throughout the test even after tightening the packing glands. Packing in the valves is Teflon impregnated asbestos of the split ring type.

There seemed to be some correlation between the leakage rate, the size of the valve, and the temperature of the fluid. The larger valves and those with higher fluid temperatures usually leaked at a greater rate than the small and low fluid temperature valves. The valves checked range from $\frac{1}{2}$ in. in size to 4 in. The maximum fluid temperature and pressure was 489°F and 600 psig. The average leakage rate for the forty-four valves measured was slightly greater than 0.1 lbs per day per valve. Assuming the same average leakage rate to hold throughout the plant, the total plant leakage was $200 \times 0.1 = 20$ lbs per day.

From data obtained on the gas dissociation tests, the total light water leakage was determined to be approximately 63 lbs per day at 20 Mw of reactor power. Assuming the same dissociation rate with heavy water, this would correspond to approximately 70 lbs per day of D₂O. The total dissociated gas flow rate is a function of reactor power. At 20 Mw, the 63 lbs per day of hydrogen and oxygen indicated above corresponds to an average gas decomposition rate of 55 cc of gas per liter of condensed steam.

The total measured plant losses corresponded to $10\frac{1}{2}$ gal of water per day. The plant water inventory was determined before and after a seven day running period. This inventory check indicated that the total losses were approximately eighteen gal per day. Although the accuracy of the inventory check is not very good, it indicated greater leakage than the measured rate. Since all possible leak points in the plant were not measured, it is believed that the total plant losses were between 10 and 15 gal per day.

Conclusions. The test results were summarized as follows:

1. For this plant, the present average total light water leakage from valves, etc. is approximately 20 lbs per day.
2. Approximately 63 lbs of equivalent water weight per day is discharged from the air ejector to the stack in the form of dissociated hydrogen and oxygen with the plant operating at 20 Mw of power. In addition to the dissociated gas discharged from the ejector, approximately 5 lbs per day of water in the form of contained moisture in the gases is discharged.
3. The light water in-leakage into the system is estimated at 1.2 lbs per day.

Comparative Cost Figures. As a power plant, the EBWR is presently operating with very low water losses. However, for heavy water operation these losses must be reduced to an even lower value, as the below listed comparative D₂O loss cost figures indicate. Cost figures are based upon a 12 percent capitalization factor, 80 percent load factor, and D₂O charge of \$28.00 per lb. The estimated costs are for one year of operation.

1. Plant operating as at present and with light water losses assumed to be D₂O:

$$\text{Est. unaccounted for losses} = \frac{15 \times 365 \times 0.8 \times 28}{0.12} = \$1,020,000$$

$$\text{External equipment leakage cost} = \frac{20 \times 365 \times 0.8 \times 28}{0.12} = 1,360,000$$

$$\text{Gas dissociation loss cost} = \frac{70 \times 365 \times 0.8 \times 28}{0.12} = 4,760,000$$

$$\text{Air ejector moisture loss cost} = \frac{5 \times 365 \times 0.8 \times 28}{0.12} = 340,000$$

$$\text{Total capitalized cost} = \$7,480,000$$

2. Expected plant operation with D₂O with equipment which will be or is presently being installed (see below under "Changes Prior to Heavy Water Operation").

$$\text{External equipment leakage cost} \leq \frac{0.5 \times 365 \times 0.8 \times 28}{0.12} = \$34,100$$

$$\text{Gas dissociation loss cost} \leq \frac{0.4 \times 365 \times 0.8 \times 28}{0.12} = 27,300$$

$$\text{Air ejector moisture loss cost} = \frac{0.001 \times 365 \times 0.8 \times 28}{0.12} \approx 100$$

$$\text{Est. unaccounted for losses} = \frac{0.1 \times 365 \times 0.8 \times 28}{0.12} \approx 6,800$$

$$\text{Total estimated capitalized cost} \leq \$68,300$$

$$\text{Cost in mills/kw-hr} = \frac{68,300 \times 0.12 \times 1000}{5000 \times 8760} = 0.187$$

Changes Prior to Heavy Water Operation. Prior to operation with heavy water the following steps should be taken:

1. As a first step towards reducing the leakage from valves, a test group of valves from which a leakage record is available should be repacked with a different kind of packing. The leakage rates of these valves during operation will be compared with the previous leakage rates.
2. All valves in the plant which do not now have a lantern ring and leak-off connection should be replaced with those that do, or, in some locations valves with sealed bonnets or with a bellows type seal may be used.
3. If the above steps do not reduce the total equipment losses to less than 0.5 lbs per day of water, it may be necessary to pipe the valve leak-off connections to central collection tanks, and thence to the liquid recovery system.
4. Some valves will be eliminated, especially drain valves on heaters, and others where practical.
5. The catalytic recombiner for reclaiming the dissociated deuterium and oxygen should be installed. Tests are presently being conducted on this unit and it is believed that it will operate at better than 99 percent efficiency.
6. The alumina dryer which has been received should be installed to reclaim the moisture in the air ejector gases. It is believed that this unit will reduce these water losses from 5 lbs per day to 0.001 lbs per day.
7. It may be desirable to locate leak detectors at appropriate locations throughout the plant for annunciation in the control room.

EBWR CONDENSER EVACUATION RATE

(Test Report No. 16, by V. C. Hall)

Before any quantity of steam is exhausted into the condenser, it is necessary to remove the air from the condenser. Non-condensable gases are removed by an air ejector (vacuum pump) so that pressure will not build up in the condenser.

The ejectors used on EBWR are a twin two stage type with inter-cooler and aftercooler. The valving is such that all gases (including "Hogging") pass through both stages of the ejector. During initial evacuation at startup both sets of ejectors are in operation. Condensate from the intercooler is returned to the main condenser through an 8 ft water leg. Condensate from the aftercooler drains to the recovery system and is then returned to the main condenser. The gases from the air ejector are discharged out of the building through the stack.

The graph (Fig. 16-1) shows the measured rate of evacuation as well as the design estimated rate of evacuation.

Fig. 16-1
Time required to evacuate
condenser.

STEAM BYPASS VALVE - GENERATOR LOAD

(Test Report No. 17, by V. C. Hall)

The test was performed to measure the flow characteristics of the steam bypass valve. The load on the generator was varied and the resultant bypass steam flow and valve positions noted. The reactor power remained constant throughout the experiment. The condenser vacuum changed, due to the change from steam flow through the turbine to steam flow through the bypass valve. The condenser pressure corresponds to the water vapor saturation pressure at the hotwell temperature.

Data on steam flow through the turbine as a function of admission valve position is shown in Table I and plotted in Fig. 17-1. Steam flow through the bypass valve vs bypass valve position and steam flow through the turbine vs generator output are shown in Figs. 17-2 and 17-3.

It should be noted that the bypass steam flow indication at low flows is not too accurate. At 5,000 lbs per hour the error may be as great as plus or minus 50 percent.

Table I - Power Generation for Various Valve Positions.
Reactor at 20 Mw. Flow 60,000 lbs per hr

Power generated, kw	Admission valve position	Bypass		Reactor pressure ^a	Condenser vacuum	Hotwell temp., °F.
		valve position	Steam flow			
3,000	31	16	22,000	600	1.55	
4,000	38	10	10,000	600	1.50	
4,500	44.5	7	2,000	600	1.37	91
4,500 ^b	44.5	7	2,000	600	1.35	90
4,000	38	10	9,500	600	1.46	92
3,500	34	13	16,000	600	1.50	93
3,000	31	16	22,000	600	1.55	94
2,500	25.5	18.5	27,000	600	1.57	96
2,000	21.0	22.0	32,000	600	1.61	96
1,500	18.0	24.0	37,500	600	1.69	97
1,000	14.5	27.0	43,000	600	1.75	98
500	7.5	30	49,000	600	1.79	100
200 ^c	4.5	32	53,000	600	1.81	100
1,000	13.0	27	43,000	600	1.76	99
2,000	22.0	22	31,000	600	1.67	97
0 ^d	0	34.5	60,000	600	1.91	102

^aAn overall drop of 10 lb was noted on the reactor pressure recorder.

^b63°F inlet condenser water.

^c64°F inlet condenser water.

^dTurbine trip valve closed.

Fig. 17-1. Turbine valve position vs steam flow.

Fig. 17-2. Bypass steam flow vs bypass valve position.

Fig. 17-3. Generator output vs turbine steam flow.

SHIELD COOLING SYSTEM

(Test Report No. 18, by V. C. Hall)

The shield cooling system is designed to cool the shielding surrounding the reactor vessel. The system consists of thirteen copper coil circuits in parallel embedded in the 3 in. thick lead shielding inside the concrete shield. All cooling coils were installed in duplicate so that a spare coil is available for either additional cooling or failure of a coil. No connections of tubing were made inside the shielding. The closed circuit has two pumps and two filters (one set for standby), a heat exchanger, and an expansion tank. The flow through each circuit is 2 gpm.

The original design estimated the cooling requirement for the system at 30 kw, 12 kw from thermal conduction and 18 kw from radiation attenuation.

The shield cooling system is presently removing approximately 15 kw of heat per hr. The amount of heat removed is not a function of reactor power level but rather reactor temperature. The heat removed per circuit on the side coils is in direct ratio to the coil length.

TEST OF HIGH PRESSURE BORIC ACID SYSTEM

(Test Report No. 19, by E. A. Wimunc)

The purpose of this test was to study the dynamic effects on the high pressure boric acid system due to injection because the piping in this system had been rerouted. During initial installation the line was welded into one of the lower 6 in. forced circulation inlet nozzles. The line is now welded into the feedwater line downstream of the feedwater shutoff valve. The 6 forced circulation thimbles are not equipped with thermal sleeves to reduce the thermal stresses in the welds attaching the nozzle to the lower head. The risk of a nozzle failure was high and the probability of a repair was nil (due to radio-activation of the vessel wall and the inaccessibility of repair).

Method. The high pressure boric acid tank was drained and the solution (60 lb C. P. boric acid per 160 gal H₂O) saved. All tests were then conducted by using 160 gal of distilled water (without boric acid) per test at pressures of 500, 1000 and 1600 psig.

Four tests were made having the following conditions:

- Case (1) Air at approximately atmospheric pressure was in the 3 in. pipe line of the HP boric acid system between the valves on the main floor and the water level just above the feedwater stop valve.

- Case (2) The same line was filled with water by building up sufficient pressure in the reactor with the startup heater and by venting this line at its high point on the main floor. A vent was installed in the line specifically for this test.
- Case (3) In addition to Case (2) above, the $2\frac{1}{2}$ in. pipe between the injection valves was filled with water; this condition gave a solid line of water between the HP tank and the reactor.
- Case (4) The piping was modified to maintain an air cushion in the pipe line at atmospheric pressure at all reactor pressure levels. This was accomplished by removing the $2\frac{1}{2}$ in. check just downstream of the injection valves on the main floor and welding the check valve just above the reactor water level. Just upstream of the check valve, a small vent valve was installed and is kept open at all times to bleed any water that might leak past the check valve.

Discussion. A summary of the data taken is shown in Tables I and II plus the 6 attached graphs. (Figs. 19-1 to 19-6).

Since the smoothest and quietest injections occurred when the 3 in. line was filled with air, the most expedient method of maintaining this condition at all pressures was to place the check valve as close to the reactor water level as possible (Case 4). The injection of the HP boric acid is now routed so that the solution enters the feedwater ring. This location will be maintained until the forced circulation piping and pump are installed. At that time the boric acid line should be welded into one of the 6 in. forced circulation inlet nozzles along with a thermal sleeve.

Data was not taken during a trial run of Case 4 at 1600 psig in the boric acid tank inasmuch as each pressure curve differed only with the tail end of the curve which depended on reactor pressure. The injection was very smooth and the check valve sealed almost completely after the test.

The three curves of HP boric acid tank pressure vs time after opening the injection valve are plotted for Case 1 (tests 1, 2 and 3) and are shown in Fig. 19-1.

A transducer was installed just above the reactor water level on the injection line just downstream of the feedwater stop valve. Pressure curves for tests 1, 2, 3, 4, 6, 8 and 9 are enclosed. Tests 1, 2 and 3 indicate the highest pressures were reached after the water left the pipe and the point where the air followed. Tests 4, 6, 8 and 9 indicate instantaneous peak pressures just after the injection valves were opened; the pressure then leveled off until the air entered the line and is shown by the second gradual rise in the pressure.

Table I - H.P. Boric Acid Injection Test

Date	Test no.	H.P.B.A. tank pressure (psig)	Injection time (water only)	T.V. reactor press		Reactor water temp. (°F)	FW flow lb/hr	*Reactor water levels						S.D.E.C. drain	Reactor vent	Remarks
				Before inj. psig	After inj. psig			Before			After					
								T.V.	Rec.	Ind.	T.V.	Rec.	Ind.			
4-25-57	1	501	13 sec.	0			0	11'9"	0	4.68'	12'4"	+7	5.17'	open	open	Air was in 3" line between check valve on main floor to reactor water level. Smooth injection; shock only at elbow below main floor.
4-26-57	2	998	9.2 sec.	0	70	110	0	11'7"	-3"	4.45'	11'11"	+2.5"	4.82'	open	open	Same as above.
4-29-57	3	1600	9.85 sec.	0	180	97	0	11'7"	-2.2"	4.53'	12'2'	+4.3"	4.8'	closed	closed	Same as above.
4-30-57	4	495	14.3 sec.	10	40	268	65,000	11'3-1/2"	-7.8"	4.13'	12'1"	+1.3"	4.8'	closed	closed	Vent was added to high point of 3" line on main floor. Line was full of water between check valve and reactor. Elbow below main floor deflected 3/8" horizontally. Loud shock noise in pipe.
4-30-57	5	498	13 sec.	10	80	205	70,000	11'8"	-5"	4.35'				closed	closed	Water was in line up to 2-1/2" check valve on main floor. Line had been vented on main floor. Elbow below main floor deflected 3/8" horizontally. Loud shock noise in pipe.
5-1-57	6	998	10 sec.	10	**150	260	61,000	11'4"	-6.8"	4.2'	12'3-3/4"	+5-1/2"	5.1'	closed	closed	Same as above except elbow below main floor deflected 7/8" horizontally. Very loud shock force in pipe.
5-1-57	7	497	13.5 sec.				0	11'6"	-5.6"	4.3'				closed		Same as test no. 5.
5-2-57	8	1000														Repeat test of no. 6 (Demonstration Test).
5-2-57	9	532	13.2 sec.	45		200	0	11'8"	-5.3"	4.3'	12'1"	+2.1"	4.85'	closed		Boric acid line was full of water between H.P.B.A. tank and reactor - this included line between injection valves. Elbow below main floor deflected 1/2". This test gave the loudest shock report.
5-5-57	10	1600	8 sec.	45	195		60,000							closed	closed	The check valve had been relocated from the main floor to a point just above the reactor feed water line. This was done to maintain an air pocket (vented to atmosphere) in 3" line. Smooth injection. Elbow below main floor deflected 1/2" horizontally. Check valve loop deflected 1/16".

*Each inch of reactor water level indicates approximately 25 gal.

**Pressure on recorder indicates 118 psig.

A pressure drop through the high pressure boric acid system was calculated to be 705 psi; this was based on the assumption of a flow rate of 960 gpm, a water temperature of 60°F, and a pressure differential of 1000 psi (1600 psig in the HP boric acid tank and 600 psig in the reactor).

Table II - Pressure vs Time after Opening Injection Valves

Test Time (sec)	HP Boric acid tank pressure							
	1	2	3	4	5	6	7	9
0	501	998	1600	495	498	998	497	532
2	420	800	1220	430	440	800	440	460
4	370	670	1020	390	390	700	390	415
6	350	620	900	350	350	630	350	375
8	320	550	790	320	330	570	320	340
10	290	510	640	300	300	520	295	315
12	260	470	540	270	280	470	275	295
14		400	450	255	270	420	255	285
16		350	400	245	250	350	235	250
18	230	290	320	200	210	300	200	200
20	150	240	280	160	170	270	165	170
22	120	200	240	140	140	240	130	140
24	100	165	220	120	120	210	110	115
26	85	140	200	100	100	190	92	100
28	60	120	200	90	90	175	80	90
30	55	105	180	80	82	160	72	83
32	53	95	180	76	78	145	68	79
34	50	90	180	75	75	135	65	76
36	50	85	180	74	75	132	64	75
38		83	180	74	74	130	64	75
40		82	180	74	74	130	63	75
		+		+	+	+	+	+

Fig. 19-1. HP boric acid pressure.

Fig. 19-2. Boric acid system injection tests. Line filled with air.

Fig. 19-3. 500 psig boric acid injection test.

Fig. 19-4. 1000 psig boric acid injection test.

Fig. 19-5. 1000 psig boric acid injection test.

Fig. 19-6. 500 psig boric acid injection test.

Conclusion: Case 4 was the most expedient method for maintaining an air cushion between the injection valve and the reactor water for smooth injection. The air volume amounts to 3.9 ft³. This will be the condition maintained until the forced circulation system is installed. At this time the injection line will probably have to be welded into one of the 6 in. forced circulation inlet lines after a thermal sleeve has been installed in this nozzle.

With the system in good mechanical and dynamic condition, a test involving the injecting of boric acid into the reactor should be made soon so that its effect on the reactor can be studied. However, a detailed study is deemed necessary prior to this test as to the effects it will have on components besides the reactor. Namely, the pressure in the condenser will rise from 1.5 in. Hg abs. to about 11 psig when a 1600 psig boric acid is injected into a 600 psig reactor pressure. As a secondary effect the vent condenser in the liquid recovery system will be pressurized and will probably rupture.

DETECTION OF CIRCULATING WATER INLEAKAGE TO THE CONDENSER

(Test Report No. 21, by H. Y. Cadoff)

The condenser of EBWR is equipped with conductivity cells on the hotwell exits designed to detect leaks of secondary cooling tower water into the primary loop. High sensitivity of this system will be important when the reactor is changed to boiling D₂O operation. Contamination of D₂O with light water, even in small percentages, markedly changes the moderator's characteristics. The proposed test was designed to find what rate of leakage into the hotwell is detectable with the present system.

Test Procedures. The test was run with the reactor shut down and under the following conditions:

1. The reactor below 200°F and filled with water.
2. The condenser vacuum breaker open.
3. The sample inserted into the feedwater filter screen assembly on the suction side of the feedwater pump. (Note: since the feedwater line was at a positive head, the cooling water had to be fed in from a higher point than the level of the hotwell.)
4. A feedwater pump started and run a few minutes to bring the system to equilibrium.
5. The level of the hotwell checked and recorded.
6. 10 cc per min of cooling tower water introduced in the sample tap. (First filling the delivery line.)
7. The hotwell conductivities followed on the plant recorder (control room) and each point recorded individually as it was plotted.
8. A note made of the time of alarm, total time of experiment and amount of cooling water added until the alarm was given.
9. Since in this case the north half of the waterbox is the only circulating section of the condenser and the flow rate was known, the concentration of cooling water versus conductivity recorded could be plotted. The required concentration for alarm could also be found.
10. The condenser was drained after the test for a scheduled inspection.

Discussion. Cooling tower water for use in the test had a conductivity of 2.56×10^3 mhos (390 ohms), and a pH of 5.9. A rubber tube was connected to the petcock on the screen at the inlet to the feedwater pump. The tube end was brought above the level of the condenser and filled with cooling tower water. A 50 milliliter burette was attached to the top of the tube to meter flow. The system was isolated by closing the inlet valves to the full flow feedwater filters. The feedwater pump was started and

the water in the condenser was allowed to recirculate until equilibrium conductivity values were reached. The equilibrium conductivities were considerably higher than the usual operating conductivity of 0.2 micromhos (see Table I). Additions of cooling tower water and conductivity readings on both north and south hotwell conductivity cells were synchronized on a time scale to allow correlation of cause and effect.

Table I

Decimal time, minutes	Ml cooling water added	Percent cooling water addition	North half conductivity, micromhos	Percent total change north	South half conductivity, micromhos	Percent total change south
0	0	0	1.48	0	1.54	0
0.5	11	12.1	1.48	0	1.54	0
1.0	13	14.3	1.49	1.96	1.53	0
2.0	17	18.6	1.62	27.5	1.67	25.5
2.33	-	-	1.60	23.5	1.62	15.7
2.67	-	-	1.60	23.5	1.67	25.5
3.0	23.7	26.0	1.62	27.5	1.61	13.7
4.0	30.7	33.6	1.65	33.3	1.67	25.5
5.0	37.4	41.1	1.69	41.2	1.73	37.3
6.0	43.6	47.7	1.73	49.0	1.78	47.0
7.0	51.7	56.7	1.76	54.9	1.85	60.8
8.0	57.6	63.1	1.80	62.7	1.88	66.7
9.0	61.9	67.8	1.83	68.6	1.92	74.5
10.0	65.2	71.5	1.86	74.5	1.89	68.6
11.0	70.0	76.7	1.86	74.5	1.90	70.7
12.0	74.9	82.2	1.89	80.4	1.96	82.3
13.0	78.8	86.5	1.90	82.3	1.96	82.3
14.0	85.5	93.7	1.93	88.2	1.99	86.2
14.75	91.2	100	-	-	-	-
15.0	-	100	1.94	90.2	2.02	94.0
16.0	-	100	1.99	100	2.02	94.0
16.33	-	100	1.99	100	2.05	100

It was believed that only the north waterbox contents would be recirculating. The data recorded showed both north and south waterbox conductivity cells sensed a change in conductivity. Visual inspection of the recirculating system in operation at that time (through the upper man-hole on the condenser) confirmed the postulation that only the north waterbox contents were recirculating and that the south waterbox should be essentially stagnant. The unexpected increase in readings (and finally high conductivity alarm) of the south waterbox conductivity cell was attributed to the close physical proximity of both cells to each other and to a tee joint, in which flow from both halves of the condenser hotwell mixes before entering the feedwater pump. Water from the north side of the tee is forced to make a 90° turn to enter the pump suction line. A swirl formed at this bend could easily come in contact with the south conductivity cell and cause erroneous readings. This result presents the possibility that the hotwell conductivity cell readings may also be somewhat difficult to interpret during normal operation.

An additional test was performed with the south conductivity cell disconnected at the cell. When the south cell was disconnected the reading of the north cell dipped about 0.02 micromhos. Upon reconnection of the south cell the north cell responded with a similar rise in conductivity reading. This dependence of one cell reading upon the other was verified when the reactor was running. The full flow conductivity cells require drainage of the system prior to removal for servicing.

Fig. 21-1. Conductivity change vs time.

The data was compiled and correlated by plotting percentage of total conductivity change versus time (Fig. 21-1), percentage of total cooling water addition versus time (Fig. 21-1), and percentage of cooling water addition versus percentage of conductivity change (Fig. 21-2). The relationship between conductivity and cooling water addition in the range investigated (~1.5 to 2.0 micromhos) proves to be linear. The results were linearly extrapolated to the normal operating range and a minimum leakage rate to cause an alarm calculated. This minimum rate is 98.2 milliliters per min or about 13 lbs per hr. Anything less than 13 lbs per hr of leakage would not cause a hotwell conductivity alarm at the present alarm setting. Since leakage is continually flushed out of the

Fig. 21-2. Cooling water addition vs conductivity change.

0.5 micromhos, the lowest feasible alarm setting with this system is 0.6 micromhos. This would reduce the rate of inleakage detectable by a hotwell alarm to 21.8 milliliters per min or about 2.9 lbs per hr. Theoretically if the alarm system was modified the experiment showed it possible to detect a leak equivalent to 2.75 milliliters per min at 60.2 gpm per hotwell half. This would be about 0.39 lbs per hr inleakage. This would require a setting of the alarm at 0.055 micromhos above the normal operating level of the hotwell (0.2 micromhos). The normal variation of the instrument would not allow such a close setting. It appears, however, that a 0.3 micromho setting of the alarm (about 0.72 lbs per hr) would probably be safe in a modified system.

condenser by low conductivity condensate, a smaller than alarm-causing leak would tend to build up reactor vessel conductivity to an alarm (Note: the reactor cleanup system is very slow). Unfortunately this reactor vessel alarm would probably require enough inleakage to seriously damage a D_2O moderated system. The high concentration of chlorides in the cooling tower water would also make this inleakage dangerous in a light water system.

The present multipoint recorder can be set to alarm at only one conductivity for all points. Since the reactor water conductivity (the highest of the group) is normally

Conclusions.

1. The present setting of the high conductivity alarm on the hotwells is too high. Both heavy and light water systems could be endangered by excessive inleakage allowed by this alarm setting.
2. The causes for the present interdependence of conductivity cell readings should be investigated and the conductivity cells repaired if necessary.
3. The system should be modified in the following manner:
 - a) The conductivity cells should be changed to a type which allows inspection and service of the cells to be accomplished without plant shutdown and system drainage.
 - b) Thought should be given to relocation of the conductivity cells to more advantageous positions.
 - c) The recorder-alarm system should be modified to allow an alarm setting at 0.1 micromho or less above normal hotwell operating conductivities.

STEAM POWER RELIEF VALVE (FOSTER RELIEF VALVE)

(Test Report No. 27, by V. C. Hall)

The objective of the test was to obtain performance data on the Foster relief valve. The subject valve was disassembled by outside vendor personnel to check dimensions, port openings, and general valve conditions. At this time a medium range spring was installed in the valve pilot assembly.

Fig. 27-1. Flux chart for Foster valve at 5 Mw.

The valve was subsequently tested. It was found that the maximum set pressure was less than the 600 lb reactor operating pressure. The valve showed erratic operation at 5 Mw as shown by the attached flux chart (Fig. 27-1). The cycling of the valve also caused a noticeable change in noise at the valve as the steam flow changed. The valve opening for a steam flow of 17,000 lb per hr was 0.056 in. of a total valve travel of 0.875 in.

The test was discontinued because of the valve cycling. The reactor was shut down, allowed to cool and a heavy ($\frac{1}{2}$ in. wire) spring installed in the valve pilot. The valve was then adjusted to open at a pressure of approximately 510 lb with the reactor at 5 Mw power. It was stable at flows up to 14,000 lb per hr. The reactor power was increased to 20 Mw for the valve test. The results of the test are shown in Table I.

Figure 27-2 is a plot of steam flow vs system pressure. The proportional band as shown is approximately 15 lbs for a steam flow change of 60,000 lb per hr. The valve was stable throughout the test. However, the noise level at the valve was very high making accurate measurements difficult to obtain. At the high flow rate, insulation was vibrating off the associated piping.

Fig. 27-3 is a plot of steam flow vs valve seat opening. The valve was open approximately $\frac{1}{4}$ in. for a flow of 60,000 lb per hr. The total valve travel is .875 in. indicating the valve has ample capacity.

Table I - Flow Test Steam Power Bypass Valve

Pressure TV gauge	Flow, thousand lb per hr				Lift		Temp., °F	Remarks
	Feedwater	Main steam	Corrected steam	Steam through Foster	Diaphragm	Seat		
500	60	68.5	60	0	0.003	-	301	
520	59.5	60	52.6	8.4	0.009	0.050	311	
530	59.5	51	44.7	16.3	0.0225	0.075	329	
525	60	40	35.1	25.9	0.024	0.110	335	
525	60	30	26.2	34.8	0.025	0.150	335	
-	60	20	17.5	43.5	0.0255	0.195	335	Reactor water level decreasing.
530	60	9	7.9	53.1	0.0255	0.265	335	
525	62.5	20	17.5	43.5	-	0.200	335	
522	62.5	40	35.1	25.9	-	0.115	335	Reactor water level rising.
520	62.5	60	52.6	8.4	-	0.065	335	
500	-	70	-	0	-	0.015	335	Average feed- water flow 61,000 lb/hr.

At the conclusion of the test the valve was set to open at 655 psig. The shutdown switch contacts are actuated at a flow rate of 20,000 lb per hr and reset when the flow is decreased to 10,000 lb per hr. The temperature downstream from the valve is approximately 280°F, indicating that the valve is leaking. The valve was found to be satisfactory as to capacity and performance.

Fig. 27-2. Relief valve vs system pressure.

Fig. 27-3. Steam flow vs valve opening.

REACTOR VESSEL OPERATION DATA

(Test Report No. 28, by G. E. Robinson)

Scope. This report presents an accumulation of data on the pressure vessel for the first six months of operation. It covers a period from the original startup to the first major shutdown. Included is information on the gasket closure, vessel temperatures during startup and shutdown, and thimble cooling.

Gasket Closure. A double gasket with an annulus provided with monitoring taps between the two gaskets is the type of seal used for the pressure vessel. The originally proposed gaskets, stainless steel Flexitallic with asbestos filler, resulted in a leakage rate of about 350 cc per hr at 1200 psig at the vendor's shop when a cold hydrostatic test was put on the vessel (Table I, Test 1). This gasket also showed signs of air leakage at 15 psig. Two types of Akron metallic gaskets were also tried (at Argonne): (1) a stainless steel corrugated jacket with asbestos core, (2) a stainless steel smooth jacket with asbestos core (Table I, Tests 2 and 3).

The last gasket tested and the type that was used for startup and the first six months of operation was a Flexitallic gasket with a zinc-plated, carbon steel core with Teflon filler. The radial clearance between the gasket and groove was filled with stainless steel type 430 strip $\frac{1}{32}$ in. thick by $\frac{3}{16}$ in. wide. The strip was placed so that a continuous $\frac{1}{32}$ in. shim was effected around both the inside and outside circumference of each gasket (Table I, Tests 4 and 5).

The gasket that was used for the first six months of operation leaked under hydrostatic test at 1040 psig (Table I, Test 5). At 1200 psig the leakage was approximately 184 cc/hr. After approximately 14 thermal cycles on the vessel and about 400 hrs of steam production at various loads, a cold hydrostatic test was put on the vessel and related piping. At 600 psig a leak-rate of 50 cc per min was noted. At 1200 psig a rate of 1700 cc per min resulted. However, during the entire six months of operation, there was no sign of leakage while the reactor was operating. This means that with 600 psig of saturated steam there was no detectable leakage past the gasket.

Prior to removing the reactor cover for fuel element transfer, another cold hydrostatic test was put on the vessel. Approximately 35 thermal cycles had been put on the vessel (Fig. 28-1). At 210 psig the leakage rate was 150-175 cc per hr. A higher pressure was impossible because of external steam system valve leaks.

Table I

LEAKAGE AND GENERAL DATA

Test no.	Date and location of test	Type of gasket	Amount of leakage
1	May 8, 1956 Barberton, Ohio (B & W) Vessel in horizontal position	Flexitallic Gasket Co. Style Spec R (spiral wound gasket) 304 S.S. asbestos filler $\frac{5}{8}$ in. wide, .175 thick	5.7 cc water per min at 1200 psig
2	October 16, 1956 Job site	Akron Metallic Gasket Co. Type DC (double corrugated gasket) 304 S.S., asbestos filler - $\frac{5}{8}$ in. wide, $\frac{1}{4}$ in. thick.	500 cc water per min at 1200 psig
3	October 19, 1956 Job site	Akron Metallic Gasket Co. Type DS (double jacket gasket) 304 S.S., asbestos filler - $\frac{5}{8}$ in. wide, $\frac{1}{4}$ in. thick	1143 cc water per min at 900 psig
4	November 20, 1956 Job site	Flexitallic Gasket Co. Style Spec R (spiral wound gasket) plated CRS, Teflon filler - $\frac{5}{8}$ in. wide, .175 in. thick - with $\frac{1}{8}$ in. high x .030 in. thick - 304 S.S. 48 in. long restraint strips	$\frac{1}{4}$ cc water per min at 900 psig. (Gasket buckling at restraint strip laps where laps were poor)
5	December 16, 1956 to June 26, 1957 Job site	Same as Test no. 4 except filler strips continuous and made from S.S. type 430	Zero leakage to 1050 psig. (3.2 cc water per min at 1200 psig)

MEASUREMENTS

Test no.	Studs and nuts		Flange separation		
	Stud elongation	Turns on nuts	After nut tightening	At pressure	After pressure
1	.009 in.	-	-	-	.007 in.
2	.010 in.	200°	-	.015 in. (1200 psi)	.013 in.
3	.010 in.	160°	.028 in.	-	-
4	.012 in.	130°+	.020 in.	.020 in. (600 and 900 psi)	.020 in.
5	.011 in.	120°+	.022 in.	-	.018 in.

Table I (cont'd.)

MEASUREMENTS

Test no.	Gaskets			
	As manufactured		After test	
	Thickness	Width	Thickness	Width
1	.183 in.	.620 in.	.140 in.	.692 in.
2	.265 in.	.615 in.	.155 in.	.689 in.
3	.260 in.	.615 in.	.147 in.	.689 in.
4	.177 in.	.620 in.	.150 in.	-
5	.177 in.	.620 in.	.144 in.	-

<u>Test no.</u>	<u>Comments</u>
1	<p>Before test - Since the vessel was in a horizontal position, the gaskets were stuck into the gasket grooves with molasses prior to installing the cover.</p> <p>After test - The gaskets had buckled locally, generally over each gasket. The gaskets filled the grooves tightly. The cover was scored by the gaskets and these marks indicated relative motion between the gaskets and the cover of $\frac{1}{32}$ in.</p>
2	<p>After test - The gaskets filled the grooves tightly. The cover was scored by the gaskets and these marks indicated some relative motion between the gaskets and cover. Inspection under a microscope showed the metal had flaked at the profile bends of the gaskets.</p>
3	<p>During test - Intermittent groaning sounds were heard as pressure was raised, probably due to relative motion between the gasket and cover.</p> <p>After test - The gaskets filled the grooves tightly. Inspection under a microscope showed the metal was flaked at the profile bends of the gaskets.</p>
4	<p>After test - The gaskets and restraint strips filled the grooves tightly. No localized buckling had occurred except at those locations where the restraint strips had not completely butted together. Film of the Teflon filler was on the cover (easily rubbed off), but score marks were relatively minor compared to Test no. 1.</p>

Table I (cont'd.)

Test no.	Comments
5	Before test - Differs from Test no. 4 only in that restraint strips were single lengths rather than short (4 ft) sections. After test - Same as Test no. 4 except inner gasket appeared to be waterlogged and extrusion of Teflon was pronounced. Teflon burned into cover plate facing.

Note: This data taken from letter to EBWR Chronological File from N. Balai and T. L. Kettles dated January 8, 1956.

Fig. 28-1. Pressure cycles on vessel.

A comparison of the initial stud elongation and flange separation measurements to those obtained after approximately 35 thermal cycles showed that the closure bolting stresses had relaxed to 20 percent of the original bolt stress, from approximately 25,000 psi to 5,000 psi, as confirmed by the hydrostatic tests having an initial leakage point of 1040-1050 psig and final value os 200 psig. An attempt was made to re-stress the bolt studs to the initial stress value of 25,000 psi. About

50 percent of the nuts permitted an additional 30° rotation which would theoretically produce a .010 in. bolt elongation or 25,000 psi. Limitations of compressed air to the impact wrench stopped further attempts to re-stress the studs.

From the appearance of the closure gaskets after six months of service, it was apparent that the inner gasket carried the major portion of the bolt load. The outer gasket showed very little evidence of Teflon extrusion while Teflon extrusion on the inner gasket was extensive.

Galling of Studs, Washers and Nuts. It was observed after the first hydrostatic pressure test on the vessel that the closure studs, washers and nuts had galled to an unacceptable degree. Therefore during the subsequent three hydrostatic and gasket tests on the vessel various types of lubricant, washer materials and washer finishers were used in an attempt to eliminate this galling. The conclusions made after these tests are as follows:

1. A mixture of graphite and oil used as a lubricant eliminated galling of the studs and nuts. It also virtually eliminated galling of the washers regardless of the washer finish or material.
2. Of the two materials used, namely SA-193 Gr. B-14 and SA-306, the SA-306 material appeared to have a lesser tendency to gall although this is not conclusive because no direct comparison has yet been made of the two with exactly the same finish condition.
3. Of the various finishes used, namely lap, machine, rusty machine and acid pitted the lap finish appears to have the maximum galling tendency. Table II summarizes the observations made during the four tests.

During the removal of the cover plate from the vessel after six months of operation, two studs were removed from the ring forging with the nuts frozen to the studs. Neither the nuts or studs were recoverable. Graphite and oil had been used as a lubricant.

Table II

Test no.	Description of studs	Description of nuts	Description of washers	Lubricant used
1	SA-193, Gr. B-14, CL III fit with SA-105, Gr. II ring forging.	SA-194, Gr. 3, CL III fit with studs.	SA-193, Gr. B-14 No. 125 microinch machine finish.	Molykote and alcohol.
2	Same studs as for Test no. 1 after re-working galled threads.	Same nuts as for Test no. 1 after re-tapping galled threads & after re-machining bearing surfaces.	Same washers as for Test no. 1 after remachining, chrome plating & lap finishing.	Molykote and alcohol.
3	Same studs as for Test no. 2 except those that had galled were replaced with spare studs. The galled studs were re-worked for future use.	Same nuts as for Test no. 2 except those that had frozen on studs were replaced with spare nuts. The frozen nuts after removal from the studs were reworked for future use.	SA-306 chrome plated (42)* lap finished Same as above except pitted (2)* by acid as test.	Graphite and oil.
4	Same studs as for Test no. 3.	Same nuts as for Test no. 3.	SA-306 chrome plated (34)* lap finish. SA-193, Gr. B-14 machine finish (10)* rusty.	Graphite and oil. (27)* Molykote and oil. (3) Graphite and alcohol. (2) Molykote and alcohol. (2) Graphite and oil. (3) Molykote and oil. (3) Graphite and alcohol. (2) Molykote and alcohol. (2)

*Number of studs so treated.

Table II (cont'd.)

Test no.	Stud tendency to gall and comments	Nut tendency to gall and comments	Washer tendency to gall and comments
1	Minor to bad galling on all studs. One stud could not be removed from ring forging and was machined out and replaced.	Minor to bad galling on bearing surface of all nuts. Half of nuts froze on studs and had to be removed by various machine shop practices.	Minor to bad galling on all washers.
2	Minor to bad galling on all studs. Two studs could be removed only part way out of ring forging and were left permanently in these positions.	Minor to bad galling on bearing surface of majority of nuts. Half of nuts froze on studs and had to be removed by various machine shop practices. Five of these nuts had to be cut off.	Bad galling on all washers.
3	No galling. (No attempt made to remove studs out of ring forging.)	No galling. All nuts could be removed from studs.	Some minor galling on a few of the 42 lap finished washers. No galling on the 2 acid pitted washers.
4	No galling. (No attempt made to remove studs out of ring forging.)	None to minor galling. All nuts could be removed from studs.	None to minor galling. (27) No galling. (3) Minor galling. (2) Minor galling. (2) No galling. (3) Minor to bad galling. (2) Minor galling. (2) No galling. (2)

This data taken from letter to EBWR Chronological File from N. Balai and T. L. Kettles dated Dec. 5, 1956.

Vessel Temperatures. A change in the operating temperature of the reactor induces thermal gradients through the vessel walls and particularly through the head and the ring forging bolting flange, which result in thermal stress in addition to the normal steady state stress. Therefore, the manufacturer set the maximum permissible rate of temperature increase of the vessel at 150°F per hr and a maximum permissible cooling rate under normal conditions (600 psi and 488°F) at 185°F. To test these rates thermocouples were installed in the vessel as shown in Fig. 28-2.

Fig. 28-2. Thermocouple layout.

1. Startup:

Normal startup is accomplished in two steps. The startup heater is used to preheat the reactor water to a temperature of 325°F prior to the withdrawal of rods. From 325°F to the normal operating temperature of 488°F, nuclear heat is used.

A startup heater test (Test Report no. 1) was performed with maximum laboratory steam flow through the startup heater. Table III and Fig. 28-3 show the reactor vessel temperature profile for this test. A maximum of 126°F at a given instant was observed between thermocouples 3 and 4 (time 1355). The maximum rate of rise of temperature was observed on thermocouple 5 with a change of approximately 74°F in a ten minute period; however, the total rise for the hour was 144°F. The manufacturer has confirmed that these temperature rises are not excessive.

Table III - Startup Heater Test. Full Flow

Date	Time	Water temp.	Thermocouple no. (°F)													Press. psig.	Remarks
			1	2	3	4	5	6	7	8	9	10	11	12	13		
3/24	0955	-	109	114	123	196	200	200	200	199	200	199	199	198	200	0	Started shutdown cooling
	1100	-	140	143	148	147	150	162	198	198	199	199	199	198	198	0	
	1245	99	97	97	98	97	98	108	195	197	197	197	197	197	199	0	Stopped shutdown cooling
	1305	-	95	95	95	95	113	162	189	194	197	197	197	197	199	0	Startup heater on
	1315	136	96	95	95	120	187	200	199	199	199	199	198	197	199	0	
	1330	-	96	95	95	166	204	212	214	214	214	212	212	207	200	0	
	1345	142	97	96	95	204	218	223	223	224	224	224	223	219	208	0	
	1355	151	99	98	98	224	232	236	236	236	238	238	236	232	219	0	
	1415	-	113	122	168	246	250	252	254	255	255	255	255	249	233	20	
	1430	242	148	179	218	264	268	271	271	273	274	274	273	267	249	39	
	1445	-	180	210	240	280	286	290	291	293	293	293	293	286	264	59	
	1500	265	194	236	258	296	302	308	309	310	314	314	313	304	279	73	
	1515	292	199	257	273	309	316	320	321	323	325	324	324	317	293	96	
	1525	319	213	279	291	319	325	331	332	334	337	337	334	331	308	100	Test completed.

Fig. 28-3. Reactor vessel temperature profile.

In a normal startup, the head and top of the vessel are apt to be quite warm prior to startup because of heat stored from the previous run (see Table IV and Fig. 28-4). The bottom of the vessel lags in temperature probably because of poor circulation caused by the control rod guide plate blocking off the circulation. This is also probably the cause of the extreme temperature differentials caused by thimble cooling noted in a later paragraph.

Once the reactor has been brought up to 325°F, nuclear heat is added and the reactor is brought up to pressure. Fig. 28-4 shows the typical temperature profile for a startup.

2. Thimble cooling:

The purpose of thimble cooling is to maintain the control rod seals at a temperature of less than 150°F. As previously mentioned, large temperature differentials, probably caused by the control rod guide plate blocking circulation, are obtained when uncontrolled thimble cooling is in use (Table V; Fig. 28-5).

Normal operational practice has been to put a total flow of less than 1 gpm through the thimbles which, according to the temperature of the seals, have need for cooling. This is done after the reactor is up to operating pressure and operating at 5 Mw or above.

Table IV - Example of Startup

Date	Time	Press. psig	Temp.* °F	Thermocouple no. (°F)													Remarks
				1	2	3	4	5	6	7	8	9	10	11	12	13	
2/9	0045	0	130	123	123	123	123	136	182	178	201	222	307	259	275	239	Startup heater on
	0100	0	130	123	123	123	142	187	198	200	201	222	306	259	275	284	
	0130	0	150	135	137	136	198	206	210	210	212	222	305	256	268	278	
	0200	0	230	138	187	198	220	223	224	224	225	225	239	255	268	276	
	0230	25	250	184	233	233	239	240	242	242	242	244	244	255	268	275	
	0300	50	270	223	248	248	255	258	260	262	264	264	264	264	268	273	
	0330	65	290	236	258	263	270	273	276	276	277	278	278	278	275	275	
	0400	100	300	250	270	275	306	310	318	318	319	320	320	320	311	302	
	0430	125	325	260	304	307	318	329	336	338	338	344	344	344	333	321	Startup heater off. Started to pull rods.
	0500	200	330	274	318	318	330	336	348	348	348	349	349	349	349	336	
	0530	350	400	322	374	372	384	392	396	396	396	394	394	394	390	364	
	0600	440	440	368	428	424	428	432	433	438	438	440	440	440	430	396	
	0630	570	470	430	458	450	452	452	466	466	466	468	468	468	460	430	
	0700	600	485	458	478	468	470	480	480	480	480	480	480	480	480	452	
	0730	600	485	478	482	478	470	480	480	480	480	482	482	482	482	460	

*Reactor water temperature.

Table V - Example of Thimble Cooling

Date	Time	Press. psig.	Temp.* °F	Thermocouple no. (°F)													Remarks
				1	2	3	4	5	6	7	8	9	10	11	12	13	
1/7	1900	350	430	459	458	447	440	446	441	440	440	470	471	474	484	469	Reactor scrammed then started up again.
	2000	580	488	475	480	472	480	479	479	479	479	480	480	480	480	472	Thimble cooling started at 2030
	2100	600	488	445	486	473	480	486	485	481	481	485	485	485	486	472	
	2200	600	488	425	486	473	480	486	485	485	485	485	485	486	486	474	
	2300	400	480	424	486	473	480	486	485	485	485	485	485	486	486	474	
	2400	-	400	245	435	438	438	442	442	441	442	462	465	470	476	462	Reactor shut down. Thimble cooling off at 0005
1/8	0100	600	488	301	457	450	460	467	467	467	467	470	471	471	471	460	
	0200	590	488	470	486	475	480	485	483	480	480	483	485	485	485	472	

*Reactor water temperature.

Fig. 28-4. Reactor startup temperature profile.

Fig. 28-5. Thimble cooling.

A maximum temperature differential of approximately 35°F is allowed between vessel thermocouples 1 and 2. It has been found that under these conditions the control rod seal temperatures can be maintained below 150°F.

3. Shutdown:

Normal shutdown presents little problem as the reactor is "bottled up" and allowed to cool down over an extended period of time by slowly cooling the water with the water purification and normal shutdown cooling system.

An example of a relatively fast cooling down period is shown in Fig. 28-6 and Table VI. A fairly large temperature differential exists between thermocouples 5 and 6. This is the spot where the feedwater ring is located and where the water cooled from the water purification and normal shutdown cooling system is returned to the vessel. In this example, one shutdown cooling unit was in service for the first four hours of shutdown. Note the slow cooling rate of the ring forging in Fig. 28-6. Since the ring forging and head are the primary reason for the 185°F per hr rate of cooling and since these components are exceedingly slow in cooling under any conditions, it can be seen that shutdown from a vessel temperature standpoint does not pose any problems.

Fig. 28-6. Shutdown temperature profile.

Cooling the reactor down with the emergency cooler is reported previously in Test Report no. 4B. A vessel temperature profile is recorded in the report and shows that at no time did the cooling rate exceed 185°F per hr.

Table VI - Example of Shutdown

Date	Time	Press. psig	Temp.* °F	Thermocouple no. (°F)													Remarks
				1	2	3	4	5	6	7	8	9	10	11	12	13	
1/25	1700	440	450	395	437	431	446	449	452	450	452	450	452	455	445	411	Shutdown cooling on Ion Exchange System no. 2.
	1710	440	450	434	450	446	444	447	447	447	446	453	453	453	452	431	Blew down reactor from 400 psig to 175 psig.
	1818	160	300	332	330	330	323	330	347	358	367	420	431	434	435	416	
	1900	150	250	277	276	277	270	278	300	351	362	407	422	428	428	411	
	2000	125	195	205	205	207	204	210	277	338	351	389	408	417	417	402	Ion Exchange System no. 2 off.
	2100	110	165	173	173	173	170	175	275	326	348	378	398	408	408	397	
	2200	95	150	162	160	160	160	166	262	318	332	367	389	399	399	390	
	2300	80	145	160	160	159	160	171	257	304	323	357	380	392	392	384	
	2400	8	-	160	160	158	162	173	255	300	318	350	371	384	387	380	
1/26	0100	7	-	150	149	149	148	156	220	290	300	345	360	372	375	370	
	0200	6	-	149	149	147	149	157	220	257	265	333	357	367	369	367	
	0300	6	-	149	147	153	150	151	220	283	285	318	345	363	365	363	
	0400	6	-	147	153	153	155	163	218	278	283	310	340	352	355	355	

*Reactor water temperature.

TEST HEAT EXCHANGER

(Test Report No. 32, by A. Gibson)

A test heat exchanger was placed in service on 8-14-57 with 18 gpm flow of cooling water and sufficient steam at atmospheric pressure for a 5°F water temperature rise. This was to simulate proposed condenser operating conditions. The heads were pulled on 9-25-57 and observed by outside vendor personnel, after which the unit was placed back in service. On 10-10-57 the heads were again pulled, tubes were observed, and those indicated in Fig. 32-1 were cleaned. The unit was again placed in service.

Observation of tubes indicated considerable aluminum oxide residue and some corrosion pitting on tube no. 7. Tube no. 14 also had oxide residue to a smaller degree than no. 7. The least amount of corrosion and oxide was observed in tubes numbered 15 which were clean and smooth to the touch.

Legend:

- 7 - 3/4" x 18 BWG Alum. Alloy 3003 - H14 Alclar Inside Tubes
- 14 - 3/4" x 18 BWG Alum. Alloy 3003 - H14 Plain Tubes
- 15 - 3/4" x 18 BWG Alum. Alloy 6061 - T4 Plain Tubes

Shaded tubes indicate those tubes which were cleaned on 10-10-57.

Fig. 32-1. Heat exchanger

TURBINE INSPECTION

(Test Reports Nos. 36, 53, 53A, by H. L. Gann,
R. E. Bailey, C. A. Hatstat)

The inspection was made during the month of February, 1958. The unit had been in service a total of 2839 hours for a total power output of 11,665,000 kw hr. The steam flow through the turbine during this period was 155×10^6 lbs.

The turbine, manufactured by the Allis-Chalmers Company, is rated 5000 kw at 3600 rpm. It has one impulse and 11 reaction stages. The throttle pressure is 560 psig, and the temperature 482°F. It exhausts at 2.5 in. Hg abs.

Procedures. Only a minimum crew was present during the initial operation. They wore protective coveralls, gloves, shoe covers, and face masks until a radiation survey had been made to determine if particulate activity was being released.

Since there is no flanged connection, the eight in. main steam line was cut at the turbine, between the trip-throttle valve and the steam chest. This was done by hand with a hacksaw since there is insufficient space for power tools. A cutting torch was not used because it was not known whether the inside surface of the pipe was activated.

The upper turbine casing was lifted two inches. A radioactivity check was then made to determine if particulate activity was being released. Air samples and smears were taken from several locations within the turbine with probes. No increase in radiation level in the area was apparent, so the casing was removed and the work continued.

Preliminary Survey. A complete survey of the turbine was made (10 days after shutdown) with results as follows:

<u>Location</u>	<u>Activity level</u>
No. 1 high pressure steam seal	2.0 mr/hr
First row of blades	0.2 mr/hr
Blades in all succeeding rows	0.2 mr/hr or less
Admission valve stems	3.0 mr/hr
Nozzle inlets	5.0 mr/hr
Inlet side of trip-throttle valve	2.0 mr/hr

Gamma Emitting Activities. The high activity found at the steam nozzle inlets was probably due to the tendency of the steam to condense in this region, and since there is little washing of the metal surface by

condensate here, particulate activity would tend to adhere. This would contrast to the blades where there is considerable washing.

During the period of service there had been a constant evolution of fission product gases (Kr and Xe) with the steam. They came from an estimated 30 to 40 mg of tramp uranium located on the outside of the fuel plates. In the investigation the daughter products from these gases were used as tracers to determine to what extent these fission products plate out in the steam system.

A sample removed from the eighth row of blades showed there was a deposit of approximately 1 mg per square in. This indicates that the turbine blades were as clean after one year of operation as when first put into operation.

Samples were taken from the outboard end of the high pressure gland seal; from the first, fifth, ninth, and thirteenth rows (numbered from the high pressure side) of turbine blades; from the condenser, and from the inlet steam line just upstream of the throttle valve. They were taken with a bearing scraper, removing metal as well as deposited scale. The gross gamma activity from the samples was so low that they had to be analyzed in an iron room where normally whole body gamma spectrums are made on humans. This work was done by the Radiological Physics Division. An example of the type of spectrum obtained from all the samples is shown in Fig. 53-1. It is essentially the spectrum of Co^{58} .

Fig. 53-1. Gamma spectrum from scraping taken from the entrance of the steam throttle valve on the EBWR turbine.

The scrapings were separated by chemistry to isolate the ruthenium and cobalt. It was found that there was no Ru^{106} - Rh^{106} . This isotope was selected since it is easily recognized, has no gaseous precursor and has a long half-life ($t_{1/2} = 1$ yr). After the cobalt had been separated, the residue was found to contain Ba^{140} - La^{140} , and traces of Fe^{59} and Cr^{51} .

The percentages of the long-life activities found in the turbine were estimated as follows:

Co^{58}	92%
Ba^{140} - La^{140}	8%
Co^{60} , Cr^{51} , Fe^{59}	trace

By using a geometrical efficiency given by C. E. Miller,¹ and a photoelectric efficiency or photofraction given by W. F. Miller,² it was possible to calculate roughly the concentration of Co^{58} activity at various points through the turbine. The data obtained are listed in Table I.

Table I - Co^{58} Concentration in the EBWR Turbine
(Microcurie per square inch)

High pressure gland	108×10^{-4}
First row of blades	5.23×10^{-4}
Fifth row of blades	46×10^{-4}
Ninth row of blades	28.8×10^{-4}
Thirteenth row of blades	1.78×10^{-4}
Condenser tubes	0.206×10^{-4}

The concentration of Co^{58} activity in the steam line could not be determined, since it was necessary to scrape a wide area to get enough sample to count.

The activity variation as shown in Table I may be due to the washing effect of condensate in the steam. In the region of the first row of blades there is little condensation as compared to the fifth row. Therefore, there is little deposition of particulate activity and little corrosion deposit. From the fifth row on through the condenser, the concentration decreases since both the area of blading and the washing effect from condensate increases.

¹C. E. Miller, not published.

²W. F. Miller et al., "Efficiencies and Photofractions for Sodium Iodide Crystals," The Review of Scientific Instruments, XVIII (1957) pp. 717-719.

All the observed activities can be explained. The Co^{58} , Co^{60} , Cr^{51} , and Fe^{59} are all present in the reactor water. There is a known decontamination factor of about 5×10^3 between the reactor water and the steam for solids. It is known that Co^{58} is the major long-life gamma emitting isotope in reactor water, and that there are smaller amounts of Cr^{51} , Co^{60} , and Fe^{59} . Thus, it is probable that these activities were carried over in the steam and deposited in the turbine. The Ba^{140} - La^{140} are released in the steam by Xe^{140} through the following chain:

The short half-life Xe^{140} , being a gas, is stripped from the reactor water by boiling and enters the steam. Moving through the steam system, it decays to Cs^{140} , which can deposit and decay to Ba^{140} . This mechanism works only for gases with half-lives short enough to decay in the steam system before being removed by the air ejectors of the main condenser. This indicates that a direct cycle boiling reactor will cleanse itself of fission product gases whose half-life is not too short, if a fuel element fails.

Mechanical Condition. In general, the turbine was in excellent condition. No wear or erosion could be noted on the blades or diaphragms. The blades were clean with only a small amount of deposit or discoloration. There was a small amount of greenish deposit on the no. one high pressure seal retainers and some also on top of the shrouds where the blade ends were peened. The bearing surfaces were in good condition with only a small amount of dirt accumulation in the grooves.

The steam seal labyrinths were sharp and showed no detectable wear. There was some rust on the rotor at both the high-pressure and low-pressure seals in the air pressure and vacuum regions. Pitting under the rust was slight.

The trip-throttle valve was clean and in good condition. The stem was measured and showed no change. The admission valve stems were covered with a very thin dark blue deposit. The valve seats were smooth and in good condition except for no. 6 which had been leaking. All were lapped in before assembly. The nickel plating (in general, everything but the rotating surfaces inside the turbine are plated) was in good condition.

The turbine oil was drained and the tank cleaned. The drained oil was settled for several days to separate sludge and water, then replaced in the turbine with makeup added as required. The oil coolers were cleaned and inspected.

The main steam line was beveled with a cutting torch and dressed with a grinding wheel for rewelding. The heliarc process was used for the first two passes. The completed weld was stress relieved by induction heating. The weld temperature was raised to 1160°F at the rate of 400°F per hour, then held for two hours. It was cooled down to 800°F at the same rate, then the coil turned off and the assembly allowed to cool naturally to room temperature. The weld was not radiographed.

Generator Check. The turbine-generator was shut down and immediately thereafter the resistance of the generator and exciter insulation was measured with a megger. The stator temperature was 40-44°C as obtained from the generator stator temperature indicator at the time of the measurement. The generator stator insulation was checked at the neutral, with the generator OCB open and the generator PTS in the disconnected position. The neutral was opened at a bolted connection on the high side of the neutral grounding transformer. Hence the measurement included neutral cable, generator stator windings (three phases in parallel), the generator cables to the switchgear and the surge protection equipment.

The generator rotor insulation was checked at the slip rings, with the brushes lifted. The insulation resistance of the exciter armature was measured at the commutator with the brushes lifted. All exciter fields were checked with one measurement at the gauge board, terminal board 88. Leads 1 and 5 were lifted and a measurement made on Lead 5 with the exciter brushes lifted, hence this reading included exciter field leads.

Results The readings obtained, using the procedure outlined below, are:

Generator stator	5 megohms
Generator rotor	.90-100 megohms
Exciter armature.	.20 megohms
Exciter field.	.14 megohms

It is recommended that all future readings be obtained in accordance with the above procedure, as a means of checking changes in the generator and exciter insulation values.

The end bells were removed from the exciter end of the generator. The rotor and stator windings were clean and in excellent condition. No abnormal conditions were found. The exciter was cleaned and reassembled with new brushes. New brushes were also installed on the slip rings. The air filters were replaced.

EBWR PRESSURE VESSEL GASKET CLOSURE

(Test Report No. 38, by J. M. Barrett)

The seventh set of gaskets installed duplicate those used for Test no. 6,¹ that is, Flexitallic 304 stainless steel spiral wound with teflon filler with 304 stainless steel restraint strips.

During the hydrostatic test, leakage was first observed at 800 psig. The following leakage rates were observed:

800 psig	-	1.6 cc/min
825 psig	-	6.8 cc/min
850 psig	-	7.2 cc/min

Five nuts were unusually hard to turn during removal of the vessel cover. All studs, washers and nuts were lubricated with graphite and oil.

EXAMINATION OF THE KANIGEN PLATING
IN THE EBWR MAIN STEAM LINE

(Test Report No. 58, by R. E. Bailey)

An examination was made of the Kanigen² coating in a vertical section of the main steam line just before it enters the turbine throttle valve. The samples used for hardness tests and for photomicrographs were taken from a ten in. section of this pipe. Metal supports had originally been welded to the outside of this section of pipe in two locations. Hence, there were two spots on this pipe which had been heat treated. An examination of the pipe with the naked eye revealed three things:

1. Two very smooth and shiny blue sections which were due to the heat treatment by the welding operation.
2. Small reddish brown out-cropping or blisters informally scattered over the interior surface of the pipe, except at the zones of heat treatment.
3. The rest of the surface was a light dirty brown.

Representative samples of each type of surface were taken for microscopic examination. Tukon hardness measurements were taken on the heat treated and untreated sections using the Knoop scale.

¹See Test Report no. 28 for prior tests.

²Trade name for a chemically deposited nickel plating process of the General American Transportation Corporation.

By means of the microscopic examination, it was found that the blisters were located on the surface of the Kanigen and did not penetrate through. Several of the blisters were removed and analyzed spectrographically. It was found that these blisters were mostly iron, hence, were weld spatter from the time when the pipe was originally welded to the turbine throttle valve.

The hardness of the Kanigen coating in the region of heat treatment was 8700, and 7800 in the untreated region (Knoop scale). This hardening has a beneficial effect on the durability of the Kanigen plating, as can be seen in Fig. 58-1. Observe that there is little evidence of erosion on the heat treated Kanigen, while there is considerable erosion on the untreated Kanigen. One discouraging item is the cracks which penetrate through to the base metal in the hardened Kanigen. These cracks could afford ample opportunity for crevice corrosion, especially in the highly oxygenated steam (28 ppm O_2) of EBWR.

It should be mentioned at this point that this observed erosion of the Kanigen plating was also noted on the turbine internals during the last inspection (Spring, 1958).

Whether or not a hardening of the Kanigen by heat treatment would have been beneficial to its protective quality is still a question, since the protective coating must have sufficient ductility or elasticity not to crack due to the expansion and contraction of the surface to which it is attached.

In conclusion, it can be stated that the Kanigen plating is not holding up under EBWR steam conditions.

NITOL ETCH

HEAT - TREATED KANIGEN

NITOL ETCH

NONHEAT - TREATED KANIGEN

50 X

Fig. 58-1

DETERMINATION OF AVERAGE EBWR CONTAINMENT VESSEL AIR TEMPERATURE

(Test Report No. 74A, by L. J. Weidner)

Data were taken over four separate 24 hr periods with a string of 15 iron-constantan thermocouples running in an east-west direction at three different elevations. These locations were:

<u>Date</u>	
July 9-10	2 ft above the turbine room crane.
July 10-11	Containment vessel dome with dome circulating fans off.
July 11-12	On turbine room floor level with the turbine stop valve.
July 17-18	Containment vessel dome with dome circulating fans off.

One thermocouple was mounted on the turbine stop valve hand wheel, which is near the center of the containment vessel. This point was used as a checkpoint and clearly indicated when the air conditioner was on or off.

Due to the limited time available for this test, pressure and humidity measurements were not made but outside temperature readings were obtained from meteorology to compare with the average vessel temperature. Data were tabulated in such a manner as to show the deviation in millivolts of all readings from the middle thermocouple of the string. This was done because the standard leak rate test averages five thermocouples strung vertically in the center of the containment vessel.

These data show that, when readings were taken in the containment vessel dome, the deviation from the center thermocouple is consistent over the 24 hr period. However, when the data from the readings taken beneath the containment vessel dome are compared with the center thermocouple, the deviation is random varying from plus to minus in one hour. For example, on July 11 and 12, between 2200 and 0400, the data show a deviation of:

T.C.	2200	2300	2400	0100	0200	0300	0400
E-10	+28	-18	-26	-5	+26	+18	-29

and on July 17 and 18, the same point when located in the containment vessel dome during the same time period, the data show the following deviation:

E-10	-29	-32	-33	-32	-36	-30	-30
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One other aspect of these data is the fact that the deviations of all readings taken in the containment vessel dome are consistent in sign i.e., all deviations are of a negative sign while the deviations of the readings taken below the dome are of a random sign plus or minus.

Also the maximum difference in temperature between the center thermocouple in the containment vessel dome and the thermocouple mounted on the turbine stop valve hand wheel was -526 Mvolts when the dome fans were not running and -334 Mvolts when the dome fans were running.

From these data, the following conclusions were derived:

1. During a regular leak rate test, the air conditioning equipment for cooling the air should be shut off though the fans should be left on.
2. One string of horizontal thermocouples should be mounted in the containment vessel dome and one string mounted on the turbine room floor, along with the previous thermocouples used, to determine the average temperature to use in the leak rate test.

EBWR BUILDING SHELL PRESSURE LEAK RATE TEST

(Test Report No. 78C, by D. J. Roy)

A leak rate test of the containment shell was conducted from 1100 hrs April 15, 1959 to 2300 hrs April 16, 1959. The leakage rate was found to be 309 ft³ of air at STP per day.

The equipment used in conducting the subject test consisted of a K-2 potentiometer for thermocouple readings (see Fig. 78C-1) and an absolute manometer for shell pressure readings.

A unique feature of this test is that data was processed by the IBM 704 while the shell was at pressure,¹ which allowed readings to be taken at fifteen minute intervals, rather than one hour intervals as in previous tests. The curves of the allowable leakage (1000 ft³) air at STP per day and the leakage rate of this test are shown in Fig. 78C-2.

A shell leak rate test is now considered to be routine.

¹Readings taken from 2200 hrs through 0800 hrs were processed at 0800 hrs.

Fig. 78C-1. Thermocouple layout

Fig. 78C-2. Containment shell leakage rate.

Test Report nos. 78, 78A, and 78B have been assigned to the original leak rate tests which are described in a paper entitled "Containment for EBWR" by A. H. Heineman and L. W. Fromm, P/1891 Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958. Volume II, page 139.

EBWR TURBINE SEAL LEAK TESTS, PRELIMINARY RESULTS

(Test Report No. 101A, by R. J. Gariboldi,
D. R. Jacobson, H. M. Sroka¹)

The EBWR turbine, feed pumps, throttle valve, bypass valve and turbine inlet valves employ seals designed especially for D₂O operation. A detailed description of the seals is presented in ANL-5607. To evaluate the performance of the seals and associated vapor recovery system, test apparatus has been installed and a series of tests are being conducted. To date, sufficient data have been recorded to draw conclusions about the condition of the seals.

In view of the immediate interest in this type of seal system for D₂O reactors, a summary of the data from the completed tests is reported herein.

Summary of Test Results

	<u>Measured²</u>	<u>ANL-5607 design</u>
1. Air Flow to Seals		
a. Turbine high pressure seal	8.89 scfm at +6.0 in. W.G.	7.53 scfm
b. Turbine low pressure seal	9.56 scfm at +6.0 in. W.G.	7.53 scfm
c. Feed pump seal	3.29 scfm at +6.0 in. W.G.	6.20 scfm
d. Inlet throttle and bypass valve seals	<u>.11</u> scfm at +6.0 in. W.G.	<u>0.70</u> scfm
Total	21.85 scfm at +6.0 in. W.G.	21.96 scfm
2. Air Flow Recovered		
a. Turbine high pressure seal	4.96 scfm at -6.0 in. W.G.	3.69 scfm
b. Turbine low pressure seal	5.17 scfm at -6.0 in. W.G.	3.69 scfm
c. Feed pump seal	2.00 scfm at -6.0 in. W.G.	3.10 scfm
d. Inlet throttle and bypass valve seals	<u>~.07</u> scfm at -6.0 in. W.G.	<u>0.50</u> scfm
Total	12.20 scfm at -6.0 in. W.G.	10.98 scfm

¹ Sargent & Lundy

² scfm corrected to 0°C and 760 mm Hg.

	<u>Measured¹</u>	<u>ANL-5607 design</u>
3. Air Lost to Atmosphere (also equals makeup flow)		
a. Turbine high pressure seal	3.93 scfm	3.84 scfm
b. Turbine low pressure seal	4.39 scfm	3.84 scfm
c. Feed pump seal	1.28 scfm	3.10 scfm
d. Inlet throttle and bypass valve seals	<u>~.04 scfm</u>	<u>0.20 scfm</u>
Total	9.64 scfm	10.98 scfm
4. Assuming recovery system can supply dry air with an average dew point less than -50°F, the moisture lost to the containment vessel would be:		
a. Turbine high pressure	0.53 lb/mo	
b. Turbine low pressure	0.60 lb/mo	
c. Feed pump	0.18 lb/mo	
d. Inlet throttle and bypass valve seals	<u>0.005 lb/mo</u>	
Total	1.3 lb/mo	
5. The average rate of atmospheric makeup air is nearly equal to the recovered air flow. Assuming the dew point of the dried makeup air is equal to the dew point of the dried recovered air, then half of the moisture lost through the seals is moisture that entered via the makeup system. If the primary steam is D ₂ O, approximately 0.7 lb per month of D ₂ O would be lost to the containment shell and 0.7 lb per month of H ₂ O would enter the primary system as a contaminant.		
6. Moisture recovered from the seals:		<u>Design ANL-5607</u>
	<u>Actual</u>	
a. Turbine high pressure	1.06 lb/min	0.55 lb/min
b. Turbine low pressure	1.06 lb/min	0.55 lb/min
c. Feed pump	not measured	
d. Inlet throttle and bypass valves	<u>0.37 lb/min</u>	<u>0.17 lb/min</u>
Total	2.49 lb/min	1.27 lb/min

¹ scfm corrected to 0°C and 760 mm Hg.

7. During the period that the tests were run, a failed motor bearing prevented operation of the recovered air desiccators. As a result, the dew point of the dried recovered air was $+40^{\circ}\text{F}$ to $+60^{\circ}\text{F}$. The dew point of the makeup air was measured as -50°F and the dew point of the air mixture was $+20^{\circ}\text{F}$ to $+40^{\circ}\text{F}$. The measurement of air flow through the seals is not seriously affected by the added moisture introduced as a result of the desiccator malfunction.
8. As a result of these tests, it can be concluded that the special seals used on the turbine are operating in accordance with design. Over the 4 yr period of operation, it does not appear that the effectiveness of the air-seal has been reduced. Recovered vapor rates indicate that the steam flow through the labyrinths is somewhat greater than design.
9. Difficulty was encountered with the inlet and bypass valve seals. Water was found to be back flowing into the dry air line. This is considered serious as the leakage could be D_2O at a rate of about 10-20 lb per week. Note that this does not represent a loss, since the stem packing appears to retain the leakage in the dry air chamber of the seal. The cause of the malfunction has not been fully analyzed, but it is believed that it can be corrected by pitching the vacuum line to cause condensate to drain to the vent condenser, rather than to the valve seal. This difficulty was not detected in the throttle valve seal which has the gland properly exhausted.
10. Air flow to the valve seals was much lower than originally expected, consequently, the valve seal flowrators selected had a range too high to accurately measure the flow. But, since the flows indicated were of such small magnitude, the lack of accuracy is considered unimportant to the gross results.
11. The rate of D_2O leakage is mainly dependent upon the quality of dry air achieved by the desiccators when the seals are operating properly.
12. Plant operating conditions during tests:
 - a. Turbine 3600 rpm, $\sim 1 \text{ Mw(e)}$
 - b. Total steam flow rate $\sim 20,000 \text{ lb/hr}$
 - c. Reactor pressure 600 psi
 - d. Seal steam pressure 2 psi

VOLUME OF WATER IN THE EBWR REACTOR

(Test Report No. 103, by C. E. Carson)

Information relating the reactor water level to the volume of water in the primary system was desired, particularly for use in determining boric acid concentration in the reactor. The results of several measurements, observations, and calculations are presented here. All necessary volumes are tabulated for the cold reactor. With the water level gauge reading 12 ft, water is contained in the pressure vessel (including the control rod drive thimbles and housings, forced circulation outlets and inlets) as well as in the startup heater circuit, shutdown cooling circuit, water column, and the old feedwater inlet line. At operating temperatures a few extra volumes would have to be considered, such as the condenser hotwell, various traps in lines, etc. These are not included here.

Tabulation of VolumesIn the pressure vesselNote: (+) indicates a containment of H₂O.(-) indicates a displacement of H₂O.

	Vol. (ft ³)	Vol. (gals)	
Clean pressure vessel (to ht. of 12 ft)	437.9	3275.9	(+)
Lower fixed shroud	2.7	20.2	(-)
Upper control rod shroud	1.0	7.5	(-)
Control rod guides	3.8	28.4	(-)
9 Control rods and followers (10 in. and 3 in.)	3.2	23.9	(-)
9 C.R. drive thimbles and housing (net)	5.7	42.6	(+)
Thermal shield	20.45	153.0	(-)
Shock shield (to 12 ft level)	.2	1.5	(-)
Core grid plate	4.2	31.4	(-)
Core hold-down assy.	.67	5.0	(-)
2 forced circ. outlet pipes (12 in. dia)	19.3	144.4	(+)
4 forced circ. inlets (6 in. dia)	6.3	47.1	(+)
		3239.1 gals	(+)

External to pressure vessel

Water column (to 12 ft level)	2.4	18.0	(+)
input and drain lines	.3	2.2	(+)
3 in. pipe between valve and old feedwater ring	1.55	11.6	(+)

	<u>Vol. (ft³)</u>	<u>Vol. (gals)</u>	
<u>Shutdown cooling</u>			
2 in. pipes between reactor and purification system (incl. 10 ft of reactor drain line)	2.4	18.0	(+)
1½ in. pipes between coolers and pumps (excl. 1.X. loops)	1.1	8.2	(+)
2 regenerative coolers	3.3	24.7	(+)
2 secondary coolers (tube side only)	1.9	14.2	(+)
2 purification pumps	estimated	2.0	(+)
<u>Startup heater circuit</u>			
3 in. pipe	3.15	23.6	(+)
4 in. pipe (to 12 ft level)	2.6	19.45	(+)
Startup heater	1.2	9.0	(+)
		150.95 gals	(+)

Displacement of H₂O by Fuel Elements

Thin-plate element:	1.05 gal/element
Thick-plate element:	1.32 gal/element
Spike element:	1.34 gal/element
Dummy element:	.11 gal/dummy
Neutron source:	1.6 gals.

For a core loading consisting of 58 thicks, 56 thins, 32 spikes and 2 dummies, total H₂O displaced by all elements and the neutron source = 180.1 gal. (-)

$$\text{H}_2\text{O Vol.} = 3239.1 + 150.95 - 180.1 - 9.0 = 3201 \text{ gal H}_2\text{O}$$

This value, 3201 gal, applies to the case in which the reactor is cold, with water level at 12 ft. The shutdown cooling system is operating (the ion exchange portion of the purification system is valved off). The startup heater itself is closed off by valves but the 3 in. and 4 in. pipes listed would contain water.

For each inch of water level change, the various volumes change as seen below:

Clean pressure vessel	23.99 gal/in. (+)
Shock shield	.1324 gal/in. (-)
Water column	.1249 gal/in. (+)
4 in. pipe between startup heater and steam drier	.0552 gal/in. (+)

Adding these changes algebraically results in a net change of 24.0 gal per in., in the vicinity of the 12 ft level.

Note: When the removable sections of the core riser are installed the amount of water in the system, at a level of 12 ft, decreases by 3.0 gal. Also, the calibration of the pressure vessel changes from 24.0 gal/in. to 23.9 gal/in., a small effect.

For the cold reactor, containing the complete spiked core, the results are summarized in the equation below:

$$\text{H}_2\text{O Vol.} = (3201 + 24 h) \text{ gal; where } h = (\text{water level} - 144) \text{ with} \\ \text{water level in inches.}$$

Note: "h" may be either a positive or negative number.

SECTION II POWER OUTPUT

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
8	Power Output Versus Galvanometer Reading, J. O. Schuyler	137
11	Decay Heating and Power Calibration Tests, A. Gibson	140
23, 23A, 23B	Power Run with Zero Steam Bypass, W. C. Lipinski.	156

EBWR POWER OUTPUT VERSUS GALVANOMETER READING

(Test Report No. 8, by J. O. Schuyler)

Data was taken on EBWR at various steady-state power levels to determine the variation of reactor steam flow and galvanometer reading with the reactor developed power. (Tables I and II).

The reactor developed power was determined by assuming saturated steam conditions and separate ion exchanger and feedwater loops. A schematic of the system is shown in Fig. 8-1.

$$\text{Reactor power (in kw)} = \frac{(h_1 - h_2)W + (h_{1X} - h_{2X})W_X}{3413}$$

where

h_1 = enthalpy of saturated steam at reactor water temperature.

h_2 = enthalpy of saturated liquid at hotwell temperature plus energy added by feed pump.

h_{1X} = enthalpy of saturated liquid to ion exchanger.

h_{2X} = enthalpy of saturated liquid from ion exchanger.

W = reactor steam flow in lb/hr.

W_X = ion exchanger flow in lb/hrs.

For an ion exchange flow of 7 gpm, the heat losses amounted to 206 kw. Figure 8-2 shows the variation of EBWR steam flow and galvanometer reading with developed power.

Table I - Calculated Reactor Output

h_1	Hotwell temp., °F	h_2	h_{1X}	Water from purification system temp., °F	h_{2X}	$h_1 - h_2$	$h_{1X} - h_{2X}$	W	$W_X, \times 10^3$	$(h_1 - h_2)W$ $\times 10^6$	$(h_{1X} - h_{2X})W_X$ $\times 10^6$	Reactor output,		
												Btu/hr $\times 10^6$	Mw	Galv.
1203	96	65	474	308	278	1138	196	60	3.58	68.2	0.7	68.9	20.2	22.0
1203	94	64	474	307	277	1139	197	60	3.58	68.2	0.7	68.9	20.2	22.0
1203	92	62	474	306	276	1141	198	50	3.6	57.1	0.7	57.8	16.9	18.0
1203	92	62	474	306	276	1141	198	50	3.6	57.1	0.7	57.8	16.9	18.0
1203	92	62	474	306	276	1141	198	50	3.6	57.1	0.7	57.8	16.9	18.0
1203	85	55	474	306	276	1148	198	40	3.75	45.9	0.7	46.6	13.7	14.2
1203	83	53	474	306	276	1150	198	40	3.75	46.0	0.7	46.7	13.7	14.2
1203	83	53	474	305	275	1150	199	40	3.75	46.0	0.7	46.7	13.7	14.2
1203	73	43	474	305	275	1160	199	30	3.55	34.8	0.7	35.5	10.4	10.9
1203	73	43	474	305	275	1160	199	30	3.55	34.8	0.7	35.5	10.4	10.9
1203	72	42	474	305	275	1161	199	30	3.55	34.8	0.7	24.1	10.4	10.9
1203	62	32	474	304	274	1171	200	20	3.58	23.4	0.7	24.1	7.1	7.4
1203	62	32	474	304	274	1171	200	20	3.58	23.4	0.7	24.1	7.1	7.4
1203	62	32	474	304	274	1171	200	20	3.58	23.4	0.7	12.7	7.1	7.4
1203	50	20	474	302	272	1183	202	10	4.38	11.8	0.9	12.7	3.7	3.6
1203	49	19	474	301	271	1184	203	10	4.38	11.8	0.9	12.7	3.7	3.6
1203	50	20	474	300	270	1183	204	10	4.38	11.8	0.9	12.7	3.7	3.6

 h_1 - enthalpy of reactor steam h_2 - enthalpy of reactor feedwater

W - reactor steam flow in thousand lbs. per hr.

 W_X - purification system flow h_{1X} - enthalpy of reactor water h_{2X} - enthalpy of water from purification system

Reactor steam conditions: saturated vapor at 600 psig

Table II - Flow through Purification System

Time	Steam flow, 1000 lb/hr	Period	Gallons	gpm	lb/hr
1700	60	35 min	250	7.15	3580
1800	50	25 min	180	7.2	3600
1900	40	60 min	450	7.5	3750
1955	30	55 min	390	7.1	3550
2055	20	60 min	430	7.15	3580
2200	10	55 min	480	8.75	4380
2225	10	25 min	90	3.6	1800

Fig. 8-1. EBWR flow diagram

Fig. 8-2. Steam flow and galvanometer reading vs developed power

DECAY HEATING AND POWER CALIBRATION TESTS

(Test Report No. 11, by A. Gibson)

Purpose. The tests were run to; (1) measure the decay heat produced by the core after reactor shutdown; (2) to measure the power produced by the reactor at constant power levels during a cold startup, for calibration purposes; (3) to observe any instability occurring during the cold startup.

Decay Heat Test. After a run of approximately $1\frac{1}{2}$ hours at 20 Mw, the reactor was shut down and cooled by means of the emergency cooler, and allowed to sit for a period of 72 hours. The reactor water purification system was then valved to bypass the ion exchange column, and to leave the regenerative heater in service. Cooling water to the cooler was shut off. Flow through the purification system thus permitted readings to be taken of reactor water temperature, with a minimum removal of heat. Additional readings were taken over a period of approximately 16 hours of the reactor skin temperature and the shield cooling water temperature.

Power Calibration Test Number 1. The reactor vent was left open to the condenser and the reactor water purification system was left as in the decay heat test. The control rods were pulled as near as possible in a bank to criticality and left at approximately a constant setting (12.5 in. approximate) for $22\frac{3}{4}$ minutes. The rods were then "scrammed" and the water and metal temperatures allowed to stabilize. Readings were taken of vessel and water temperatures, and linear flux readings vs time. The water temperature at the end of the run had reached the boiling point. During all runs, the turbine by-pass and stop valves were closed and the feed-water regulating valve was closed.

Since the observed temperature rise of the component parts of the vessel and the reactor water were not such as to lend themselves directly to the rate of heat increase at a given instant, the total heat input for the entire run was calculated by observing the initial and final stabilized temperatures. The average rate of heat rise was then obtained by dividing the total heat input by the total elapsed time that the unit was at power. It is noted on Fig. 11-1 that the temperature of the water stabilized at 219F, and it was assumed in the Calculations that the reactor was at the corresponding saturation pressure of 2 psig and some steam was escaping through the vent pipe (approximately 24 lbs/hr).

The observed water level of 12 ft -0 in. on the TV gage remained constant throughout the test. At a point at which boiling is believed to have commenced in the core, oscillations in the linear flux were observed as recorded by the micro-microammeter. These oscillations were approximately 8 percent of the full scale reading.

Fig. 11-1. Reactor heating test

Power Calibration Test Number 2. The reactor vent was closed and the purification system was re-valved to include the ion exchange column and cooling water to the cooler. The control rods were again withdrawn in a bank to approximately a constant setting (approximate 13.9 in.) and left for the run period of $1\frac{3}{4}$ hours. Readings were taken as in the preceding test, and in addition, the reactor pressure and condenser hotwell level were obtained. Power was calculated on this run by utilizing the rate of rise of temperature since it was observed to be fairly constant and since it was not allowed to level off after the run.

The observed water level was noted to drop 6 in. in the reactor during the run. Oscillations in the linear flux as recorded by the micro-microammeter were observed soon after criticality was reached. These oscillations were fairly regular in occurrence varying from $\frac{1}{15}$ cps to $\frac{1}{5}$ cps with a magnitude of ± 3 percent of the full scale reading. During these oscillations the pressure was gradually rising, and as noted above, the water level was falling. At a reactor pressure of 20 psi, and water level of 11 ft -8 in., the oscillations reached a maximum of approximately ± 6 percent. The oscillations gradually reduced in magnitude as pressure increased until at a pressure of 140 psig and water level of 11 ft -6 in., the nature of the oscillations changed. At approximately 15 second intervals the flux recorder pen dropped sharply approximately 8 percent of the full scale reading, and then rose sharply to level out at the initial value until the next dip 15 seconds later. These oscillations continued in decreasing magnitude as the pressure gradually rose. At approximately 175 psig and a level of 11 ft -6 in., water was added at a rate of 20,000 lbs per hr to the reactor to bring the water level up to 12 ft -0 in. As water was added the oscillations ceased. Power first increased by a factor of 2 as the effect of the cold water was felt in the reactor, and then decreased to the initial value of approximately $0.5 \times 1 \times 10^{-6}$ amps.

It is noted that the observed water level does not reflect the true water level in the reactor. If it is assumed that the water column was at 150°F, when the reactor water temperature was at 360°F and the indicated level at 11 ft -6 in., considering only density differences and not foaming at the surface due to boiling, etc., the level in the reactor was at 12 ft -8 in.

Decay Heat Test Results. The total decay heat generated during a 16.25 hour period was 2,446,000 Btu. The reactor had been operating at 20 Mw for $1\frac{1}{2}$ hours, after which it was inoperative for 72 hours, prior to the 16 hour test period.

The above heating rate corresponds to an average power rate of 44 kw (151,000 Btu/hr).

Power Calibration Test Number 1 Results. A total of 1,652,000 Btu (485 kw-hrs) was generated during the $22\frac{3}{4}$ minute period. This corresponds to 4,360,000 Btu/hr, or 1280 kw at an average micro-microammeter reading of $0.497 \times 5 \times 10^{-7}$ amps (2.49×10^{-7} amps). The approximate galvanometer reading was 218 kw.

Power Calibration Test Number 2 Results. The average heating rate was 5,027,000 Btu/hr or 1470 kw corresponding to an average micro-microammeter reading of $0.497 \times 1 \times 10^{-6}$ amps (4.97×10^{-7} amps).

Comparison Between the Two Power Calibration Runs. A quite favorable correlation was obtained between the two runs as indicated below:

	<u>Micro-microammeter</u>	<u>Calculated</u>
Power ratio between tests:	$\frac{4.97 \times 10^{-7}}{2.49 \times 10^{-7}} = 2.0$	$\frac{1470}{1280} = 1.15$

It is believed that the difference in correlation is the result of non-linear flux changes with linear changes in power at the location of the ionization chamber, and is not due to inaccuracies resulting from the methods used to compute power. It is noted that the average slopes of the temperature curves on Fig. 11-1 for the two runs are approximately the same and since power is proportional to the rate of change of temperature (e.g. $dQ/dO = W c_p dT/dO$), it would appear that the second run was approximately at the same power level as was the first test.

It was also noted that on Test No. 1, the micro-microammeter reading of $0.495 \times 5 \times 10^{-7}$ amps is approximately a factor of 10^{-2} lower than the 20 Mw reactor power reading of $0.5 \times 5 \times 10^{-5}$ amps and therefore corresponded roughly to the galvanometer reading of 218 kw. This was to be expected, since the ionization chambers are adjacent to each other in the instrument tubes. Again, it is believed that the galvanometer and micro-microammeter are reading low by a factor of approx. $1200/200 = 6$ due to non-linear flux changes (perhaps caused by shadow effects from control rods, etc.)

Calculations.

1. Zero Power Heating Test. (1730 on 4-15-57 to 0945 on 4-16-57)
(Rods all inserted throughout test)

a) Vessel and Shield Weights

Wt. steel in vessel	129,500 lb
Flange	9,300 lb
Thermal shield	22,000 lb
Structural core	44,200 lb
Total	<u>205,000 lb</u>

Lead biological shield around vessel:

Approximate dimensions:

Bottom 1/2 - 8.5' id x 4" thick x 12' high

$$\text{Volume} = \frac{\pi}{4} \left[(8.5)^2 - (7.83)^2 \right] \times 12 = \frac{10.8 \times \pi \times 12}{4} = 102 \text{ ft}^3$$

Upper 1/2 - 8.5' id x 3" thick x 11' high

$$\text{Volume} = \frac{\pi}{4} \left[(8.5)^2 - (8.0)^2 \right] \times 11 = \frac{8.2 \times \pi \times 11}{4} = 71 \text{ ft}^3$$

Total Volume = 173 ft³

Lead = 709 lb/ft³

Approximate weight: 709 x 173 = 122,600 lb.

b) Heat Added to Upper Portion of Vessel (metal)

From test data and Fig. 11-1, the average metal temperature of the upper 1/2 of the vessel (exclusive of flange) followed slightly below the readings obtained from thermocouple 11. It is assumed 1/2 of the vessel steel weight followed this cycle.

Temperature rise $\approx 179 - 158 = 21^\circ\text{F}$

$$Q_{\text{upper vessel}} = W C_p \Delta t = \frac{129,500}{2} \times 0.12 \times 21 = 163,000 \text{ Btu.}$$

c) Heat Exchange from Top of Reactor

From thermocouple 13, temperature drop 188 - 177 = 11°F

$$Q_{\text{flange}} = 9300 \times 0.12 \times 11 = 12,300 \text{ Btu}$$

Assume that this heat raised the upper wall temperatures down to and including thermocouple 12.

d) Radiation from Top of Reactor to Room

Three hold down beams

Surface area of beams exposed to air

$$\text{Top surface} = 3 \times 2.25 \times 22 = 149 \text{ ft}^2$$

$$\text{Side surface} = 4 \times 0.417(2 \times 1.125) \times 22 = 86 \text{ ft}^2$$

$$\text{Total Beam Surface} = 235 \text{ ft}^2$$

Metal weight of beams at top of reactor:

$$W/\text{beam} = [0.417 \times 2.25 \times 2 + 0.167 \times 1.125] \times 22 \times 489 = \\ 2.068 \times 22 \times 489 = 22,200 \text{ lb } 3 \text{ beams} = 66,600 \text{ lb}$$

Top of water tank over reactor - exposed surface:

$$A_1 = \frac{\pi r^2 \theta}{360} - \frac{r^2 \sin \theta}{2}$$

$$\sin \frac{\theta}{2} = \frac{4.5}{6} = 0.750 \quad \frac{\theta}{2} \cong 48.5^\circ$$

$$\sin \theta = 0.9925 \quad \theta = 97^\circ$$

$$A_1 = \frac{\pi \times 36 \times 97}{360} - \frac{36 \times 0.9925}{2} = 30.5 - 17.9 = 12.6 \text{ ft}^2$$

$$\text{Area of arc, } A_2 = r \theta \times 0.5 = 6 \times \frac{97}{90} \pi \times 0.5 = 10.2 \text{ ft}^2$$

$$\text{Total area 2 arcs} = 2 \times 22.8 = 45.6 \text{ ft}^2$$

Weight of metal - top of tank:

$$W = \pi r^2 h \rho = \pi \times 36 \times 0.5 \times 489 = 27,600 \text{ lb}$$

Radiation losses from beams:

$$q = \alpha F_e F_A A_1 (T_1^4 - T_2^4)$$

$$T_1, \text{ est. avg. temp. at top of beams} = 120^\circ\text{F} (580^\circ\text{R})$$

$$T_2, \text{ room temp.} = 75^\circ\text{F} (535^\circ\text{R})$$

$$A_1, \text{ Surf. area} = 235 \text{ ft}^2$$

$$\alpha, \text{ Stefan Boltzman Const.} = 0.174 \times 10^{-8}$$

$$F_e, \text{ emissivity factor} \cong 0.8 \text{ for completely enclosed body,} \\ \text{small compared to enclosure,}$$

$$F_A, \text{ Configuration factor} = 1$$

$$\therefore q = 0.174 \times 10^{-8} \times 0.8 \times 1 \times 235 (580^4 - 535^4)$$

$$q = 0.174 \times 10^{-8} \times 0.8 \times 2.35 \times 10^2 \times 3.15 \times 10^{10} = 1.03 \times 10^4 \text{ Btu/hr} \\ = 10,300 \text{ Btu/hr}$$

Radiation losses from exposed top of water tank:

$$\text{Est. avg. temp. top of tank at } 109^\circ\text{F} (650^\circ\text{R})$$

$$q = 0.174 \times 10^{-8} \times 0.8 \times 1 \times 46 (650^4 - 535^4)$$

$$q = 0.174 \times 10^{-8} \times 0.8 \times 4.6 \times 10 \times 9.75 \times 10^{10} = 0.624 \times 10^4 \text{ Btu/hr}$$

Total radiation losses from top of reactor:

$$q_{\text{total R}} = (1.03 + 0.624) \times 10^4 \cong 1.65 \times 10^4 \text{ Btu/hr}$$

- e) Free Convection Losses from Top of Reactor to Room
(From Brown and Marco - "Introduction to Heat Transfer," page 115.)

$$h_c = C K \left(\frac{a \theta}{L} \right)^{\frac{1}{4}} \quad \text{If } L > 2 \text{ use } L = 2 \text{ for horizontal plates}$$

$$(\text{From page 213}), \text{ For average temperature } \frac{140 + 75}{2} = 107^\circ\text{F}$$

$$\text{for air, } a = 1.2 \times 10^6 \quad K = 0.0147$$

$$\theta = \text{temperature drop} = 140 - 75 = 65^\circ\text{F}$$

$$aL^3 \theta = 1.2 \times 10^6 \times 8 \times 65 = 62.4 \times 10^7 < 10^9$$

So above equation is correct.

(From page 114), $C = 0.71$ for horizontal plates facing up

$$h_c = 0.71 \times 0.0147 \left(\frac{1.2 \times 10^6 \times 65}{2} \right)^{\frac{1}{4}} = 0.71 \times 0.0147 \times 2.5$$

$$h_c = 0.0261 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

$$q = h_c A \theta = 0.0261 \times 281 \times 65 = 0.476 \times 10^3$$

$$= 476 \text{ Btu/hr}$$

- f) Total Convection and Radiation Losses from Top of Beams Over Reactor

$$q_{\text{total}} = 16,500 + 500 = 17,000 \text{ Btu/hr}$$

For total period of 16.25 hours

$$Q_{\text{top}} = 17,000 \times 16.25 = 277,000 \text{ Btu}$$

- g) Actual Heat Losses from Inside Reactor through Top
Heat capacity of beams and top of water tank:

$$\text{Beams } 66,600 \times 0.12 \quad 8,000$$

$$\text{Top of tank } 27,600 \times 0.12 = 3,300$$

$$\text{Equiv. wt. of water} \quad \underline{11,300 \text{ lb}}$$

Weight of water in water tank:

$$W_{H_2O} = \frac{\pi}{4} \times 11.5^2 \times \frac{11}{12} \times 62.4 = \frac{\pi \times 132 \times 11 \times 62.4}{4 \times 12}$$

$$W_{H_2O} = 5930 + 11300 = 17230 \text{ lb} = W \text{ total top}$$

$$Q = W C_p \Delta t$$

$$\Delta t = \frac{277,000}{1 \times 17,230} = 16^\circ\text{F}$$

Since no condensation of steam was occurring at the top of the vessel during the run, and since there is insulation between the water tank and the top of the vessel, it is assumed that there were no losses from the top of the vessel itself. All radiation and convection losses from the top of the beams and water tank were supplied by the heat capacity of the tank and beams, which, in turn dropped 16°F during the run.

h) Heat Added to Bottom Half of Vessel (metal)

Bottom half of vessel averaged the readings shown on thermocouple 3.

$$Q_{\text{bottom}} = W C_p \Delta t = \frac{129,500}{2} \times 0.12 \times (184 - 132) = 404,000 \text{ Btu}$$

i) Heat Added to Shielding (thermal)

Assume temperature follows water temperature rise.

$$Q_{\text{thermal shield}} = W C_p \Delta t = 22,000 \times 0.12 \times (186 - 138) \\ = 127,000 \text{ Btu}$$

j) Heat Added to Core

$$W \text{ core total} = 44,200 \text{ lb}$$

$$\text{Approx. uranium weight} = \frac{14,000}{30,200}$$

$$\text{Est. Zr. wt.} = 0.8 \times 30,000 = \underline{24,000 \text{ lb}}$$

$$\text{Rods and misc.} = 6,200 \text{ lb}$$

Equivalent water weight of core:

$$\text{Uranium} = 14,000 \times 0.028 \cong 400$$

$$\text{Zirconium} = 24,000 \times 0.067 = 1,600$$

$$\text{Stainless} = 6,200 \times 0.12 \cong \underline{\underline{500}}$$

$$\text{Total Equiv. Weight} = 2,500 \text{ lb}$$

Assume core temperature follows water temperature

$$Q_{\text{core}} = W C_p \Delta t = 2500 \times 1 \times 48 = 120,000 \text{ Btu}$$

k) Heat Losses to Reactor Water Purification System

Average temperature rise of water = 5°F

Calc. of flow rate with IX column bypassed:

(From test data on 4-22-57 by N. Kershaw - with IX in service.)

$$P_1 = 5.0 \text{ psig}$$

$$P_2 = 3.9 \text{ psi}$$

$$P_3 = 8.6 \text{ psi}$$

$$P_4 = 10.3$$

$$\text{Max. Flow} = 5.7 \text{ gpm}$$

From manufacturers pump curve,

At 5.7 gpm, TDH = 120 feet

Total drop thru IX loop = 15.6 psi = 36.0 feet

120 total

-36 IX loop

84 feet thru rest of system including reg. htrs., cooler
and piping at 5.7 gpm

$$\frac{\Delta P_2}{\Delta P_1} = \left(\frac{W_2}{W_1} \right)^2 \quad W_2 = W_1 \sqrt{\frac{\Delta P_2}{\Delta P_1}}$$

Assume $\Delta P_2 = 119$ feet:

$$W_2 = 5.7 \sqrt{\frac{119}{84}} = 6.8 \text{ gpm} = \text{Flow rate thru RPS with IX Column bypassed reg. htr. and cooler in service.}$$

From test data, meas. flow rate $W/\text{col. in service} = 8.1 \text{ gpm}$.
Assuming same ratio as above holds, flow rate with IX bypassed is:

$$W = \frac{6.8}{5.7} \times 8.1 = 9.67 \text{ gpm}$$

Heat losses:

$$q = W C_p \Delta t = 9.67 \times 1 \times 5 \times 500 = 24,200 \text{ Btu/hr}$$

For 16.25 hrs.

$$Q_{IX} = 16.25 \times 17,000 = 392,000 \text{ Btu.}$$

(Note: It is assumed that the observed pump flow rate of 8.1 gpm with the reactor at pressure is greater than the observed 5.7 gpm, with no press. in reactor due to less cavitation.)

- 1) Heat Lost to Lead Biological Shield Thru Insulation
Observed water temperature in and out of shield rose from 99°F to 100°F throughout test run.

$$Q_{\text{biological shield}} = W C_p \Delta t = 122,000 \times 0.031 \times 1 \\ \approx 4000 \text{ Btu}$$

- m) Heat Added to Reactor Water
(From C. Zitec)

The measured water quantity in the reactor is 3100 gallons at 12' - 0" on TV gage (water at ambient temperature)

It is believed that the observed level in the TV gage does not change markedly with changing reactor temperature since the water column stays relatively cold. The observed level during test was 12' - 0"

$$\therefore \text{Weight of water} = 3100 \times 8.33 = 25,800 \text{ lb}$$

$$\text{Observed temperature rise} = 48^\circ\text{F} (186 - 138)$$

$$Q_{\text{water}} = W C_p \Delta t = 25,800 \times 1 \times 48 = 1,240,000 \text{ Btu}$$

- n) Total Decay Heat Added by Core - in a period of 16.25 hours of decay heating 72 hrs after a reactor power run of 20 Mw for a period of $1\frac{1}{2}$ hours.

$$\text{kw-hrs} = \frac{2.446 \times 10^6}{3.413 \times 10^3} = 0.717 \times 10^3 = 717 \text{ kw-hrs}$$

$Q_{\text{upper vessel}}$	163,000 Btu
$Q_{\text{bottom vessel}}$	404,000
$Q_{\text{thermal shield}}$	127,000
Q_{core}	120,000
Q_{IX}	392,000
Q_{water}	1,240,000
Q_{total}	<u>2,446,000 Btu</u>

o) Average Heating Rate

$$q_{avg} = \frac{2.446 \times 10^6}{16.25} = 151,000 \text{ Btu/hr}$$

$$\text{Average Power} = \frac{151,000}{3413} = 44.1 \text{ kw}$$

2. Power Calibration Test. (Run 46-44 on 4-16-57) (from 1033 $\frac{1}{4}$ to 1056) (Rods at approx 12.5")a) Heat Added to Upper Portion of Vessel

From test data, thermocouples 8 thru 12 or approx. $\frac{1}{2}$ of the vessel, were at approximately the same temperatures throughout the test. Average temperature rise (216 - 180) = 36°F to stable upper temp.

$$Q_{upper \text{ vessel}} = W C_p \Delta t = \frac{129,500}{2} \times 0.12 \times 36 = 280,000 \text{ Btu}$$

b) Heat Added to Flange

From thermocouple 13, temperature rise (212 - 177) = 35°F

$$Q_{flange} = W C_p \Delta t = 9300 \times 0.12 \times 35 = 39,000 \text{ Btu}$$

c) Heat Added to Lower Portion of Vessel

(From test data and Fig. 11-1)

Average temperature rise = 217 - 185 = 32°F to stable upper temp.

$$Q_{lower \text{ vessel}} = \frac{129,500}{2} \times 0.12 \times 32 = 248,000 \text{ Btu}$$

d) Heat Added to Thermal Shield

Assume 1 that the thermal shield temperature rise is the same as thermocouple 6.

$$\text{Temperature rise} = 218 - 185 = 33^\circ\text{F}$$

$$Q_{thermal \text{ shield}} = W C_p \Delta t = 22,000 \times 0.12 \times 33 = 87,000 \text{ Btu}$$

e) Heat Added to Core

From calc. in 1.j., equivalent water weight for core = 2500 lb

Assume core temperature rise = 35°F

$$Q_{core} = 2500 \times 1 \times 35 = 88,000 \text{ Btu}$$

f) Heat Loss to Reactor Water Purification System

From calc. in l.k., flow rate = 9.67 gpm

Average temperature rise of water in and out of system during run = 6°F

$$q_{IX} = 9.67 \times 500 \times 1 \times 6 = 29,000 \text{ Btu/hr}$$

Test time 22.75 minutes

$$Q_{IX} = \frac{29,000 \times 22.75}{60} = 11,000 \text{ Btu}$$

g) Heat Added to Reactor Water

Temperature rise = 219 - 188 = 31°F

TV level = 12' - 0" throughout test

∴ (from l. m., calc., water) = 3100 gal. 25,800 lb

$$Q_{\text{water}} = 25 \times 800 \times 1 \times 31 = 800,000 \text{ Btu}$$

It is noted that the water temperature rose to 219°F with the reactor vent open. Inasmuch as this line is a long $\frac{1}{2}$ in. internal bore pipe, it is believed that the 219°F was correct. Air, and later some steam, were escaping in sufficient quantities to cause a 2 psi pressure drop to the condenser resulting in 2 psig pressure in the reactor and the corresponding 219°F saturation temperature. Heat added to steam generated in reactor:

800 ft³ in whole reactor, or 400 ft³ approx. in steam space.

$$W_s = \frac{400 \text{ ft}^3}{23.4 \text{ ft}^3/\text{lb}} = 17.1 \text{ lb}$$

$$Q_{\text{steam in vessel}} = 17.1 \text{ lb} \times 966 \text{ Btu/lb} = 16,500 \text{ Btu}$$

Heat lost through vent pipe:

Est. 200 equivalent feet of $\frac{1}{2}$ in. id pipe

$$\Delta P_{\text{total}} = 2 \text{ psi} \quad \Delta P/100 = \frac{2 \text{ psi}}{200} = 1 \text{ psi}/100$$

$$\Delta P/100 = \frac{3.36 \times 10^{-4} \times f \times W^2 \times \bar{v}}{d^5} \quad d^5 = 0.0931$$

$$W^2 = \frac{\Delta P/100 \times d^5}{3.36 \times 10^{-4} \times f \times \bar{v}} \quad f = 0.02$$

$$\bar{v} = 23.4$$

$$W^2 = \frac{1 \times 9.31 \times 10^{-2}}{3.36 \times 10^{-4} \times 0.02 \times 23.4} = 5.92 \times 10^2 = 592$$

$$W = 24 \text{ lb/hr}$$

From Fig. 11-1, the water was above 212°F for approx. 0.2 hour before it stabilized at 219°F

$$\therefore Q_{\text{ventpipe}} = 24 \times 0.2 \times 966 \approx 4600 \text{ Btu}$$

h) Radiation and Convection Losses from Top of Reactor

It is assumed that the radiation loss rate from the beams is approximately the same as that calculated in i.f., or 17,000 Btu/hr. Since these losses are relatively small, no attempt will be made to accurately determine the rate of heat transfer between the vessel cover, or flange, and the water tank and beams. It will be assumed, that since this test run period was small and steam was generated only part of the time, only $\frac{1}{2}$ of the losses were supplied to the beams, and they continued to drop in temperature (This should be on the high side).

$$\therefore Q_{\text{radiation}} \approx \frac{17,000}{2} \times \frac{22.75}{60} = 3200 \text{ Btu}$$

i) Heat Losses to Piping and Steam Dryer

It is assumed that the main steam line and the 6" line to the safety valves (and the steam dryer) filled with steam and were brought up to saturation temperature with no radiation losses.

Approx. 120' of 6" sch 40 pipe at 19 lb/ft	= 2280 lb
Steam dryer	= 6800
Total metal weight	= 9080 lb

Estimate initial average temperature at	150°F
---	-------

Estimate final average temperature at	219°F
---------------------------------------	-------

Δt	= 69°F
------------	--------

$$Q_{\text{piping and dryer}} = 9080 \times 0.12 \times 69 = 75,200 \text{ Btu}$$

(approximately 75 lb of steam)

j) Total Heat Added by Core During Run 46-44

$Q_{\text{upper vessel}}$	280,000 Btu
Q_{flange}	39,000
$Q_{\text{lower vessel}}$	248,000
$Q_{\text{thermal shield}}$	87,000
Q_{core}	88,000
$Q_{\text{IX System}}$	11,000
$Q_{\text{reactor water (to sat.)}}$	800,000
$Q_{\text{steam in vessel}}$	16,500
$Q_{\text{radiation and convec. top}}$	3,200
$Q_{\text{steam thru vent pipe}}$	4,600
$Q_{\text{piping and dryer}}$	75,200
Q_{total}	= 1,652,500 Btu

$$\text{kw-hrs} = \frac{16.53 \times 10^5}{3.413 \times 10^3} = 4.85 \times 10^2 = 485 \text{ kw-hrs during } 22\frac{3}{4} \text{ minute period.}$$

k) Average Heating Rate

$$q_{\text{avg}} = \frac{16.53 \times 10^5 \times 60}{22.75} = 43.6 \times 10^5 = 4,360,000 \text{ Btu/hr}$$

$$\text{Average Power} = \frac{4,360,000}{3413} = 1280 \text{ kw}$$

This corresponds to an average micro-microammeter reading of

$$0.497 \times 5 \times 10^{-7} \text{ amps}$$

and approximately 218 kw on a galvanometer.

3. Power Calibration Test. (Run 46-45 on 4-16-57 from 1215 to 1400)

It is noted on this run that after the water temperature stabilized, at approximately 1245, the rate of rise of temperature was approximately constant and the same for all thermocouples.

$$\text{Rate of rise} = 90^\circ \text{ F/hour} = \frac{dT}{d\theta}$$

$$d_q = \frac{dQ}{d\theta} = W C_p \frac{dT}{d\theta}$$

a) Heat Added to Vessel and Core

Vessel steel = 129,500 x 0.12	= 15,500 lb
Flange = 9,300 x 0.12	= 1,100
Thermal Shield = 22,000 x 0.12	= 2,600
Core (from page 5)	= 2,500
	<u>21,700 lb</u>

$$d_{q_{\text{reactor metal}}} = 21,700 \times 1 \times 90 = 1,950,000 \text{ Btu/hr}$$

b) Heat Added to Water

Water level varied from 11' - 10" to 11' - 6"

$$4" \times 24 \text{ gal/inch} = 96 \text{ gal.}$$

$$\text{Average water thru test} = 3100 - 48 = 3052 \text{ gal.}$$

$$3052 \times 8.33 = 25,400 \text{ lb}$$

$$d_{q_{\text{water}}} = 25,400 \times 1 \times 90 = 2,290,000 \text{ Btu/hr}$$

c) Radiation and Convection Losses from Top of Reactor

It is assumed that the radiation loss rate from the reactor averaged approximately 17,000 Btu/hr during the run. (1. f.)

d) Heat Lost to IX System

8.1 gpm

Average temperature rise of water thru system (valued for IX column on and cooling water on) = 100°F

$$d_{q_{\text{IX}}} = 8.1 \times 500 \times 1 \times 100 = 405,000 \text{ Btu/hr}$$

e) Heat Lost Thru Traps to Condenser (condensed steam)

$$h_{fg \text{ avg.}} = \frac{966 + 862}{2} = 914 \text{ Btu/lb}$$

Hotwell level rise = 2.2 in. $\approx$ 84 gallons = 700 lb condensate

$$d_{q_{\text{traps}}} = \frac{700 \text{ lb} \times 914 \text{ Btu/lb}}{1.75 \text{ hrs}} = 365,000 \text{ Btu/hr}$$

f) Total Heating Rate During Run 46-45

$d_{q_{\text{reactor metal}}}$	1,950,000 Btu/hr
$d_{q_{\text{water}}}$	2,290,000
$d_{q_{\text{radiation and convec.}}}$	17,000
$d_{q_{\text{IX system}}}$	405,000
$d_{q_{\text{traps}}}$	<u>365,000</u>
	5,027,000 Btu/hr

$$\text{Average Power} = \frac{5,027,000}{3413} = 1470 \text{ kw}$$

This corresponds to an average micro-microammeter reading of $0.497 \times 1 \times 10^{-6}$ amps.

EBWR POWER RUN WITH ZERO STEAM BYPASS

(Test Reports 23, 23A, 23B, by W. C. Lipinski)

Introduction. EBWR has always been run with some steam bypass. The purpose of the experiment was to determine the feasibility of operating with zero steam bypass limits by moving control rods. The results of the experiment will provide information which can be applied to the design of an automatic reactor control system.

Experimental Procedure.

1. Flux, control rod position, reactor pressure, reactor steam flow, feedwater flow, generator electrical output were maintained and observed.
2. The first phase of the experiment involved calibrating the center control rod to determine the exact rod position for 20 Mw output. During the course of the experiment, reactor power changed by moving the center rod, and under no conditions was the rod withdrawn beyond the calibrated 20 Mw rod position.
3. With the turbine at 20 Mw and the reactor delivering 60,000 lb/hr steam flow, the operator ran in the center control rod and closed the steam bypass valve. The parameters the operator observed were: bypass valve position, reactor pressure, and rod position. The data taken were: rod position and time. Of primary interest was the degree of difficulty which the operator experienced in adjusting rod position to affect a change in steam bypass valve position.
4. The P_0 (pressure reference) setting on the steam bypass system was increased 10 psi and the operator attempted to maintain reactor pressure within 590 to 610 psi by manipulating the center control rod. In the event the operator had misjudged and called for a large rod motion and resulting power increase, the steam bypass valve would have functioned automatically at 610 psi and regulated the pressure.
5. The generator load was increased, and the reactor operator adjusted the center control rod position to maintain reactor pressure. A Brush recorder with an expanded scale was provided for the operator to observe differential pressure changes. The generator load was increased to 4.0 megawatts at 200 kw per minute.

6. The generator load was decreased from 4.0 megawatts and the reactor operator adjusted the center control rod position in the following manner: When the steam bypass valve position indicator read 10%, the operator ran the center rod in until the reading was reduced to 5%. As the electrical load was decreased further, the bypass valve position was eventually increased to 10%. The operator repeated and ran the rod in until the bypass valve was 5% open. Electrical power was decreased to 2.0 Mw at 200 kw per minute.
7. The generator load was increased from 2.0 Mw to 4.0 Mw at 200 kw per minute. The reactor operator adjusted the center rod in the following manner: When the bypass valve closed the center rod was withdrawn until a reading of 5% was obtained on the bypass valve position indicator. As the electrical load was increased the bypass closed. The operator withdrew the control rod until the bypass valve position indicator read 5%.

Test Data.

1. Preliminary to running with zero bypass, a check was run on the operation of the Foster steam powered back pressure relief valve. Reactor pressure was increased from 600 to 617 psi on the televised pressure gauge, and reactor steam flow decreased from 60,000 lb/hr. to 52,000 lb/hr. Reactor pressure was then reduced to 610 psi, and reactor flow was 56,000 lb/hr. The pressure was further reduced to 600 psi, and reactor steam flow was 60,000 lb/hr. On the basis of these results it was decided to perform the zero bypass test at pressures below 600 psi.
2. The conditions prior to the start of the test were:
 - Reactor steam flow 60,000 lb/hr.
 - Rods 1, 3, 5, 7, and 9 at 26.0 inches.
 - Rods 2, 4, 6, and 8 at 47.0 inches.
 - Reactor pressure 590 psi.
 - P₀ setting 625.
 - Bypass valve position 20%.
 - Generator output 20 Mw.
3. At 10:30 A.M. reactor power was decreased by lowering the center rod to 11.0 inches, and the bypass valve was closed. The P₀ setting was increased to 640. Reactor pressure dropped at 0.2 psi/sec. to 562 psi. At 10:36 the center rod was withdrawn to 15.0 inches and the pressure increased at 0.12 psi/sec. to 582 psi. At 10:38 the rod was run in to 14.0 inches. Reactor

pressure decreased at 0.084 psi/sec to 576 psi. At 10:39½ the rod was withdrawn to 14.5 inches. Reactor pressure reached an equilibrium value of 574 psi. Slight variations were noted in generator output which may be associated with line frequency variation or with governor control characteristics. Reactor pressure varied slowly with time and at 11:02 increased to 584 psi. At 11:40 the pressure had decreased to 540 psi, and the center rod was withdrawn to 14.6 inches. The pressure stabilized at 540 psi. At 11:45 the rod was withdrawn to 14.7 inches and pressure increased to a new value of 543 psi. At 11:58 the rod was withdrawn from 15.0 inches to 16.0 inches. Reactor power increased by 10% and the pressure increased from 550 psi to 580 psi at 11:61. The reactor was operated with the bypass valve closed until 13:59. The final reactor pressure was 560 psi and the rod at 16.5 inches.

The above results show the self-controlling effect due to the negative temperature coefficient. The long time fall-off in pressure was due to xenon build-up after a reduction in power from 100% to 50%.

4. At 13:58 the reactor operator was instructed to try to increase reactor pressure by running the rod at 0.5 in/sec. for one second every five seconds or an average of 0.1 in/sec. Information for moving the rod was obtained from the reactor pressure indicator on the control console. The rod was run out when the pressure dropped to 540 psi and stopped at 550 psi. The rod was run in when the pressure reached 560 psi and stopped at 550 psi. Reactor flux and reactor pressure oscillated sinusoidally with a frequency of 0.0056 cps. Reactor flux showed a peak to peak variation of 40% with a phase load of 72 degrees.

The above results show that running a control rod at constant velocity in response to a pressure error is an unstable method of control which will oscillate continuously. The frequency of oscillation is determined by the rod velocity and allowable pressure deviation. If the rod velocity were made slow enough, the negative temperature coefficient would be effective and would provide a stabilizing influence. The data of paragraph 3 showed that rod motions of 0.1 inches every 60 seconds would allow reactor pressure to reach new equilibrium values without overshoots. This slow rod motion would limit the rate at which reactor power could follow generator load changes.

5. The reactor operator was instructed to watch the Brush recorder chart and to adjust reactor power as the generator load was increased at 200 kw/min. to 3900 kw and then decreased to 2000 kw. By observing the chart and taking advantage of noting the rate at which reactor pressure was changing, the operator was able to manipulate the control rod without exceeding 10 psi from the average pressure.
6. The P_0 setting was reduced to 600 and the TV gauge indicated 560 psi with 10% bypass. Generator power was increased from 2 Mw at 200 kw/min. The control rod was run 0.1 inches every five seconds and the operator tried to hold 10% bypass. The generator was loaded to 4.0 Mw. The control rod was at 30.6 inches, reactor steam flow was 56,000 lb/hr, with 10% bypass. Generator power was decreased to 3.0 Mw at 200 kw/min. The control rod was run in 0.1 inches every five seconds. A final rod position of 24.2 inches was reached. The operator experienced no difficulty in trying to move the control rod in response to the bypass valve position changes.

In a practical control system the maximum travel of the rod used for control must be limited. The control rod limit should be based on maximum power plus maximum xenon build-up. Under full power equilibrium conditions, reactor power could be increased by running the rod to its mechanical limit. Therefore high flux trips must be used to monitor the operation of an automatic control system.

Bypass Test Number 2. The purpose of the experiment was to repeat the work performed previously except an on-off relay amplifier was used to move the center control rod in response to a pressure error.

1. The initial conditions were as follows:
 Control rods 2, 4, 6, 8 at 47.0 inches.
 Control rods 1, 3, 5 at 27.5 inches.
 Control rods 7 and 9 at 28.0 inches.
 Reactor Steam flow 60,000 lb/hr.
 Generator output 2.0 Mw.
2. The center control rod was run in to 15.0 inches and the bypass valve closed. P_0 was changed from 640 to 650. At 9:32 A. M. the system was placed on automatic rod control with a $\pm 0.5\%$ pressure dead band for two minutes. In this time interval the control rod, flux, and pressure went through two cycles of oscillation with diverging amplitude.

3. The dead band was increased to $\pm 5\%$ and the system was placed on automatic. Increasing the dead band increased the amplitude and period of the divergent oscillation.
4. P_0 was reset to 600 and the center rod was withdrawn to 21.4 inches. The generator was delivering 1950 kw. The bypass valve position was 10%, and reactor pressure was 570 psi. The system was placed on automatic rod control with a $\pm 0.5\%$ dead band, and generator output was increased to 3800 kw at 200 kw/min. The control rod was withdrawn automatically in 1.8 inch increments to 30.1 inches. The minimum bypass valve position was 10%.
5. The generator load was decreased from 3800 kw to 2000 kw at 200 kw/min. The bypass valve opened to 18% before any control rod correction signals were made.

The automatic rod control with bypass was stable and did not exhibit any tendency to oscillate.

SECTION III RADIATION AND FLUX

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
7	Plant Radiation Levels, C. B. Zitek	163
25, 25A	EBWR Flux Plots, R. E. Bailey.	169
25B	First Annual Core Inspection, C. V. Pearson.	173
34	Prediction of N ¹⁶ Gamma Activity, A. Gibson, N. A. Kershaw	180
34A	Distribution of N ¹⁶ in EBWR, M. H. Theys, R. L. Mittl	191
35	Flux Distribution in Tangential Instrument Tube, A. Hirsch, W. Lipinski	201
47	Neutron Survey Beneath EBWR, S. Meyers	201
49	Gamma Flux Determination of EBWR Core, C. A. Pesce, J. Pohlman	205
49A	Measurement of Gamma Flux in EBWR, C. A. Pesce, J. Pohlman	211
56	Radioactive Crud Cleanup in Nozzles below EBWR Reactor, E. A. Wimunc	219
71	Determination of Thermal Neutron Flux in EBWR by Cobalt Activation, R. E. Bailey, R. F. S. Robertson . .	221
102, 102A, 102B	Relative Effectiveness of Sb-Be Source and Core- Induced Neutrons, A. Hirsch.	227

PLANT RADIATION LEVELS

(Test Report No. 7, by C. B. Zitek)

Most of the previous radiation level data of the components in the EBWR plant were taken during full power operation (20 Mw). This reactor experiment was made in part to obtain radiation levels and air and air-gas mixture activities at various reactor power levels.

Reactor Operation Data. The reactor power level was reduced from 20 Mw (60,000 lbs/hr steam flow) to 3.3 Mw (10,000 lbs/hr) in decreasing steps of 10,000 lbs/hr at one hour intervals. Other operating conditions which must be taken into consideration when comparing or predicting radiation levels in the plant are:

1. Turbine on turning gear - all steam by-passed to the condenser.
2. Air ejector gases discharged directly to the stack, no charcoal trap or recombiner in the system
3. Both inlet and outlet ventilation valves open.
4. Reactor water conditions:
 - a. pH = 7.0 - 7.2
 - b. Conductivity = 0.6 micromho
 - c. No gas or chemical addition to reactor feedwater.
5. No. 2 reactor water purification system in service (3 to 4 gpm).

Although some of the above conditions do not appear to have any connection with radiation levels or air activity in the plant, it is an established fact that they do.

Data and Sample Description. The data obtained and samples taken during the experiment were:

1. Data from sixteen plant monitors (ionization chambers) being continuously recorded on the radiation monitor recorder in the control room.
2. Six locations being locally monitored (three of these locations were also covered by the recording monitors for comparison).
3. Three air samples (both beta and gamma activity counted):
 - a. Radioactive particulate matter filtered from air sample taken on turbine floor.
 - b. Continuous air sample monitor recorder (Esterline-Angus) on turbine floor in plant.
 - c. Total air sample, two liter sample taken from air being discharged out the stack in plant attic.

4. Three continuous gamma activity monitors (gamma pulse height analyzers):
 - a. Stack gas in plant attic. Discriminator set at 72.
 - b. Fission gas monitor, air ejector gases in aftercooler, 1 - .51 Mev and 1 - 8.0 Mev.

The local monitoring and four of the six air and gas sample activity data were taken by the Radiation Safety group. All of the data given is in tables and shown graphically (Tables I-II), (Figs. 7-1 through 7-5).

Summary - Radiation Levels. The radiation levels at the various components of the plant checked very closely with the values obtained previously at a 20 Mw power level. Practically all of the values approached zero asymptotically with power level. The three locally monitored values checked the plant monitor values quite well as the power level was decreased.

Only two areas covered by this survey (shown on Fig. 7-4) did not show the general decrease with reactor power level. The start-up heater and the 3 in. pipe elbow on the line leading to the startup heater and reactor water purification system contain reactor water and the activity level would depend on the rate at which the radioactive reactor water reached these points. The only reason for the increase in radiation level of the startup heater with decrease in reactor power would be an increase in the natural circulation in this outside leg. Apparently a change in circulation rate or reactor water flow pattern takes place as the reactor power decreases permitting an increase in natural circulation in the side loop.

Air and Gas Activities. The three curves of the gamma activity monitors of air-gas mixtures (Fig. 7-5) showed the expected decrease in level with decreased power. However, when plotted on semi-log paper (Fig. 7-1), they did not yield straight lines which indicates a mixture of radioactive isotopes rather than a single isotope. This is in accord with the knowledge we have of the air ejector gases. The primary activity is N^{16} with smaller amounts of fission gases ($Xe^{135,138}$, $Kr^{88,89,90}$ and A^{41}).

It must be remembered that the air and air-gas activities reported here are very low compared to previous operating data due to the fact that the air ejector gases were being discharged directly to the stack during this period of operation.

The three curves of particulate matter in air and particulate plus air-gas mixture were unexpected but can be readily explained. Both beta and gamma activities are included and the daughters of the fission gases Xe and Kr are beta emitters. The fission gases released during the 20 Mw run had time to decay but the particulate matter had not been swept out of the building by the time the 16.6 Mw and 13.3 Mw run samples were taken.

TABLE I - EBWR Plant Radiation Levels

Reactor power level				Plant Radiation Monitors													
Date	Time	Steam flow, lb/hr	Mw	No. 1 turbine surface	No. 2 main floor area	No. 3 steam dryer	No. 4 air ejector	No. 5 condenser hotwell	No. 6 feedwater filter	No. 7 control rod room	No. 10 IX column No. 2	No. 11 prefilter No. 2	No. 12 plant air exhaust duct	No. 13 shield vent filter	No. 14 retention tanks	No. 15 shield cooling water pump	No. 16 vapor recovery system
4-11-57	1700	60,000	20	3	1	120	360	36	11	13	426*	420*	3	18	3	3	3
	1800	50,000	16.6	3	1	80	280	28	11	12.5	425	430	3	15	3.5	3.5	3
	1900	40,000	13.3	3	1	40	170	16	10	11	440	470	3	12	3.5	3.5	3
	2000	30,000	10	3	1	30	90	9	5.5	8.5	450	510	3	9	3.5	3.5	3
	2100	20,000	6.6	2.5	1	13	51	7.5	4	8	465	520	1	6	3	3	2.5
	2200	10,000	3.3	2	1	3	11	6	3	7	470	520	1	3	1	2	2
	2400	60,000	20	4	1	120	380	37.5	12	13	400**	400**	2	16	3	3	3
1-12-57	0600	60,000	20	not connected	2	130	330	39	6	12	1150 [†]	1800 [‡]	2	75	4	4.5	not connected

Reactor power level				Locally monitored [‡]					
	Time	Steam flow, lb/hr	Mw	Steam dryer	Air ejector (after-cooler)	Startup heater	Steam line (4 ft from steam dryer)	Condenser hotwell	3 in. pipe elbow to startup heater
4-11-57	1700	60,000	20	130	400	140	45	40	205
	1800	50,000	16.6	105	320	155	32	22	190
	1900	40,000	13.3	60	210	180	19	10	170
	2000	30,000	10	27	120	190	11	6	150
	2100	20,000	6.6	12	50	190	6	2	120
	2200	10,000	3.3	3.5	13	160	2	1.5	90
1-12-57	0600	60,000	20	100	450	250	-	30 to 50	400 [†]

NOTE - All data expressed in mr/hr - gamma radiation.

*Reactor Water Purification System No. 2 on at about 3 to 4 gpm

**Reactor Water Purification System No. 2 turned off at 2205

[†] Reactor Water Purification System No. 1 being used to cleanup reactor water after the boric acid poison period of operation without continuous cleanup operation[‡] SR June Survey Meter used, No. 243.

Reactor Water Conditions - pH 7.0 - 7.2 Conductivity 0.6 micromho, no gas or chemical being added to reactor feedwater

Turbine not operating - all steam by-passed to the condenser.

Air ejector gases discharged directly to the stack.

TABLE II - EBWR Plant Air and Air Ejector Gas Activity (4-11-57)

Time	Reactor power level		Air sample (particulate) turbine floor, d/m/M ³ *	Stack air count, c/m**	Stack gas sample, c/m†	Fission gas Monitors air ejector gases		Continuous air monitor instruments† (Esterline-Angus recorder),		
	Steam flow, lb/hr	Mw				Gamma energy up through 0.51 Mev, c/m	Gamma energy up through 8 Mev, c/m*	Reading	Scale	c/m
1700	60,000	20	4670	-	2888	3131	7339	0.6 to 0.7	2K	1300
1800	50,000	16.6	5500	9,274	3859	1839	5304	0.7 to 0.8	2K	1500
1900	40,000	13.3	4955	5,932	2382	798	3508	0.8 to 0.9	2K	1700
2000	30,000	10	4167	3,211	949	364	2054	0.8 to 0.9	2K	1700
2100	20,000	6.6	3368	1,916	517	63	1030	0.7 to 0.8	2K	1500
2200	10,000	3.3	2660	924	191	12	346	0.6 to 0.7	2K	1300
2400	60,000	20		12,652		3136	7170	0.5 to 0.6	2K	1100

*, **, and † Data taken by Radiation Safety (Mr. R. Swanson).

*Particulate activity in plant air - 10 cubic meter sample taken during 20 minute period - counted after 4 minutes.

**Gamma activity with sodium iodide crystal in stack in attic.

† Two liter gas sampler with G-M counter tube.

‡ Located near turbine on turbine floor (4) - Instrument measures particulate activity in air. Air flow rate of 10 m³ per hour.

It would be advisable to repeat this phase of the activity survey to confirm this data. In addition to sampling during a similar run, it would be well to reverse the order and come up in reactor power level for comparison and verification of the explanation.

Another run should be made with the turbine operating at various (at least two or three) power levels with and without the air ejectors being discharged to the stack to obtain pessimistic air activity conditions for record.

Fig. 7-1. Air and air ejector gas activity.

Fig. 7-2. Plant activities - from plant monitors.

Fig. 7-3. Plant activities - from plant monitors.

Fig. 7-4. Locally monitored plant activities.

Fig. 7-5. Air and air ejector gas activity.

EBWR FLUX PLOTS

(Test Report 25, 25A, by R. E. Bailey)

Flux Plot along Fuel Element at Core Index Location No. 129-370.

A radiation traverse along the above mentioned fuel element was obtained using chemical dosimetry. This fuel element is one of the four center elements and is thin and enriched.

The solution used in this measurement was 0.8 N FeSO_4 . Plastic capsules, 2.25 in. long, were filled with the solution, stacked one atop the other, and taped to a pipe. This in turn was lowered into a hollow aluminum tube ($\frac{1}{8}$ in. wall) and brought to the fuel element from the element's side. This assured an equal irradiation time for all of the capsules. The capsules were held adjacent to the element for twelve minutes. They were then emptied and the contents were examined for ferric ion concentration by using ultraviolet light with a Beckman DU spectrometer. The wave length was 302μ and the slit width was 0.4 mm. The measurements were taken at 25°C .

The results are plotted in Fig. 25-1. It can be seen that the peak flux is 19 in. from the bottom of the fuel plate and that the radiation level at this point on the plate is approximately 120,000 R/hr. $\phi_{\text{MAX}}/\phi_{\text{AVE}}$ is 2.04. This agrees with the value 2.17 which was determined in loading No. 46.¹

Fig. 25-1

Fuel element radiation
traverse

It should be noted that this type of flux plot is a summation of all flux distributions undergone by this plate up to the time these measurements were taken.

Northwest Quadrant Flux Plot for Core Loading No. 46. A flux plot of the northwest quadrant of the EBWR core was obtained by using chemical dosimetry. The main assumption here is that the gamma flux is proportional to the neutron flux and the mass of U^{235} present.

¹ Thie, J. A., "EBWR Physics Experiments" ANL-5711, July, 1957, P. 15.

The dosimeter used in this determination was 0.8 N FeSO_4 . This solution was put in small plastic vials which were about 2 in. long. Upon exposure to gamma energy the ferrous ion is converted to the ferric ion at a rate dependent on the intensity of the gamma field. The resultant ferric ion is then analysed by measuring the fluorescence of the solution.

The procedure used for these measurements was to flood the reactor vessel and pull an element out of the core and up to within about 6 ft of the top of the indexing plug. A hollow aluminum tube with the bottom sealed was then lowered down next to the fuel plates until the bottom of the tube and the bottom of the plates coincided. After this, a pole with the 2 in. vials of solution was lowered into the hollow aluminum tube for a given exposure. Three different exposure times were used over the length of the element so that the ferric ion would be in the optimum concentration range for analysis. The length of exposure time is a function of the gamma field and, therefore, several trial runs had to be made initially so that the exposure times would be correct. On the most center element the exposure was as listed below:

<u>In. up from bottom of plate</u>	<u>Exposure time (min.)</u>
0-30 9	4
30 9-38 4	5
39.4-45 8	15

A background reading was taken with all the elements in the core. The background was found to be negligible.

The results were graphed and each curve was integrated. The areas under the curve were then normalized to an enriched thin element since there are different amounts of U^{235} in each of the three types of elements studied. There is 1.4 times as much U^{235} in an enriched heavy plate. This gave the energy produced per element for each element in the quadrant assuming that this quadrant produced 5 Mw of heat at 20 Mw (Table I). The fuel element positions are shown in Fig. 25-2. The max/average was also obtained for each element as well as for the core. These values are 1.70 and 3.39 respectively. Assuming that the average power per element is 178.5 kw, the estimated maximum heat flux in the northwest quadrant is 146,458 Btu/hr ft² when the reactor is operating at 20 Mw.

A radial flux plot was made by going from the center of the core to the northwest. Four curves have been drawn with the parameter "inches from bottom of plate." Fig. 25-3. Also included is an axial plot along the most radioactive element. Fig. 25-4.

It is interesting to note that the area under the curve for the element in position 129-370 is 2.31 when compared to a similar plot obtained previously and discussed above in this report. The ratio of the

Table I - Data Concerning the N.W. Quadrant of the EBWR Core.
Loading #46

Fuel element position	Type of element	% of contribution to total power generated in quad.	<u>Max.</u> <u>Avg.</u>	Activity level at 16" from the bottom of the plate, rad/min
129-370	ET	5.35	1.66	3815
109-311	ET	5.94	1.73	4284
109-254	ET	5.80	1.69	4162
95-219	ET	6.09	1.78	4631
95-294	ET	6.31	1.76	4753
101-266	ET	7.01	1.75	5052
77-276	ET	5.01	1.73	3740
77-191	ET	3.38	1.65	2421
89-233	ET	6.61	1.74	4937
89-264	ET	4.79	1.74	3502
59-259	EH	2.46	1.69	2509
59-170	EH	2.73	1.66	2727
72-202	ET	4.46	1.79	3434
80-237	ET	3.69	1.73	2747
72-254	ET	5.34	1.60	3692
34-237	EH	1.53	1.64	1489
34-145	EH	1.53	1.58	1435
54-179	EH	1.76	1.60	1686
54-241	EH	1.92	1.64	1877
63-231	ET	3.26	1.70	2349
63-206	EH	2.14	1.66	2162
27-149	H	1.84	1.67	892
44-181	EH	1.83	1.74	1938
44-220	EH	2.00	1.71	2071
47-203	EH	1.88	1.76	2009
23-188	EH	1.33	1.93	1493
23-172	EH	1.67	1.61	1615
27-219	ET	2.34	1.64	1649
Average			1.70

Fig. 25-2

Location of fuel elements in
northwest quadrant

Mwd exposure for each case is 2.23. The first curve was obtained 35 days after shutdown while the second curve was obtained 14 days after shutdown. If the first curve had been obtained 14 days after shutdown the area would have been greater and the ratio of the two curves would have been less than 2.31 and closer to 2.23.

Fig. 25-3. Radial flux plot center of core to northwest.

Fig. 25-4. Flux plot along enriched thin plate in position 101-256.

In summary, the curves obtained show the nvt as a function of position on the fuel element and not nv. No correction was made for the decay over the four days required for the determination. It is thought that the error introduced by not checking decay is only of several percent in light of the preceding paragraph.

RESULTS OF THE FIRST ANNUAL INSPECTION OF THE EBWR CORE

(Test Report No. 25B, by C. V. Pearson)

On December 24, 1957, the Experimental Boiling Water Reactor at the Argonne National Laboratory was shut down prior to its first annual inspection. The 20 Mw reactor had generated 3060 Mwd of thermal power (out of a possible 7300 Mwd for continuous 20 Mw operation). This report outlines the observations and presents an empirical interpretation of the findings during inspection of the core and vessel internals. No attempt was made to interpret and present theoretical postulates.

Core Description. The core loading for the EBWR during the year 1957, designated Core Loading 46, is shown in Fig. 25B-1.

Fig. 25B-1. Core loading #46.

The heterogeneous boiling water core consisted of 114 fuel elements, employing four different types of fuel plate and enrichment arrangements. The center portion of the core consisted of 54 (1.44 percent) enriched thin fuel elements with 2 natural thin fuel elements in the very center to act as flux depressors. Surrounding this central section were 50 (1.44 percent)

enriched thick elements placed in a concentric manner around the enriched thin elements. The circular four foot diameter core geometry was completed by the addition of 6 natural thick fuel plate elements, and 2 additional enriched thin fuel plate elements. The neutron source for the reactor was an antimony-beryllium source located at the outside of the four foot diameter core.

Nine control rods are used in the reactor. The 5 central control rods are hafnium-Zircaloy 2. The 4 outer control rods are 2 percent boron-304 stainless steel.

Test Program. A chart showing an integrated plot of the reactor operations for the 12 month period ending December, 1957, is shown in Fig. 25B-2.

Fig. 25B-2. Power level for 1957.

During this period the program of test and operation were as follows:

1. January - Reactivity tests and measurements using boric acid.
2. February, March, and April - Long and short period critical experiments were conducted at various pressures and control rod settings.
3. May and June - Power level experiments at various pressures and power generating conditions.
4. June 21 to August 12 - The reactor was shut down - plant alterations and repairs were made.

5. August - Low pressure power experiments were conducted during the remainder of the month.
6. September and October - Neutron flux measurements at various pressures and power levels were made. During the last part of October the reactor vessel was opened and new recirculation rate thermocouples installed in the core.
7. November - Thermodynamic and sodium activity experiments were conducted.
8. December - Control rod oscillation experiments were conducted to determine reactor stability at various power levels and pressures, prior to higher power operation. The reactor was then operated at full power. It was found that the reactor was capable of delivering 50 Mw for a short term power level run and 34 Mw was found to be the maximum long term operating level with xenon equilibrium.

During the year's operation the average banked control rod setting for the 48 in. high core was approximately 30 in. However, during the first 1000 Mwd (i.e., January to June) the reactor was operated with the 4 boron stainless control rods set at 48 in. and the 5 hafnium-Zircaloy control rods run in to a setting of approximately 28 in. at stable operation of 20 Mw. This was done in an attempt to prolong boron burnup in the outer 4 control rods. From August to December (the remaining 2000 Mwd) the control rods were for the most part run in a bank at the average setting of 30 in.

Power Distribution in the Core. One of the most rewarding areas of investigation during this annual inspection was that conducted by R. E. Bailey and J. Farris. The investigation consisted of a series of ferrous sulfate, chemical dosimetry measurements of the fission product decay distribution along the length of each fuel element in the northwest quadrant of the core. Lack of time prevented a complete examination of the entire core. However, since the core is essentially symmetrical about the vertical axis, with the exception of the 2 flux depressing elements in the center, a fairly accurate analysis of the core can be made. This report deals primarily with the findings and the subsequent extrapolated proportional distributions of the thermal neutron flux, thermodynamic heat flux, and power distribution in the northwest quadrant.

Extrapolation of the fission product decay activity distribution, as measured by the chemical dosimetry survey, to power generated in the northwest quadrant permits one to determine the effective power distribution. Figure 25B-3 shows an isobar plot of percent power contributed by each fuel element in the northwest quadrant. From this isobar plot of power distribution it can be seen that the hottest elements are not in the center of the core but are located some 20 to 25 cm radially out from the center. Two factors seem to combine to produce this effect. The one factor is the 2 flux depressing elements adjacent to the central elements; the other factor is the control

By equating this fission product activity to thermo neutron flux, a comparison between the initial computed thermo flux patterns and that obtained from the dosimetry was made. Figures 25B-5 and 25B-6 show the comparison of the computed and measured data along the length of several fuel elements in the radial and axial directions.

Fig. 25B-5. Maximum thermal flux and percent steam void in the axial direction.

Fig. 25B-6. Thermal flux and percent steam void in the radial direction

The arbitrary values used in plotting these figures were based upon an average unit of thermal flux as determined by integrating over the entire quadrant, and then selecting an average value that agreed with comparable total net power as determined by the theoretical cosine distribution for the initial computations. Thus the abscissa of the axial thermal neutron distribution plot was a convenient scale for relating arbitrary units of thermal neutron flux.

Since the initial computed data was intended primarily for determining a maximum flux and also since the actual heterogeneous core had a varied nonsymmetric distribution, Fig. 25B-5 merely attempts to compare the distribution pattern of the hottest element at the core centerline, and at points 25 cm and 36 cm from the core centerline, with the maximum theoretical distribution. The close geometric agreement is most gratifying.

A graphic plot of the dosimetry measurements produced an accurate pattern of the fission product distribution in each fuel element. By equating this fission product activity of the thermal neutron flux, a comparison between the initial computed thermal flux pattern and that obtained from the dosimetry was attempted. Figures 25B-5 and 25B-6 show the correlation between the calculated thermal neutron flux and the measured fission product distribution in the axial and radial directions. Because of the complexity of calculating the thermal flux for a heterogeneous reactor core containing 4 types of fuel elements plus control rods and control rod channels, less refined calculations were used in calculating the theoretical thermal neutron flux in the EBWR core. Consequently, what Figs. 25B-5 and 25B-6 actually show is a comparison between the measured case as compared to the simplified calculations for a heterogeneous reactor without control rods, and control rod channels, using an approximate water to uranium volume ratio of 3.5 to 1 at room temperature. The fuel mixture used in this initial simplified case was an assumed uniform mixture of 18 natural thin, 18 natural thick, 38 enriched (1.44%) U^{235} thin, and 38 enriched thick elements, having a total of 112 elements in a four foot diameter right circular core. This being the case, the close correlation between the theoretical and the actual cases as shown in Fig. 25B-5 take on even greater significance for the less refined engineering calculations.

Further graphical analysis of the data obtained by dosimetry during this first annual inspection revealed the interesting pattern shown in Fig. 25B-7. This figure attempts graphically to present the data as measured. The isobar flux distribution of a vertical plane and the northwest quadrant running from the center of the core out to the northwest direction, reveals the hottest portions of the core. From this information it would appear that a 3 dimensional concept of power distribution in the flux depressed center type right cylinder core is not spherical but instead forms a torus type of distribution, with the hottest or the highest powers being generated some 8 in. radially out from the core and 15 in. from the bottom of the core.

Fig. 25B-7. Flux distribution in northwest vertical plane.

Fuel Plate Deposit. Initial examination of the fuel element discoloration during the dosimetry indicated multicolored hues ranging from brown through blues and greens to a bright yellow. These colors and the change in hues noted during the examination cause the observer to speculate as to where bulk boiling began within the core and also as to the approximate zone of banking of the control rods. Figure 25B-7 shows the areas so measured, and compares the estimated zones with the actual zones as calculated by heat transfer calculations, and as noted in the normal control rod settings indicated by the reactor operating log.

An attempt to determine the source of discoloration of the Zircaloy-2 fuel plates revealed that a tenacious deposit had affixed itself to the plates. X-ray defraction analysis of samples of this deposit indicated a very high aluminum content. Table I charts the data obtained in this analysis by volume percentage. Further examination of the reactor core and reactor plant equipment would indicate that the primary sources of contamination are as follows:

Table I - Fuel Plate Deposit
(In Volume Percentage)

Pb	trace	Ca	0.48
Zr	trace	Cu	0.82
Sn	trace	B	1.62
Mn	0.15	Fe	4.67
Zn	0.21	Ni	10.3
Cr	0.26	Al	81.7

(1) aluminum from the (M388) aluminum dummies in the core; (2) nickel contamination from the Kanigen coating of the steam lines connecting the reactor, the steam dryer, and the turbine (which upon later inspection were found to be badly eroded); (3) the stainless steel and aluminum alloys found throughout the reactor coolant system; and (4) and iron contamination from the mild steel condenser walls,

the stainless steel pressure vessel, and core components.

Samples of crud taken from fuel element walls revealed it to exist in thicknesses from 0.003 in. to 0.005 in.

Conclusions. Investigation during the first annual inspection of the EBWR core revealed no adverse or detrimental conditions which would necessitate further investigation to give complete confidence in future performance.

PREDICTION OF N^{16} GAMMA ACTIVITY IN THE EXPERIMENTAL BOILING WATER REACTOR

(Test Report No. 34, by A. Gibson and N. A. Kershaw)

Calculations prior to this report have been made of N^{16} gamma activities by Brush¹ and Zeitlin² with respect to the EBWR and ARBOR projects. These investigations have yielded results that have been used in the design of shielding for the particular units under consideration, but in the case of EBWR have not given results that can be correlated with actual measured operating values. Therefore, an effort has been made to obtain a method that would give correlation between calculated values and those attained in actual reactor operation.

Procedure. The procedure followed in this study consisted of comparing the various calculated activities with the measured activity at one point in the system. For simplification, the point selected was in the steam pipe leaving the reactor, and upstream from the steam dryer. (See Fig. 34-1.)

¹ H. Brush, Private Communication.

² H. R. Zeitlin, Private Communication.

Fig. 34-1. Test schematic.

Two calibrated instruments were used for making the actual activity measurements. Up to 30 Mw reactor load, a portable high range Juno survey meter (AEC Model No. SLC-170) ionization chamber was used. To reduce the effect of scattered gamma activity, the instrument was shielded with lead. Above 30 Mw, a nonportable, remote reading gamma-sensitive ionization chamber was used. The above two instruments were calibrated against each other using the Juno as the standard.

Methods. Four different methods of calculating the activity were used. Two methods were similar to those used by Brush in the original EBWR N^{16} predictions, and by Zeitlin in the preliminary ARBOR N^{16} investigation. The four calculation procedures are defined as follows:

1. Sweep, perfect-mixing (S-PM):

This type of calculation assumes that all of the N^{16} generated in the core is swept out of the water, and leaves the core with the steam. "Perfect-mixing" implies that in the steam dome of the reactor vessel, the N^{16} diffuses completely and reaches an average activity before discharging to the steam line.

2. Sweep, slug flow (S-S):

This type of calculation is identical with 1 above, except that in passing through the reactor steam dome, the steam is assumed to decay by a factor of $e^{-\lambda t}$.

3. Non-sweep, perfect-mixing (NS-PM):

The assumption in this calculation is that the activated N^{16} retains the same molecular form which it had prior to activation; that is, one pound of steam carries off an amount of N^{16} activity equal to the amount contained in one pound of reactor water. "Perfect-mixing" is defined the same as in 1 above.

4. Non-sweep, slug flow (NS-S):

Same as 3 above for NS, and same as 2 above for S.

Discussion. Prior to completion of the tests, it was believed that the actual mechanics of the N^{16} carryover would fall somewhere between the "non-sweep," and "sweep" cases so described above, with the closest approximation being the "non-sweep" case. Examination of Fig. 34-2 indicates that this is actually the case. No attempt has been made in this report to determine the actual mechanism involved in N^{16} carryover. It is believed, however, that sufficient evidence has been accumulated to warrant an N^{16} prediction on the basis of a "non-sweep" - "perfect-mixing" calculation multiplied by a constant. In the case of EBWR, examination of Fig. 34-1 will indicate that the constant multiplier is 3.1.

Fig. 34-2. Activity levels.

A check of the sample calculation (see calculations) and Fig. 34-3 which was compiled from data by Paul Lottes, indicated that it was necessary to determine various parameters such as the average fast virgin flux, recirculation ratio, exit voids, etc. Exclusive of the flux, fairly large errors can be tolerated in these parameters without materially affecting the results. The average fast virgin flux is not known accurately and may be as much as 40 percent low; this will affect the results as a straight multiplying factor.

The activation cross section for the $O^{16} - N^{16}$ reaction averaged over the entire fission spectrum is 0.020 millibarns. This result was obtained from page 89 of the Rockwell Shielding Design Manual (TID-7004) dated March, 1956, and was used in all calculations.

Fig. 34-3. Test parameters in the reactor.

According to the Shielding Manual (page 89), approximately 82 percent of the N^{16} was accompanied by a high-energy gamma, with 6 and 7 Mev gammas being released in the ratio of 12.5 to 1. To simplify the calculations, the N^{16} was assumed to yield 80 percent gammas with an energy of 6.2 Mev. In addition to the N^{16} gamma activity, beta particles are released from N^{16} disintegrations; and there is an O^{17} (n,p) N^{17} reaction which yields a 3.7 Mev beta, and a 1-Mev neutron. The natural abundance of O^{17} is 0.039 percent.

Since the beta particles are absorbed by the piping, and since the gamma sensitive counters used are inefficient neutron counters, the error introduced by sensitive counters not considering the above effects is small. There are also some fission product gases in the steam emitting high energy gammas, but again, the total quantities are very small compared with the N^{16} activity.

No attempt was made in this report to evaluate the affect of various changes to the reactor water treatment on N^{16} evolution. It is realized that by the addition of, say, hydrogen, the N^{16} activity is increased. During operation, the water conductivity was maintained at less than 0.5 micromhos, and the effluent from the condenser was deaerated to less than 0.005 cc/liter. The pH was maintained at 7.0.

Radiation dose rates as a function of geometry, distance, and shielding material and thicknesses were determined by methods described in the Reactor Shielding Design Manual for cylindrical sources with self-absorption.

An attempt was made to obtain correlation between reactor pressure, and N^{16} activity, but the measured data was too incomplete to justify inclusion in this report.

Conclusions. This investigation has indicated a successful correlation between calculated and measured N^{16} activities. This correlation occurs for the "non-sweep, perfect-mixing" calculation procedure. In "non-sweep, perfect-mixing" it is assumed that the activated N^{16} retains the same molecular form which it had prior to activation (e.g., H_2O H_2N); that is, one pound of steam carries off an amount of N^{16} activity equal to the amount contained in one pound of reactor water. It is further assumed that in the steam dome of the reactor vessel, the N^{16} diffuses completely and reaches an average activity level before being discharged to the steam line. Figure 34-1 shows graphically the steam line measured activity, and calculated activity by the "non-sweep, perfect-mixing" calculation multiplied by a constant. Due to the nature of the calculations, it is believed that the multiplying factor of 3.1 is applicable to other boiling water reactors of the same general arrangement as EBWR; however, some suitable safety factor should be included to allow for the uncertainties discussed.

Figure 34-4 shows graphically the results of all methods of calculations performed, and shows that the results do not correlate very well with the measured curve, with the exception of the "non-sweep, perfect-mixing" case.

Fig. 34-4. Calculation data.

Sample Calculations - 20 Mw, 600 psig Operation.

ϕ = Average fast virgin flux $\sim 5 \times 10^{12}$ neutrons/cm² - sec

$\sigma_{O^{16}}$ = Activation cross section averaged over fission spectrum
(Rockwell Shielding Manual, page 89) = 20×10^{-30} cm²

$$n = \text{Decay const. } N^{16} \frac{0.693}{t^{\frac{1}{2}}} = \frac{0.693}{7.35} = 0.0943 \text{ sec}^{-1}$$

N^{16} decays with $T_{\frac{1}{2}} = 7.35$ sec; 80 percent decays emitting 6.2 Mev γ s.

Coolant flow area in fuel zone (exclusive of dummy elements) = 8.43 ft².

Coolant inlet velocity in fuel cluster 2.5 ft/sec (ANL 5607)

Average voids at core exit = 28.5% (ANL 5607)

Velocity of water at core exit = $2.5 + (0.285 \times 2.5) = 3.2$ ft/sec

Steam to water velocity ratio = 1.5 (ANL 5607)

Area of chimney/area of fuel $\cong 1.4$

$\therefore$ Assume greater area of chimney exactly compensates for slips,
and velocity in chimney = core exit velocity.

Recirculation ratio = 63/1 (ANL 5607)

Recirculation flow $\cong \frac{63 \times 60,000}{500 \times 0.8} = 9500$ gpm

Downcomer flow area = 23.0 ft² (ANL 5607)

Velocity in downcomer = 0.9 ft/sec

Fuel height = 4.5' (ANL 5607)

Chimney height = 4.42'

Total height = 8.92'

Average core velocity = $\frac{2.5 + 3.2}{2} = 2.8$ ft/sec

Time in core = $4.5/2.8 = 1.61$ sec = t_1

Time in chimney = $4.42/3.2 = 1.38$ sec

Time in downcomer = $8.92/0.9 = 9.92$ sec

Radial flow in and out of core = 2.00 sec (Est.)

Complete cycle = 14.91 sec = t_2

Saturation activity = A_{∞} .

For one trip through the core, the activity at the outlet is $A_{\infty}(1 - e^{-\lambda T_1})$
where T_1 is the time in the active zone. At some time T_2 later, the activity
has decayed to $A_{\infty}(1 - e^{-\lambda T_1})e^{-\lambda T_2}$. For n cycles, $A = A_{\infty}(1 - e^{-\lambda T_1})$

$(1 + e^{-\lambda T_2} + e^{-2\lambda T_2} + e^{-n\lambda T_2})$ where T_2 = time for 1 complete cycle.

$$A'_{\infty} = \phi \Sigma^{O16} \text{ disintegrations/sec-cm}^3. \quad At = At \text{ weight in gm.}$$

$$A_{\infty} = \phi N \sigma_{O16} = \frac{\phi \left(\frac{\text{dis}}{\text{sec-cm}^2} \right) \times 6.02 \times 10^{23} (\text{atoms}) \times \rho \left(\frac{\text{gm}}{\text{cm}^3} \right) \times \sigma_{O16} \left(\frac{\text{cm}^2}{\text{atom}} \right)}{At(\text{gm}) \times \rho'(\text{lb/cm}^3)}$$

$$\frac{\rho(\text{gm/cm}^3)}{\rho'(\text{lb/cm}^3)} = C \left(\frac{\text{gm}}{\text{lb}} \right) = 454$$

$$A_{\infty} = \frac{\phi \times 6.02 \times 10^{23} \times C \times \sigma_{O16}}{At} \left(\frac{\text{disintegrations}}{\text{sec-lb}} \right)$$

$$A_{\infty} = \frac{5 \times 10^{12} \times 6.02 \times 10^{23} \times 454 \times 20 \times 10^{-30}}{18} = 15.2 \times 10^8 \text{ dis/sec-lb}$$

$$1 - e^{-\lambda T_1} = 1 - e^{-(0.0943)(1.61)} = 0.141$$

$$1 + e^{-\lambda T_2} + e^{-n\lambda T_2} = 1 + e^{-(0.0943)(14.9)} + e^{-2(0.0943)(14.9)} = 1.303$$

1. Consider four cases:

Case 1 - Sweep-slug flow (S-S) - All N^{16} is swept out with the steam and assumes the N^{16} will decay by a factor of $e^{-\lambda T}$ in the steam dome.

Case 2 - Sweep-perfect mixing (S-PM) - All N^{16} is swept out with the steam and assumes the N^{16} mixes perfectly and attains an average activity before leaving the reactor vessel.

Case 3 - Non-sweep-slug flow (NS-S) N^{16} retains the same molecular form which it had prior to activation, and assumes the N^{16} will decay by a factor of $e^{-\lambda T}$ in the steam dome.

Case 4 - Non-sweep-perfect mixing (NS-PM) N^{16} retains the same molecular form which it had prior to activation, and assumes the N^{16} mixes perfectly and attains an average activity before leaving the reactor vessel.

2. Case 1 - Sweep-slug flow (S-S)

Equilibrium activity in the steam at core outlet is 64 x activity per pass since recirculation ratio is 63/1.

$$A_{\text{Equil.}} = 64 \times A_{\infty} \times (1 - e^{-\lambda T_1}) \times 0.80 \text{ where the factor 0.70 is a fraction of 6.2 Mev } \gamma\text{'s.}$$

$$A_{\text{Equil.}} = 64 \times 1.52 \times 10^9 \times 0.141 \times 0.8 = 1.10 \times 10^{10} \text{ dis/sec-lb (of 6.2 Mev } \gamma\text{'s)}$$

Since water level is approx. 3 in. above top of chimney the time of transit from top of core to water level = 1.38 sec.

$$\begin{aligned}
 A_{\text{Equil. at water surface}} &= 1.10 \times 10^{10} \times e^{-(0.0943 \times 1.38)} \\
 &= 1.10 \times 10^{10} \times 0.878 \\
 &= 9.66 \times 10^9 \text{ dis/sec-lb of steam}
 \end{aligned}$$

By definition, the N^{16} will decay by a factor of $e^{-\lambda t}$, and assume N^{16} travels at an average velocity to the dry pipe, or velocity 0.34 ft/sec, and dry pipe is approx. 7.25 feet above water level in the reactor

$$\therefore t = 7.25/0.34 = 21.3 \text{ sec.}$$

$$e^{-\lambda t} = e^{-(0.0943 \times 21.3)} = 0.134$$

For sweep slug flow,

$$A_{\text{Equil. at dry pipe}} = (9.66 \times 10^9) \times 0.134 = 1.29 \times 10^9 \text{ dis/sec-lb.} \quad (\text{A})$$

3. Case 2 - Sweep-perfect mixing (S-PM)

$A_{\text{Equil. at water surface}}$ is the same as for case 1,
 $9.66 \times 10^9 \text{ dis/sec lb of steam}$

For perfect mixing:

$$A = A_1 e^{-\lambda t}$$

Average time through vessel

$$T_3 = \frac{V(\text{cm}^3)}{F_3(\text{cm}^3/\text{sec})}$$

$$V_3 \bar{A}_3 = \int_0^{T_3} A_2 e^{-\lambda t} F_3 dt = \left[\frac{-A_2 F_3 e^{-\lambda t}}{\lambda} \right]_0^{T_3} = \frac{-A_2 F_3 e^{-\lambda T_3}}{\lambda} + \frac{A_2 F_3}{\lambda}$$

$$\bar{A}_3 = \frac{A_2 F_3}{V_3} (1 - e^{-\lambda T_3}) \quad \text{For perfect mixing } \bar{A}_3 = A_4$$

4. Where: $\bar{A}_3$ = average activity in vessel

V_3 = volume of vessel, ft^3

λ = decay constant, sec^{-1} 0.0943

F_3 = volume flow rate, ft^3/sec

t_3 = time of transit through vessel, sec

A_2 = activity at entrance to vessel

A_4 = activity at exit from vessel

$$V = \text{volume of steam dome} = \frac{(7)^2}{4} \times 10 = 385 \text{ ft}^3$$

$$F_3 = 60,000 \text{ \#/hr} \times \frac{1}{3600 \text{ sec}} \times 0.744 \text{ ft}^3/\text{\#} = 12.4 \text{ ft}^3/\text{sec.}$$

$$1 - c^{-\lambda T_3} = 1 - e^{-(.0943 \times \frac{385}{12.4})} = 0.946$$

$$\frac{F_3}{W_3} (1 - e^{-\lambda T_3}) = \frac{12.4 \times 0.946}{0.0943 \times 385} = 0.323$$

For sweep perfect mixing:

$$A_{\text{Equil. at dry pipe}} = 0.323 \times 9.66 \times 10^9 = 3.12 \times 10^9 \text{ dis/sec-lb} \quad (\text{B})$$

5. Case 3 non-sweep slug flow (NS-S)

Equilibrium activity at core outlet is:

$$A_{\text{Equil.}} = A_c (1 - e^{-\lambda T_1}) (1 + c^{-\lambda T_2} + c^{-n\lambda T_2}) \times 0.80$$

$$A_{\text{Equil.}} = 1.52 \times 10^9 \times 0.141 \times 1.303 \times 0.8 = 22.3 \times 10^7 \text{ dis/sec lb of steam (of 6.2 Mev } \gamma\text{'s)}$$

$$A_{\text{Equil. at water surface}} = 22.3 \times 10^7 \times 0.878 = 19.6 \times 10^7 \text{ dis/sec-lb of steam or water}$$

Non-sweep slug flow,

$$A_{\text{Equil. at dry pipe}} = 19.6 \times 10^7 \times 0.134 = 2.63 \times 10^7 \text{ dis/sec-lb steam} \quad (\text{C})$$

6. Case 4 non-sweep, perfect mixing (NS-PM)

Calculations are the same as case 3 except for equation C

Non sweep, perfect mixing,

$$A_{\text{Equil. at dry pipe}} = 19.6 \times 10^7 \times 0.323 = 6.33 \times 10^7 \text{ dis/sec-lb steam} \quad (\text{D})$$

7. Summary of $A_{\text{Equil.}}$ at dry pipe for four cases considered

Case 1 (S-S)	$1.29 \times 10^9 \text{ dis/sec-lb steam}$
Case 2 (S-PM)	$3.12 \times 10^9 \text{ dis/sec-lb steam}$
Case 3 (NS-S)	$2.63 \times 10^7 \text{ dis/sec-lb steam}$
Case 4 (NS-PM)	$6.33 \times 10^7 \text{ dis/sec-lb steam}$

8. Time delay from dry pipe to monitoring test point

Velocity in 6" sch 40 pipe at 600 psig and 60 000 #/hr steam flow

$$V = \frac{144 \left(\frac{\text{in}^2}{\text{ft}^2} \right) \times 60\,000 \text{ (#/hr)} \times 0.744 \text{ (ft/#)}}{28.9 \text{ (in}^2\text{)} \times 3600 \text{ (sec/hr)}} = 61.8 \text{ ft/sec}$$

length of pipe to monitoring test point = 25 ft

$$t_s = \frac{25}{61.8} = 0.405 \text{ sec}$$

$$\text{Decay factor} = e^{-\lambda T_s} = e^{-(0.0045 \times 0.405)} = 0.963$$

9. Summary of A_{Equl} at monitoring test point

$$\text{Case 1 (S-S)} \quad 1.29 \times 10^4 \times 0.963 = 1.24 \times 10^4 \text{ dis/sec lb steam}$$

$$\text{Case 2 (S-PM)} \quad 3.12 \times 10^9 \times 0.963 = 3.00 \times 10^9 \text{ dis/sec lb steam}$$

$$\text{Case 3 (NS-S)} \quad 2.63 \times 10^7 \times 0.963 = 2.53 \times 10^7 \text{ dis/sec lb steam}$$

$$\text{Case 4 (NS-PM)} \quad 6.33 \times 10^7 \times 0.963 = 6.09 \times 10^7 \text{ dis/sec lb steam}$$

$$\begin{aligned} \text{Converting dis/sec lb steam} &= \frac{\text{photons}}{\text{sec lb}} \times \frac{1 \text{ lb}}{0.744 \text{ ft}^3} \times \frac{1 \text{ ft}^3}{2.832 \times 10^4 \text{ cm}^3} \\ &= 0.475 \times 10^{-4} \text{ photons/sec} \cdot \text{cm}^3 \text{ steam} \end{aligned}$$

10. Source strength SV

$$\text{Case 1 (S-S)} \quad 1.24 \times 10^4 \times 0.475 \times 10^{-4} = 5.88 \times 10^4 \text{ photons/sec cm}^3 \text{ steam}$$

$$\text{Case 2 (S-PM)} \quad 3.00 \times 10^9 \times 0.475 \times 10^{-4} = 14.23 \times 10^4 \text{ photons/sec cm}^3 \text{ steam}$$

$$\text{Case 3 (NS-S)} \quad 2.53 \times 10^7 \times 0.475 \times 10^{-4} = 12.0 \times 10^4 \text{ photons/sec cm}^3 \text{ steam}$$

$$\text{Case 4 (NS-PM)} \quad 6.09 \times 10^7 \times 0.475 \times 10^{-4} = 28.9 \times 10^4 \text{ photons/sec cm}^3 \text{ steam}$$

11. Calculation of attenuation of activity through steam piping and insulation

From Shielding Manual (TID 7004) page 360

$$\phi = \frac{BS_V R_0'}{2(a+Z)} F(\theta, b) \text{ (for cylindrical source with self shielding)}$$

12. Where: B = buildup factor

S_V = source strength, photons/sec cm^3

R_0 = radius, cm

a = distance to dose from R_0 , cm

Z = distance inside cylinder for equivalent line source, cm

$$F(\theta b_2) = \int e^{-b} \sec \theta' d\theta'$$

$$\theta = \frac{\pi}{2} \text{ for infinite pipe length}$$

ϕ = flux, photons/sec cm^2

$$b_2 = b, + \mu_s Z$$

$$b_1 = \sum \mu_i t_i$$

μ_i = macroscopic cross section of shielding, cm^{-1}

μ_s = macroscopic cross section of source, cm^{-1}

$$\frac{a}{R_0} = \frac{2.78}{3.033} = 0.917 \quad R_0^2 = (3.033 \times 2.54)^2 = 59.4 \text{ cm}$$

$$\frac{\mu_{\text{H}_2\text{O}}}{\rho} \text{ at 6 Mev} = 0.0273 \text{ cm}^2/\text{gm} \text{ (TID 7004 page 448)}$$

$$\rho_{\text{stm}} = \frac{1}{0.744} \text{ lb/ft}^3 \frac{0.01602}{0.744} \text{ gm/cm}^3$$

$$\mu_s = \frac{0.0273 \times 0.01602}{0.744} = 5.88 \times 10^{-4} \text{ cm}^{-1}$$

$$a + R_0 = (2.78 + 3.033) 2.54 = 14.8 \text{ cm}$$

$$\mu_s (a + R_0) = 5.88 \times 10^{-4} \times 14.8 = 8.70 \times 10^{-3} = 0.00870$$

$$m = 0.00227 \text{ (TID 7004, page 362)}$$

$$\frac{\mu}{\text{magnesium}} = 0.0267 \text{ cm}^2/\text{gm} \quad \mu_{\text{mg}} = 0.0267 \times 1.73 = 0.0463 \text{ cm}^{-1}$$

$$\mu_{\text{mg}} t_{\text{mg}} = 0.0463 \times 2.5 \times 2.54 = 0.294$$

$$\mu_{\text{steel}} t_{\text{steel}} = 0.0305 \times 7.83 \times 0.280 \times 2.54 = 0.170$$

$$b_1 = 0.294 + 0.170 = 0.464$$

$$\frac{1}{m} \mu_s Z = 1.635 \text{ (TID 7004 page 363)}$$

$$\mu_s Z = 1.635 \times 0.00228 = 0.003715$$

$$Z = \frac{0.00317}{5.88 \times 10^{-4}} = 6.30 \text{ cm}$$

$$b_2 = 0.464 + 0.0037 = 0.468$$

$$F(\theta b_2) = 0.655 \text{ (TID 7004, page 385)}$$

Assume B = 1

13. From calculations . 11 (see above)

$$\phi = \frac{BS R_0^2}{2(a+Z)} F(\theta b_2) = \frac{1 \times S_v 59.4}{2(2.78 \times 2.54 + 6.30)} (0.655)$$

$$= 1.455 S_v \frac{\text{photons}}{\text{sec cm}^2}$$

For 6 mev γ 's dose rate due to

$$1 \text{ photon/cm}^2 \text{ sec} = 7.25 \times 10^{-6} \text{ r/hr (TID 7004 page 19)}$$

$$= 7.25 \times 10^{-3} \text{ mr/hr}$$

$$\therefore \phi = 1.455 \times 7.25 \times 10^{-3} S_v = 10.55 \times 10^{-3} S_v \text{ mr/hr}$$

Dose rate, mr/hr at test monitoring point:

Case 1(S-S)	(10.55×10^{-3})	$(5.88 \times 10^4) = 620 \text{ mr/hr}$
Case 2(S-PM)	(10.55×10^{-3})	$(14.23 \times 10^4) = 1503 \text{ mr/hr}$
Case 3(NS-S)	(10.55×10^{-3})	$(12.0 \times 10^2) = 12.6 \text{ mr/hr}$
Case 4(NS-PM)	(10.55×10^{-3})	$(28.9 \times 10^2) = 30.5 \text{ mr/hr}$

DISTRIBUTION OF N^{16} IN EBWR

(Test Report No. 34A, by M. H. Theys and R. L. Mittl)

In a boiling water reactor, the O^{16} nuclide of the water flowing through the core is activated by a fast neutron reaction $O^{16}(n,p) N^{16}$. The product of this reaction is a N^{16} nuclide which decays with emission of β and α rays of 6.13, 7.13 and 8.89 Mev and with a half-life of 7.35s. This radioactive N^{16} is partly stripped out of the reactor water into the steam and is the origin of the main part of the activity in the piping system of the reactor. The purpose of the study was threefold:

1. To set up an experimental method to measure the absolute concentration of N^{16} nuclides in the water and steam of the reactor vessel.
2. To obtain data about the variation of those concentrations with power.
3. To study the mechanism controlling the distribution of N^{16} in the water and steam at the boiling region and to understand the effect of various additions in the feedwater on the steam activity.

Experimental Procedures for Measuring N^{16} Activity.

1. Measurement of N^{16} activity in water and steam:

To measure the activity in the water, a continuous sample is brought outside the reactor vessel by sampling tubes. Locations

of the sampling lines are indicated in Table I. After cooling, the sample flows to a flat coil in front of a scintillation counter. The pulses produced in the counter are then analyzed in a single channel pulse height analyzer and the activity of N^{16} is discriminated.

Table I - Elevation of Sampling Line Inlets in the Reactor

	Probe no.	Fuel element	Elevation, inches
First experiment	1	77 - 191	79
	2	77 - 191	
	3	77 - 191	
Second experiment (water level report No. 62A)	A ₁	77 - 191	40
	A ₂		61
	A ₃		64
	A ₄		67
	A ₅		100
	B ₁	129 - 264	69
	B ₂		75
	C ₁	0 - 301	62
	C ₂		68
Third experiment (feedwater additives)	A ₁	59 - 301	39.5
	A ₂		99-3/8
	A ₃	0 - 227	2-1/4*

Elevations are above the fuel element hole which is at an elevation of 713' 5.5".

* This distance is from the bottom of the dummy fuel element to which the probe is attached.

Three sets of probes were installed in the reactor vessel during 3 different operating periods. The first set gave general information and was used to study and to improve the experimental set-up. The second gave information about the activity of the water - steam interface, and the third set, having one probe below the core, one above the core, and one in the steam was used during hydrogen and hydrazine additions.

The cooling and the counting system is the same as that used for water level measurements, described in Test Report No. 62.

The pulses which are counted are those corresponding to an energy higher than 4.8 Mev absorbed in the crystal. It has been verified that the counting rate is actually caused by N^{16} by taking a spectrum of the radiation and a decay curve in order to check the energy of the radiation and the half-life of the activity.

2. Corrections applied to the counting rate:

Measurement of the concentration of the sample in the reactor vessel, that is, at the inlet of the sampling line, is required for the experiment. Due to the very short half-life of N^{16} , corrections must be applied to the measured counting rate in order to obtain this primary activity of the sample.

a. Delay time correction:

Takes into account the time spent by the sample in the cooling coil and in the lines before reaching the counter. A measurement of this delay is obtained by measuring the time between the opening of the high pressure valve and the moment the activity is scanned by the counter. If t is this delay time the correction is a factor $e^{+\lambda t}$ where $\lambda = 0.0943 \text{ s}^{-1}$.

b. Counting time correction:

During the time the sample is in the counting coil, the decay is so important that the concentration of the radioactive atoms is larger at the inlet than at the outlet of the coil and the counting rate is not exactly proportional to the time spent in the counter. The correction is a

factor $\frac{1}{1 - \lambda \frac{\Delta T}{2}}$ if Δt is the time spent by the sample in the counting coil.

c. A third correction is due to the noncondensable gas contained in the steam so that the counting coil is not completely filled with sampling water. It has been shown in Test Report No 22A that the average amount of gas contained in the steam is 29 cc/lb of steam. Thus the actual volume of active sample is easily obtained (See Test Report 22A)

3. Calibration of the counter:

In order to determine the absolute concentration of N^{16} in the water and in the steam, the counter must be calibrated to convert the counting rate into terms of N^{16} atoms per gm of sample.

In this particular case, the calibration is especially difficult because of the complex geometry of the counting system and the variation of the efficiency of the crystal as a function of the energy of the radiation. As there are no practical ways to fill the counting coil with a solution emitting such high energy rays as 6.13 Mev, the calibration was made with a cobalt-60 solution of a known activity emitting 1.17 Mev and 1.33 Mev rays. Data for this correction was obtained from a Monte Carlo calculation on the IBM machine. The result is:

$$N(\text{atoms of } N^{16}/\text{gm}) = 126 C(\text{counts/sec})$$

4. Measurements at other locations:

When the EBWR is suddenly shut down a certain number of radiation monitors measure an activity decreasing to almost zero with the 7.35 sec half-life of N^{16} . They monitor the steam dryer, the air ejector, and the condenser hotwell. During the experiments readings were taken with these instruments. The result was analyzed in relation with the steam activity. These measurements were in part reported in Test Report No. 7.

5. Accuracy of the measurements:

The absolute concentration of N^{16} in the sample is finally given by the following formula:

$$N(\text{at/gm}) \propto \frac{e^{\lambda t} f_{\text{gas}} C(\text{cps})}{E \left(1 - \frac{\lambda \Delta t}{2} \right)}$$

Where:

$e^{\lambda t}$ is the delay time correction

f_{gas} is the noncondensable gas correction

C is the counting rate of pulses above 4.8 Mev

E is the number of pulses higher than 4.8 Mev per ray striking the crystal

$\left(1 - \frac{\lambda \Delta t}{2} \right)$ is the counting time correction.

The uncertainty in N is mainly due to E , t and C .

The high voltage of the single channel pulse height analyzer is not very stable, and the relation between the height of the pulse and the energy absorbed is not constant. The energy above which pulses are counted varies from 4.5 to 5 Mev. This corresponds to an uncertainty of $\pm 5\%$ in E .

An exact measure of the time delay is very difficult. The absolute error in t is of the order of 1 sec which corresponds to a relative error in the correction coefficient of about λ times the absolute error in t . This gives an uncertainty due to t of about 10%.

The uncertainty in C is the greatest because the counting rate experiences large fluctuations for water and even larger for steam. These fluctuations correspond to actual fluctuations in the sample activity and in the N^{16} concentration. Long counting time is a way to decrease this effect and give a good average value. However, there is also a gradient of the concentration in the reactor water due to the flow pattern and the flux distribution in the core, so that even sampling lines at the same elevation lead to different concentrations. A difference of about 20 percent was observed between the center and the outer radius of the core. The water level has small influence on the N^{16} concentration in the water. Finally the accuracy on the measured concentrations is of the order of 30 percent.

Results and Discussion of the Measurements. Table II lists the values of N^{16} concentration in the water and in the steam as well as the activity in the steam piping system at various reactor powers.

Table II

Power, Mw	N^{16} concentrations (10^6 atoms/gm)				N^{16} activities at radiation monitors (mr/hr)		
	Line A_1	Line A_3^W	Line A_3^S	Line A_5	3	4	5
10	2.01	3.00	0.20	0.16	31	90	10
15	2.75		0.36	0.31	50	210	24
20	3.27	5.1	0.62	0.53	75	360	40
25	3.27	6.47	0.92	0.78			65
30	4.45	7.45	1.43	1.19	150		110
40					600		240
62					720		580

(Some of the radiation levels were obtained from previous experiments)
The measured concentrations correspond to:

- A_1 - the top of the riser
- A_3^W - the water at the interface
- A_3^S - the steam at the interface
- A_5 - the steam in the dome

Radiation monitors correspond to:

3 - the steam dryer, 4 - the air ejector, 5 - the condenser hotwell.

In Fig. 34A-1 the data were plotted as a function of power. The steam activities and radiation levels are plotted as functions of the square of the power in Fig. 34A-2. We observe that the water concentrations are almost proportional to the power while the steam activity is proportional to the square of the power.

The separation coefficient α is defined as the ratio of N^{16} activity in the steam to that in the water at the interface. At 20 Mw for example, $\alpha = 0.115$. With a recirculation ratio of 0.021, it is seen that the amount of the activity which goes into the steam is only about 0.2 percent of what is produced in the core. This is in complete opposition with assumptions used until now in N^{16} calculation.

In Test Report No. 34 two hypotheses were assumed, "sweep" and "non-sweep." The "sweep" assumption assumes that all the N^{16} generated in the core is swept out of the water and leaves the core with the steam. In the "non-sweep" calculation, the assumption

Fig. 34A-2. N^{16} concentration vs reactor power squared

Fig. 34A-1. N^{16} concentration vs reactor power

is that the activated N^{16} retains the same molecular form which it had prior to activation so that one pound of steam carries the same amount of N^{16} activity as that contained in a pound of reactor water. These assumptions would give a separation coefficient larger than one for the first case and equal to one for the non-sweep assumption. This is well above the value of $\alpha = 0.115$ observed at 20 Mw. To explain this low separation coefficient, we have first to discuss the chemical form of N^{16} in the water.

1. Chemical form of N^{16} in the reactor system:

N^{16} is produced by a nuclear reaction and therefore appears in the water as a free atom, with a very high

chemical reactivity. This atom will combine with the other atoms present in the system, O, N, and H to form three types of chemical compounds:

- (1) a neutral species, nitrogen gas
- (2) an anionic species, NO_2^- or NO_3^-
- (3) a cationic species, NH_4^+

As the reactor water is of very high purity, it can be assumed that these compounds will account for most of the N^{16} produced.

Because of the very short half-life of N^{16} the usual methods of chemical analysis are inapplicable but information may be obtained by passing the water and steam samples through an ion exchanger before sending them to the counter.

By a system of valves one of three different beds may be tied into the sampling line circuit so that the sample can be brought in front of the counter directly and by passing the beds through an inactive bed, through an anion bed or through a cation bed. The beds are 2.5 in. high and $5/8$ in. diameter cylinders. The volume of each bed is 12.7 cm^3 .

The resins are, for the anion bed Nalcite SBR (Comroe
for the cation bed Nalcite HCR Laboratories)

The inactive bed was filled with a sand having about the same particle size as the resins. The time delay between the inlet of the sampling line and the counter was about the same for each of the three beds inserted in the line (~ 25 sec). The counting rate corrected for the different time delays shows that the inactive bed was not absorbing N^{16} .

Table III gives the absorption of N^{16} activities on the different beds in percent of total activity of 20 Mw

Table III - Absorption of N^{16}

	Water above the core	Water below the core	Steam
Anion bed	80% 5	85% 5	0
Cation bed	10% 5	15% 5	35% 20%
Inactive bed (cps)	21,000	9,300	1,300
Time delay (sec)	10	12	23

These results show that after its formation N^{16} combines mainly in an anionic species which remains in the water. Only the proportion associated in a cationic species and in nitrogen gas is partly stripped out in the steam. The part which is stripped out of the water increases with reactor power.

2. Effect of various additions in the feed water:

In order to check the process of the N^{16} separation into the steam, various additives were put into the feed water for disturbing the cation and anion proportions.

- a. Nitrogen and argon gas addition. Nitrogen and argon gases have no effect on the chemical combination of N^{16} and therefore no changes should be observed in the steam activity. With 4.5 liters per min of argon or 20 liters per min of N_2 in 60,000 lb per hr of feedwater no change in activity was observed in the water, steam or radiation monitor levels.
- b. Hydrogen in Borax IV- H_2 addition is reported to have increased greatly the activity of the steam.¹ The same effect is observed in EBWR but the sensitivity of the steam activity to H_2 addition is much higher. Figure 34A-3 gives curves of steam and interface activity for different H_2 addition flow rates. In Table IV data are reported about the chemical action of H_2 on the N^{16} compounds.

Fig. 34A-3.

Effect of hydrogen addition.

¹G. K. Whitham and R. R. Smith, "Water Chemistry in a Direct Cycle Boiling Water Reactor." A/Conf. 15/P/946. Geneva 1958.

Table IV - Effect of H₂ Additions

N ¹⁶ compounds, %							
	Steam			Above the core		Below the core	
	Inactive	Cation	Anion	Cation	Anion	Cation	Anion
During	10	90	0	55	45	92	8
Before	35	65	0	12	88	15	85

N ¹⁶ activity, cps						
	Radiation level (mr/hr)			Steam	Above the core	Below the core
	3	4	5			
Before	75	110	42	150	8200	3000
During	300	450	200	1600	8500	2600

The activity in the steam increases by a factor of 10 so that about 2% of the activity produced is found in the steam. The activity of the water remains almost constant inasmuch as the amount of activity stripped into the steam is a small fraction of that produced. But the effect of H₂ addition on the partition of N¹⁶ in cationic and anionic species is striking especially after the feedwater inlet. The large increase in cationic species in the water is certainly related to the large increases of N¹⁶ activity of the steam

- c. Hydrazine and ammonia. Hydrazine and ammonia were added to the reactor water in order to study the separation of ammonia in the steam and the effect of the hydrogen coming from the decomposition. Rates of addition were:

10 ml/min of commercial hydrazine as a 35% solution in water in 40,000 lb/hr feedwater (13.3 Mw)

10 ml/min of a 0.2% ammonia solution in 60,000 lb/hr feedwater (20 Mw).

In both cases the steam activity increases while the water activity remains constant, as shown in Table V.

Essentially no change in the composition is observed in the water in opposition to the case of hydrogen addition. Probably the increase of activity in the steam is due to the combination of N¹⁶ with the free hydrogen atoms produced in the core by decomposition of hydrazine and ammonia.

Table V - Effect of N₂, Hydrazine and NH₃ Addition

	N ¹⁶ activities, cps			Radiation level (mr/h)		
	Steam	Above the core	Below the core	3	4	5
Before	150	4,400	1350	60	450	30
N ₂ H ₄ add.	1400	4,500	1500	270	700	60
Before	130	8,200	3000	75	150	35
NH ₃ add.	465	8,000	3000	150	600	65

Samples Passed through Ion Beds before Counting							
	N ¹⁶ compounds, %						
	Steam			Above the core		Below the core	
	Inactive	Anion	Cation	Anion	Cation	Anion	Cation
Before	35	0	65	83	12	86	14
N ₂ H ₄ add.	27	0	73	85	15	85	15
NH ₃ add.				87	13	80	20

We can expect that in this case, the free nitrogen produced in the core by decomposition of N₂H₄ or NH₃ shall behave the same way as the free N¹⁶ produced. An increase in conductivity in the water and a higher pH in the hotwell than in the vessel was observed in both cases. This means that nonradioactive NH₃ has also a preference to pass into the steam.

Conclusion. N¹⁶ after its production by a (n,p) reaction with O¹⁶ is a highly reactive atom and combines in anionic species (90%), in cationic species (10%) or as nitrogen gas (~0.5%) in the normal operating condition of EBWR (pH = 7, 600 psi, 489F).

The N¹⁶ which goes into the steam is only a part of the cationic species and of the nitrogen gas; no anionic species is in the steam. The part of the cationic species stripped into the steam increases almost proportionately to the power of the reactor so that the N¹⁶ concentrations in the reactor water are proportional to the power, and in the steam are proportional to the square of the power.

Any addition to the feedwater, such as H₂, hydrazine or ammonia, which increases the proportion of cationic species increases the steam activity. Additions without chemical action, argon or nitrogen, do not change the steam activity.

FLUX DISTRIBUTION IN TANGENTIAL INSTRUMENT TUBE

(Test Report No. 35, by A. Hirsch and W. Lipinski)

The current output from a compensated ionization chamber was measured at various positions in the lower south tangential instrument

tube. The chamber current versus chamber position is shown in Fig. 35-1. The flux at any point can be estimated from the approximate chamber sensitivity of 10^{-14} ampere per nv.

Fig. 35-1

Ionization chamber
current vs location
in instrument tube

NEUTRON SURVEY BENEATH EBWR

(Test Report No. 47, by S. Meyers)

Purpose. The purpose of the survey was, (1) to investigate the effectiveness of the neutron shield beneath the reactor core, with particular emphasis on streaming paths, and (2) to determine the neutron level in the basement directly below the core, where the control rod drive equipment is installed. This report includes data and information from two separate surveys, one conducted by Radiation Safety (Survey No. 1). The other (Survey No. 2) comprises the main body of the report.

Survey No. 1. The equipment used consisted of a BF_3 long counter, a $3\frac{1}{4}$ in. thick cylindrical sleeve made of paraffin clad with brass and cadmium, a high voltage supply, an amplifier, and a scaler to record the counts.

Two counts were taken in the area, with the reactor operating at 34 Mw and 600 psi. The flux recorded for the first count was $1089 \frac{n}{\text{cm}^2 \text{ sec}}$. That for the second was $1041 \frac{n}{\text{cm}^2 \text{ sec}}$. With a linear relation between power and flux, at 20 Mw and 600 psi the flux would be $618 \frac{n}{\text{cm}^2 \text{ sec}}$.

Survey No. 2. This survey was done in two separate steps, using neutron sensitive film in the first and a fission counter in the second.

- Legend
- O First survey at 20 Mw of power
 - X Second survey at 34 Mw of power

Fig. 47-1. Location of film.

lower energies of the fast neutron region, about 1 Mev. Consequently, there was a fast flux of $20 \frac{n}{cm^2 sec}$ at 1 Mev resulting in a dose rate of about 3 mrem per hr.

2. Step Two: The equipment used was a fission counter with a preamplifier, a linear amplifier, a pulse height discriminator, a high voltage supply, a 2 in. thick cylindrical polyethylene sleeve, and a scaler to record the counts.

Initially a background run was made, with the counter remote from the neutron flux, to determine the correct setting of the pulse height discriminator. A setting of 600 which effectively eliminated noise and neutron background was arrived at. The counter was then placed beneath the reactor in the positions indicated in Fig. 47-2, and about five feet

1. Step One: Two neutron sensitive film badges were exposed on the grating beneath the reactor as shown in Fig. 47-1. They were left in place for 23 hours with the reactor at 20 Mw and 600 psi. The indicated flux was $19.3 \frac{n}{cm^2 sec}$. Five more film badges were exposed (Fig. 47-1) for 20 hours with the reactor at 34 Mw and 600 psi. The indicated flux was $35.8 \frac{n}{cm^2 sec}$. With a linear relation between power and flux, at 20 Mw and 6 psi the flux would be $21 \frac{n}{cm^2 sec}$.

Note that the neutron sensitive film is not responsive at energies below approximately 1000 ev. Also the tracks of those neutrons which did not record indicated the

Fig. 47-2. Neutron counter positions.

above the basement floor. The counts were recorded and tabulated. They indicated a thermal flux of $142 \frac{n}{cm^2 sec}$, and a dose of about 0.5 mrem per hour. Next the counter was inserted into the polyethylene sleeve and counts taken at the same positions as before. These counts indicated an epithermal flux of about $428 \frac{n}{cm^2 sec}$, and a dose of about 2.5 mrem per hour. Therefore, the resultant total dose due to neutrons was about 6 mrem per hour.

Calibration Calculations. The fission counter was calibrated by placing the polyethylene sleeve over the counter, and placing the source against the unit (Fig. 47-3A). To facilitate calculation the system was represented as shown in Fig. 47-3B.

A - Calibrating Technique

B - Point Source Equivalent of Fig. 3A

Fig. 47-3. Counter details

To determine how much of the Po-Be source, rated at $5 \times 10^4 \frac{n}{sec}$ was effective at the counter, the actual geometry was replaced by an equivalent point source and a solid angle technique was used to determine the effective source strength.

$$W = \frac{S}{R^2} = \frac{202.5}{188} = 1.08 \text{ steradians}$$

W = effective solid angle

S = effective surface area, 202.5 cm^2 , of the counter

R = radius to the counter, 13.7 cm

$$R^2 = 188 \text{ cm}^2$$

This represents a percentage of the total source emission by an amount related to the maximum solid angle of 4 steradians:

$$\frac{1.08}{4} 8.6\%.$$

The number 8.6% represents the proportional amount of the source strength which was scanned by the fission counter:

$$(.086) (5 \times 10^4) = 4.3 \times 10^3 \frac{n}{\text{sec}} .$$

To determine the neutron flux arising from this source, divide by the effective area of the counter:

$$\frac{4.3 \times 10^3}{202.5} = 21.2 \frac{n}{\text{cm}^2 \text{ sec}} .$$

Flux Calculations. The source flux, as determined by the calibration calculations, caused a count rate of 10.2 counts per second. Since the average number of counts beneath the reactor was 205.6 per second, the flux was

$$\frac{(205.6)(21.2)}{10.2} = 428 \frac{n}{\text{cm}^2 \text{ sec}} \text{ (epithermal)} .$$

The thermal flux was determined by multiplying the epithermal by the factor $\frac{4067}{12333}$, which represents the ratio of thermal to epithermal counts:

$$428 \times \frac{4067}{12333} = 142 \frac{n}{\text{cm}^2 \text{ sec}} \text{ (thermal)} .$$

Test Results (Survey No. Two).

- 1 Since the counts at all points are nearly equal, it is evident that there are no streaming paths through the shield.
2. The neutron level at normal operating conditions of 20 Mw and 600 psi is:

$$\text{Thermal flux } 142 \frac{n}{\text{cm}^2 \text{ sec}}$$

$$\text{Epithermal flux } 428 \frac{n}{\text{cm}^2 \text{ sec}}$$

$$\text{Fast flux } 20 \frac{n}{\text{cm}^2 \text{ sec}}$$

This total compares favorably with the results of the Radiation Safety survey (Survey No. One).

The analysis indicates that about 95 percent of the neutron flux lies in an energy range between 0.025 ev (thermal) and 1000 ev (epithermal). This results in a total neutron dose rate of about 6 mrem per hour

GAMMA FLUX DETERMINATION OF EBWR CORE

(Test Report No. 49, by C. A. Pesce and J. F. Pohlman)

The purpose of this experiment was to determine the feasibility of measuring the gamma flux in the core of EBWR by inserting a small ion-chamber probe between the fuel plates of the fuel assemblies. All measurements, tables, figures, and calculations are based on modified loading No. 46 of EBWR core (see Introduction. Fig. I-1).

Description of Apparatus.

1. Construction and operation:

A cross section of the probe assembly is shown in Fig. 49-1.

Fig. 49-1. Probe assembly.

Figure 49-2 shows the schematic diagram of the system. The probe is essentially an ion chamber; a voltage of 600 v is impressed on the center wire of the coaxial cable, and the outer casing, the .170 chamber ($3\frac{3}{8}$ " long), is ionized by the gamma flux and the resulting current is read on the Keithly micro-microammeter. By applying the probe sensitivity to the current read, the gamma flux can be directly calculated.

Fig. 49-2.

Probe schematic

2. Calibration:

The probe was calibrated in the gamma swimming pool reactor in Building 310. A dosimeter test was made at each of five sampling points in the reactor to determine the gamma flux at these points. The test was made by personnel in charge of the gamma facility. The flux determinations at the sampling points are shown in the column marked r/hr on Table I. Probe current readings were also taken at the sampling points and are also shown in Table I. The data of Table I is plotted in Fig. 49-3. An average current and average gamma flux was calculated. These are 5.98×10^{-8} amps and 3.02×10^5 r/hr respectively. From these values the probe sensitivity (P.S.) was calculated:

$$\text{P.S.} = \frac{5.98 \times 10^{-8}}{3.02 \times 10^5} = 1.98 \times 10^{-13} \text{ amp/r/hr}.$$

Table I - Probe Calibration

Sampling point	Probe current $\times 10^{-8}$ amp	r/hr $\times 10^5$
1 bottom	4.60	2.07
2	6.10	3.20
3	6.95	3.80
4	6.70	3.60
5 top	5.55	2.43
Average	5.98	3.02

Fig. 49-3. Probe current.

Results. Three particular phenomena were under inspection:

(1) vertical flux distribution of two typical fuel elements near the center of the core, (2) difference in flux level between individual plates of fuel assemblies, and (3) horizontal flux distribution at various cross sections of the core, taken at the point of maximum vertical flux ($16\frac{1}{2}$ " from bottom of fuel plate).

- Figure 49-4 and Table II show the vertical flux distribution at the center of two typical elements near the center of the core. As expected, the peak flux decreases with distance from the core center. The two curves cross at approximately 30 inches; above this point the element farthest from the core shows the greater flux. This is quite probably due to the fact that the reactor is operated with the center rod withdrawn to approximately this point; therefore, the absorbing effect of the rod would be more noticeable above this elevation than below it.

Table II - Data for Fig. 49-4

Inches from bottom of element	Gamma flux $r/hr \times 10^5$ at center of element	
	101-160	129-159
7 $\frac{3}{4}$	2.12	2.22
10 $\frac{1}{2}$	2.68	2.73
16 $\frac{1}{2}$	2.93	3.08
22 $\frac{1}{2}$	2.83	2.93
28 $\frac{1}{2}$	2.37	2.42
34 $\frac{1}{2}$	1.72	1.49
40 $\frac{1}{2}$	0.98	0.85
46 $\frac{1}{2}$	0.48	0.40
52 $\frac{1}{2}$	0.15	0.10

Fig. 49-4. Flux distribution - center of core.

Heavy emphasis must be placed on the fact that the flux as shown in Fig. 49-4, as well as in the figures following, is the flux at a particular point in the reactor core, not simply the flux at a particular point on the element. Thus the shadowing effects of control rods and adjacent elements are active and the vertical flux peak of the individual element is flattened.

- The effect of the control rods on gamma flux distribution is illustrated in Fig. 49-5 (Table III) and Fig. 49.6 (Table IV). Figure 49-5 shows the effect of the center rod on an element adjacent to it; the flux is seen to increase from fuel plate to fuel plate with distance from the rod. The flux distribution at the center of the core is fairly flat; hence, the effect of the control rod is quite noticeable.

Fig. 49-5. Effect of center control rod on gamma flux.

Table III - Data for Fig. 49-5

Inches from bottom of element 129-370	Gamma flux $r/hr \times 10^5$ between plates				
	1	2	3	4	5
$7\frac{3}{4}$	2.42	2.32	2.27	2.22	2.07
$10\frac{1}{2}$	3.03	2.78	2.78	2.73	2.58
$16\frac{1}{2}$	3.44	3.28	3.18	2.98	2.88
$22\frac{1}{2}$	3.28	3.08	3.03	2.88	2.68
$28\frac{1}{2}$	2.58	2.52	2.48	2.38	2.22
$34\frac{1}{2}$	1.82	1.67	1.51	1.62	1.39
$40\frac{1}{2}$	0.98	0.93	0.92	0.88	0.84
$46\frac{1}{2}$	0.43	0.42	0.41	0.41	0.39
$52\frac{1}{2}$	0.11	0.11	0.12	0.12	0.11

Fig. 49.6. Effect of corner control rod on gamma flux

Table IV - Data for Fig. 49-6

Inches from bottom of element 47-414	Gamma flux $r/hr \times 10^5$ between plates				
	1	2	3	4	5
$7\frac{3}{4}$	0.99	0.93	0.82	0.68	0.63
$10\frac{1}{2}$	1.16	1.15	1.03	0.88	0.78
$16\frac{1}{2}$	1.34	1.17	1.14	1.01	0.90
$22\frac{1}{2}$	1.30	1.12	1.09	0.97	0.87
$28\frac{1}{2}$	1.13	0.94	0.94	0.82	0.74
$34\frac{1}{2}$	0.73	0.65	0.65	0.49	0.49
$40\frac{1}{2}$	0.33	0.32	0.31	0.28	0.28
$46\frac{1}{2}$	0.14	0.15	0.14	0.13	0.13
$52\frac{1}{2}$	0.04	0.04	0.05	0.04	0.04

Figure 49-6 shows an opposite effect; that is, the flux is seen to decrease with distance from the control rod. However, this is a corner rod, and the flux is decreasing at a very high rate, apparently minimizing the absorbing effect of the rod.

3. Figures 49-7 (Table V), 49-8 (Table VI) and 49-9 (Table VII) show the gamma flux distribution at three separate cross sections of the core. Values plotted are the maximum vertical flux on each element as read at points $16\frac{1}{2}$ " from the bottom of the fuel element.

Fig. 49-7. Gamma flux - cross section I.

Table V - Data for Fig. 49-7

Element number	Gamma flux $r/hr \times 10^5$
0- 90	0.45
27-113	0.83
54-135	1.19
72-149	2.22
89-158	2.88
101-160	2.93
109-148	3.03
109- 99	3.18
101- 54	3.08
89- 21	2.88
72-414	2.17
54-390	1.19
27-361	0.64
0-333	0.35

Figure 49-7 (cross-section I) shows that in general the curve is symmetrical, except for the dip between fuel elements 80-25 through 129-159. This dip in the curve can probably be explained by the fact that the natural thin element which was at position 129-164

Fig. 49-8. Gamma flux - cross section II

Table VI - Data for Fig. 49-8

Element number	Gamma flux $r/hr \times 10^5$
0- 55	0.44
23- 66	0.86
44- 75	1.08
54- 73	1.20
59- 64	1.44
59- 48	1.43
54- 29	1.21
44- 9	1.15
23-399	0.85
0-371	0.55

Table VII - Data for Fig. 49-9

Element number	Gamma flux $r/hr \times 10^5$
0-371	0.55
47-414	1.14
80-25	2.58
101-54	3.08
129-159	3.08
129-370	3.18
101-266	2.98
80-237	2.53
47-203	1.12
0-159	0.57

Fig. 49-9. Gamma flux -
cross section III

was replaced just before this study by a fuel element composed of enriched thin plates. This last element was originally located at position C-4 and had been subjected only to a relatively low irradiation. Cross-section II is parallel to I, but farther away from the center of the core. As plotted in Fig. 49-8 it shows first a decrease in over-all gamma flux intensity, and second, three peaks at the control rods. This latter effect is undoubtedly caused by the activity of the shroud.

Cross-section III (Fig. 49-9) shows a symmetrical flux distribution. This is as expected since the core composition about this plane is symmetrical.

Discussion. In general principle, the flux measuring method described in this report appears acceptable. The data recorded shows that the probe is sensitive to the presence of control rods, elements of lower or higher activity, and dummy elements. If the flux reading at a particular point on an element, rather than at a particular point in the core (see section C-1) is desired, it can be ascertained with no more trouble than is required to remove an element from the core.

True flux readings, of course, depend upon the conversion of current readings to roentgens per hour. The method of determining probe sensitivity is described (2. Calibration), and perhaps leaves much to be desired. The probe detector and the ferrous sulphate dosimeter against which the probe was calibrated have different energy dependencies and do not integrate the spectrum in exactly the same manner. For the purpose of this initial study, however, the accuracy of the probe sensitivity as determined (2. Calibration) was assumed to be sufficient to the end in view. Further experimentation toward a more reliable value of probe sensitivity is being undertaken by the writers.

Minor modifications of the probe guide are also contemplated to yield flux measurements at the bottom of the fuel element, beyond the $7\frac{3}{4}$ in. minimum point used in this study.

MEASUREMENT OF GAMMA FLUX IN EBWR - A COMPARISON OF TWO METHODS

(Test Report No. 49A, by C. A. Pesce and J. F. Pohlman)

The flux of the EBWR core has been measured by chemical dosimetry (Test Report 25A) and by use of an ionization chamber probe (Test Report 49). The two methods and the values obtained by each are compared in this report. Since the data given in these two reports were taken two weeks apart (14 days after shutdown for T.R. 25A, 28 days after shutdown for T.R. 49) and by using two different procedures (one in the core, the other out of it) direct comparisons are misleading.

Prior to any meaningful comparison of the two methods for determining flux in the core, a check on the two made under identical conditions was required. This was accomplished by measuring the flux of two natural thin elements in the storage pit; these elements were formerly located at the center of the core. Results of this test are reported here. Also included in this report is a full saturation curve of the probe at two flux levels with voltage varying from 0 to 900 v.

Results of Storage Rack Test. Discussion of this test must precede any comparison of T.R. 25A and T.R. 49. Elements T-21 and T-23, formerly at positions 129-164 and 129-53, were measured, free of any surrounding radiation, in the storage tank at EBWR. T-21 was dosimetered on sides A, B and C; it was also probed on sides A, B, C, D (Figs. 49A-1-2-3) and between the plates at points 1, 3, 4 and 5.

Fig. 49A-1. Element T-21
side A dosimetry.

Fig. 49A-2. Element T-21 sides B
and D dosimetry.

T-23 was dosimetered on sides A and C (twice); it also was probed on sides A, B, C and at points 1-5. The horizontal flux plots (Fig. 49A-4 and Fig. 49A-5) are nearly identical, varying by not more than 6 percent.

Fig. 49A-3. Element T-21 side C dosimetry

Fig. 49A-4. Horizontal flux plot, element T-23

Correlation of probe and dosimetry readings is shown in Figs. 49A-1-2-3-6-7. Readings are seen to agree quite well from the top of the element

Fig. 49A-5. Horizontal flux plot, element T-21.

down to a point roughly 20 in. from the bottom of the element. The very erratic probe readings (assuming the dosimetry to be correct) below this point were due to the difficulty experienced in properly positioning the detector of the probe so that it would be midway on the fuel plate and also not more than $\frac{1}{2}$ in. from the surface of the plate. This difficulty is caused by the flexibility and bowing of the probe casing at long extensions, plus the increased distance from the observer.

The writers are convinced from the data presented here that the probe yields readings equal to those obtained by dosimetry, provided that both readings are taken at the same place on the element and at the same time after shut-down. Worthwhile probe readings of the element surface at elevations less

Fig. 49A-6. Element T-23
side C dosimetry

Fig. 49A-7. Element T-23
side A dosimetry

than 20 in. from the bottom of the element would, however, require modification in the mechanical design of the probe.

The vertical cross-section plotted in Figs. 49A-8 and 49A-9 are especially interesting. For both elements, side C is more active than A, and point 5 is more active than point 1. Furthermore, they are more active by approximately the same amounts (15 to 20%). The parabolic type plots are to be expected by the following reasoning. Suppose plates A and C to be of equal activity at these two points, the detector scans one plate in its entirety and the other 5 in diminishing degrees; at points 1 and 5 it scans two plates completely, plus diminishing parts of the other four.

Fig. 49A-8. Vertical cross section,
element T-21.

Fig. 49A-9. Vertical cross section,
element T-23.

At point 3 it scans two plates totally and larger portions of the other four plates than at points 1 and 5, or 2 and 4. Thus, if the plates A and C were equal, a parabolic form with a peak at point 3 would be expected. Since in this case C is hotter than A, the peak has been shifted to point 4.

Note should be made of the fact that the activity measured at point 3 is 71 percent higher than point A, and 48 percent higher than point C, based on average values for the two elements at their maximum points.

Comparison of Results of T.R. 25A and T.R. 49. Correlation between the two methods of flux measurement has been demonstrated in the preceding

section. The results of the two test reports in question do not appear to agree at all, as shown in Fig. 49A-10. Peak values agree, but occur in different elements. Figures 49A-11 through 49A-17 show varying degrees of correlation of the individual elements as read by both methods. Before continuing, we must reiterate three facts pertinent to this discussion.

Fig. 49A-10. Radial gamma flux plot north-west quadrant.

were thus corrected, the probe readings would then be 5 to 90 percent higher than the corresponding dosimetry values.

Second, the flux as measured in the center of an element is appreciably higher than that measured on the outside face, 44 to 72 percent in the case of elements T-21 and T-23, for example.

Effect of Surrounding Elements in Probe Readings. Probe readings in the core cannot be taken as representing the actual flux conditions until the effect of pick-up from surrounding elements and control rods is determined. This effect has been open to much conjecture. The only way that it can be positively determined is to probe a representative sample of elements in the core and removed from the core. Data collected thus far, however, can with the aid of a few assumptions yield certain information on the subject.

Fig 49A-11. Gamma flux plot element 129-370.

Fig. 49A-12. Gamma flux plot element 101-266.

Fig. 49A-13. Gamma flux plot element 80-237.

Fig. 49A-14. Gamma flux plot element 47-203.

Fig. 49A-15. Gamma flux plot element 54-179.

Fig. 49A-16. Gamma flux plot element 54-241.

Fig. 49A-17.

Gamma flux plot element 23-172.

Element T-23, formerly at the center of the core, was measured in isolation. The enriched thin element at 129-370 was measured in the core. The first assumption is that the enriched thin element would have displayed a cross-section plot similar to that shown for T-23, had it (129-370) been measured outside the core. Secondly, it was assumed that each point of the cross section will decay at a constant value, and that the decay of one element will be proportional to decay of the other at each point of the cross section. The final assumption has been previously stated; a flux drop of 27 percent occurred between time of dosimetry and time of probing.

Fig. 49A-18. In-core and isolated cross section.

corrected for a decay value of 27 percent. This indicates negligible background pickup from the control rod, which seems unlikely. However the value plotted as point A for 129-370 might as easily have been point C, which reads 15 percent higher. (If the two elements can be compared at all, this is a logical assumption.) If such was the case, this would indicate a background effect in probe readings of 15 percent (normalizing the curves in Fig. 49A-18 to agree at point A, using the lower value for 129-370, would raise the dashed-line curve 15% for points 1-5).

This analysis of environmental effects is speculative. If proven correct it would apply directly only to the particular location mentioned, the core center. However, the assumptions, if proven true, can lead to further investigations and are therefore considered to be worth making.

Probe Saturation Curve. Figure 49A-19 is the saturation curve for the probe; it shows that any voltage between 100 and 800 v is acceptable. The lowest possible voltage is preferable, however, since this results in minimum leakage current (leakage due to the resistance path from the positive battery terminal to ground). At 157 volts the probe will read as low as 10^4 rep/hr with good accuracy. Also there is evidence in Fig. 49A-19 of the fact that the probe behaves as a proportional counter at voltages above 825.

Fig. 49A-19. Probe saturation curve.

The probe sensitivity (PS) was checked at the same time the above data was recorded; for a probe voltage of 157 v the PS is still 1.98×10^{-13} amp, rep, hr.

Conclusions. The probe has proven to be an effective and reasonably accurate means for measuring gamma flux, as demonstrated in the storage rack tests. Flux readings can be taken on the surface of an element, with difficulty due only to mechanical design of the probe supporting members. From the viewpoint of operation, the probe method is admittedly superior. It is quicker, cheaper, easier, and direct, omitting several intermediate steps which consume time and increase possible error.

The prime criterion of an instrument's worth is its ability to provide the desired information. Presumably the chief interest in measuring core flux is to determine the contribution of each element to the total heat flux. This requires an indication of the average flux value for the element, which is best found by measuring the activity of each side of the element with the element removed from the core and then averaging the two values. Unfortunately, this is an arduous and time consuming task by either method. The obvious shortcut is, of course, to probe the core in place, as was done in Test Report 49. The inadequacies of this method are just as obvious. In Fig. 49A-9, for instance, the flux at point 3 is not really 72 percent greater than at point A, the probe reads that much more because at point 3 it scans

three elements on each side of it, but at A it scans six elements on only one side of it. This factor of 72 percent will not be the same for all elements, however, since it is dependent on the increase in flux from one side of the element to the other. Such being the case, probe readings taken at point 3 with the element removed from the core are not representative of a valid average flux value; a more accurate way would be to take an average of points 1 and 5. However, the latter points are significantly affected by adjacent elements if the element is scanned in the core. If the assumptions made in Fig. 49A-18 are correct, points 2 and 4, and thus 3, are seen to be affected only by a general background flux within the given element. Since this background flux would be roughly proportional to the flux of the element itself, it can be safely ignored.

The task at hand is two-fold. First, all of the assumptions made in developing Fig. 49A-18 must be either proven or disproven. Second, the relationship between the reading at point 3 (in the core) and the average of readings at A and C (out of the core) for elements at different locations in the core must be determined.

Calculations. The 27 percent decay between dosimeter readings and probe readings were determined on this basis:

The rate of emission of beta and gamma energy at time τ

$$\approx 3.7 \times 10^{10} p[(\tau - T_0)^{-0.2} - \tau^{-0.2}] \text{ Mev/sec.}$$

where $\tau - T_0$ is time after shutdown, τ is the time after startup, and T_0 is the time of operation.* A relative value for the two cooling periods is of more interest here than a definite value for emission rate. Thus the empirical part of the equation, including the p (power) term, will cancel - whatever its value it will be the same for both cooling periods. Time of operation, T_0 , is difficult to estimate; considering the age of the reactor, a T_0 of more than 300 days would seem absurd. Following are decay values assuming different times of operation:

<u>T Days</u>	<u>% Decay</u>	
10	48.4	where % decay
30	42.1	
50	38.0	$\frac{(\tau_1 - T_0)^{-0.2} - \tau_1^{-0.2} - (\tau_2 - T_0)^{-0.2} - \tau_2^{-0.2}}{(\tau_1 - T_0)^{-0.2} - \tau_1^{-0.2}}$
100	33.1	
150	30.6	where $\tau_1 - T_0 = 14$ days $\tau_2 - T_0 = 28$ days
250	28.0	
300	26.8	
∞	12.9	

*S. Glasstone, Principles of Nuclear Reactor Engineering, p 119.

The value of 27 percent was selected because it is the lowest logically possible, according to the decay equation used. An interesting corroboration of this equation and the T value of 300 days is given by elements measured in the storage pit. In this case 3 - T (the cooling period) is 71 days; comparison to 1 - T (14 days), the time at which the core was dosimetered, shows the flux should drop 56.5 percent during this period. At 3 the average maximum for T-21 and T-23 was 832 rad/min; according to the above, at 1 it should have read

$$\frac{832}{1 - .565} = 1910 \text{ rad/min.} \qquad \frac{\text{rad}}{.9128} = \text{rep} \quad .$$

Element 129-370 was dosimetered at time 1 and read 3856 rad/min. But this was an enriched thin element; T-21 and T-23 are natural thin. The ratio is $3856/1910 = 2.02$, which is approximately equal to the enrichment factor, 2.0. The 3856 rad/min value was assumed to be for the colder side of 129-370. Were it the hot side, a T of 500 days would be necessary to arrive at the above enrichment factor. This means that either the cold side of 129-370 was dosimetered, or the above analysis of decay is inaccurate.

RADIOACTIVE CRUD CLEANUP IN NOZZLES BELOW EBWR REACTOR

(Test Report No. 56, by E. A. Wimunc)

A radiation survey in the sub-reactor room indicated very high activity levels. The general background radiation levels ranged around 500 mr/hr at head height on the control rod motor platform. This location is about 3 ft below the 6 in. pipe caps of the forced circulation inlet nozzles. Activity below this platform was 50 mr/hr - 5 ft above the basement floor.

A concentration of activity was located in three of the four 6 in. forced circulation lines. The fourth contains the purification system line and appeared to be relatively clean, since a reading of 300 mr/hr at 2 in. was obtained on this line and this activity level was probably affected by the high background level.

The nine control rod bushing housings were also very active with the center rod bushing having the maximum of 65 r/hr at contact, measured with a Jordan radiation probe.

The maximum reading of the four circulation nozzles was 50 r/hr at contact and this was the cap on the south line. It was difficult to determine the exact intensities of these bushings and nozzles, since scattering and radiation from the other units affected individual readings.

This report presents the method and the effect of the cleaning. The first attempts to clean the nozzles involved preparation of the reactor for a hydrostatic test. The reactor was filled with water, bottled up, and then pressurized to 1000 psig with one of the feedwater pumps. A $\frac{1}{2}$ in. drain valve on the 6 in. forced circulation cap was opened to serve as a blow-down. Not a drop of water came out, indicating that the drain was completely plugged.

An alternate and effective method was to relieve the pressure on the reactor and vent the system to the main condenser. A $\frac{1}{2}$ in. pipe was then

attached to the shutoff valve, as shown in Fig. 56-1, associated equipment used for the cleanup is also shown. The following sequence was used:

1. Opened shutoff valve.
2. Opened air valve for about 3 sec
3. Closed air valve and quickly opened drain valve.
4. Repeated above steps until full flow was obtained from drain valve.
5. A 4 lb leather-faced hammer was used to hammer the 6 in. cap to loosen any crud that may have adhered to the pipe.

After the above procedure the activity dropped to 4 r/hr. Radiation from the control rod bushings was responsible for most of this activity. After three of the nine bushings were cleaned, the level dropped to 2 r/hr.

Fig. 56-1.

About 50 gallons of water were drained from each nozzle to the shielded plastic bottle. The crud was permitted to settle to the bottom. A filter and a pump were used to empty the bottle to the drain. Activity of the water was 3700 d/m/cc and when diluted with three parts of clean water, was dilute enough to pump to the laboratory sewer.

The major activity constituent of the crud was cobalt-58. About 2 cubic in. of crud was flushed from each of the three nozzles. This was the residue of the scale that flaked off the fuel elements during the 62 Mw operation and collected in the pockets at the bottom of the reactor vessel.

A spectrographic analysis of the crud removed during this cleanup indicated that the composition was almost identical to the deposit on the fuel elements. There was a very strong indication of Al, 20%; Ni, 15%; Fe, 0.4%; Mn, 0.15%; Cr, trace; and very minute percentages of other elements.

A solution to the problem of crud deposition in the forced circulation nozzles is to install $1\frac{1}{2}$ in. valves in place of the $\frac{1}{2}$ in. valves, and then manifold these to the purification system. By operating one line at a time in a sequence, the crud could be kept from concentrating in the nozzles.

Concentration of crud in the control bushing rods is another problem. This could be remedied possibly with a blowdown system in an enclosed rack and pinion type design.

DETERMINATION OF THERMAL NEUTRON FLUX IN EBWR BY COBALT ACTIVATION

(Test Report No. 71, by R. E. Bailey and R. F. S. Robertson)

Location of Foils. In loading No. 47, two natural, heavy-fuel elements were inserted in the reactor core. Each element contained a Co flux monitor in the form of Al-Co wire (Co 0.50%, Mn 0.50%, remainder Al) which ran lengthwise through one channel of the fuel element. In one element, the wire was bare and in the other it was wrapped with 0.040 in. Cd foil. To protect the wires from corrosion, each was sealed in a stainless steel tube 0.25 in. O.D. with a wall thickness of $1/32$ in.

Details of the loading are given below:

<u>Element No.</u>	<u>Grid Location</u>	<u>Detector</u>	<u>Orientation</u>	<u>Location</u>
H-11	27 - 255	Al-Co foil (Cd covered)	West side between 2nd and 3rd fuel plates from south	Axial entire length of core
H-13	27 - 43	Al-Co foil	East side between 2nd and 3rd fuel plates from north	Axial entire length of core

It will be noted that these are two equivalent locations, at the outside of the core and diametrically across from one another.

Experiment. The fuel elements were in the reactor for a total of 811.5 Mw days. They were removed from the reactor during a shutdown and were stored in the fuel pit. Later the stainless steel tubes were removed from the fuel elements and the Al-Co wires were withdrawn from the tubes. It was found that the tubing in Element H-13 had leaked and some of the wire was badly corroded.

Each wire was cut into 55 pieces approximately one inch long. Each piece was weighed to the nearest 0.01 mg and was dissolved in an accurately known volume of 6N HCl. An aliquot of each solution was evaporated to dryness on a stainless steel planchet. Several planchets were checked on a 200 channel X-ray pulse height analyzer and it was established that only Co^{60} γ -ray peaks were present.

Each planchet was then counted in a gas flow proportional counter. The geometry of the counter was established by evaporating an aliquot of a standard Co^{60} solution on a stainless steel planchet similar to those used before, and counting on the same counter. The counter was standardized daily.

Some of the more heavily corroded pieces of wire were covered with an oxide deposit which would not dissolve in the HCl. The residue was filtered from the solution and counted. In all cases, it was found that the activity on this residue was a negligible fraction of the total. However, the presence of the oxide layer did produce an unknown error, in that weights of samples were taken to represent only pure alloy.

Calculations. The following parameters and nomenclature were used in the calculations:

Co^{60} half life	5.23 years
Co^{60} λ decay constant	$4.203 \times 10^{-9} \text{ sec}^{-1}$
Co^{60} $\sigma_{2200-2200} \text{ m/s}$ cross section	37.0 barns
N^{59} - atoms of Co^{59} present initially	
N^{60} - atoms of Co^{60} present at end of irradiation	
A^{60} - disintegration rate of Co^{60} (DPS)	

1. N^{59} was calculated from the weight and Co content of the wire. The counting rate was corrected for the aliquot taken and was corrected to disintegrations per minute. This value was in turn corrected for decay from the end of the irradiation period to the time when counted. N^{60} was calculated from this value.

The integrated flux ϕ_t was calculated:

$$\phi = \frac{A^{60}}{N^{59}\sigma} \left(1 - e^{-\lambda t} \right)^{-1} .$$

2. Because at the short irradiation this was simplified by substituting λt for $(1 - e^{-\lambda t})$. The error is less than 2%. Thus

$$\phi_t = \frac{A^{60}}{N^{59}\hat{\sigma}\lambda} = \frac{N^{60}}{N^{59}\hat{\sigma}}$$

This calculation involves the following assumptions:

- a. Loss of Co^{59} by neutron capture is negligible compared to the amount initially present.
- b. Loss of Co^{60} by neutron capture is negligible.
- c. No self-shielding corrections are applied. This is justifiable in view of the dilute nature of the Al-Co alloy.

The value of $\hat{\sigma}$ was calculated from σ_{2200} by using the Westcott formula:¹

$$\hat{\sigma} = (g + rs) \sigma_{2200}$$

For Co^{60} $g = 1$ and a value of $s = 1.238$ at a temperature of 250°C was obtained from Ref. 1. A value of $r = 0.091$ was used for initial calculations. On this basis, $\hat{\sigma} = 41.71$ barns.

From ϕ_t , the value of the flux at 20 Mw was calculated using the fact that the irradiation continued for 811.5 Mw days. Thus,

days at 20 Mw = 40.58 d

seconds at 20 Mw = 3.506×10^6 sec.

Table I contains values for ϕ_t and ϕ for both the bare and Cd-covered wire and the data are plotted in Fig. 71-1. From Fig. 71-1, it can be seen that the uncertainty in the calculated flux is about 6% for the base wire and about 3% for the Cd-covered wire. The larger error for the base wire is undoubtedly due to some oxidation occurring on exposure to the reactor water. The data for samples 30, 38, 46 and 51 are omitted from the plot because they show wide discrepancies from the rest of the data.

It was assumed that each piece of cut wire was one inch long and the points were plotted accordingly. Since some of the wires were slightly over and some were slightly under one inch long, the axial distribution of any point may not be accurate to better than 0.2 in.

¹C. H. Westcott, Chalk River Report CRRP-580 (TNCC-CAN-7) January 1957.

Table I - Flux Data

Distance from top of fuel element (inches)	Bare wire		Cd covered wire	
	ϕ_t ($\times 10^{-19}$)	ϕ (at 20 Mw) ($\times 10^{-12}$)	ϕ_t ($\times 10^{-18}$)	ϕ (at 20 Mw) ($\times 10^{-12}$)
1	.181	.516	.294	.084
2	.271	.772	.086	.024
3	.271	.773	.156	.044
4	.264	.754	.205	.059
5	.238	.678	.304	.087
6	.224	.640	.356	.102
7	.195	.556	.418	.119
8	.239	.680	.514	.147
9	.275	.785	.671	.191
10	.386	1.102	.625	.178
11	.367	1.048	.929	.265
12	.393	1.122	1.062	.303
13	.462	1.319	1.071	.306
14	.520	1.485	1.246	.356
15	.610	1.768	1.675	.478
16	.726	2.070	1.654	.472
17	.767	2.189	2.196	.627
18	.855	2.439	2.504	.714
19	.908	2.591	2.520	.719
20	.875	2.497	2.953	.843
21	1.117	3.187	2.934	.837
22	1.205	3.438	3.177	.906
23	1.274	3.635	3.321	.948
24	1.374	3.920	3.542	1.011
25	1.447	4.128	3.578	1.021
26	1.652	4.713	3.982	1.136
27	1.693	4.830	4.186	1.194
28	1.623	4.630	4.442	1.267
29	1.830	5.221	4.610	1.315
30	1.527	4.357	4.855	1.385
31	1.884	5.375	4.784	1.365
32	1.780	5.078	4.773	1.362
33	1.781	5.081	4.825	1.377
34	1.895	5.406	5.048	1.440
35	1.997	5.697	4.815	1.374
36	1.854	5.289	5.014	1.430
37	1.994	5.689	4.978	1.420
38	1.705	4.864	4.860	1.387
39	2.012	5.740	4.754	1.356
40	1.819	5.190	4.538	1.295
41	1.819	5.190	4.264	1.217
42	1.756	5.010	4.365	1.245
43	1.666	4.753	4.194	1.197
44	1.433	4.088	3.924	1.120
45	1.237	3.529	3.723	1.062
46	0.833	2.377	3.311	9.446
47	1.081	3.084	2.923	8.339
48	0.881	2.513	2.659	7.586
49	0.994	2.835	2.164	6.174
50	0.924	2.636	1.723	4.916
51	1.338	3.817	1.249	3.563
52	1.070	3.053	0.879	2.509
53	0.929	2.651	0.597	1.703
54	0.717	2.045	0.521	1.485
55	0.493	1.405	0.496	1.414

Fig. 71-1. Flux plots for positions 27-55 and 27-43.

tion of Co^{59} by neutrons of all energies. That labelled "Epi-Cd Contribution" represents activation of Co^{59} by neutrons of energies greater than 0.5 ev. Since a value $\hat{\sigma} = 41.17$ barns was used in calculating this curve, it must not be interpreted as a fast neutron flux distribution, but merely the correction which must be subtracted from the "total" curve to give the activation by neutrons in the thermal range 0.025 to 0.5 ev.

The "thermal" contribution curve must now be corrected again. In deriving it, a value of $r = 0.091$ was assumed. From the data, a better value of r may now be calculated from the cadmium ratio.

In Table II, the values of the cadmium ratio (R_{Cd}) are given. These values are obtained from the two smoothed curves in Fig. 71-1. From the Cd ratio the value of r , which defines the proportion of epithermal neutrons in the spectrum, is obtained from the equation (1).

$$R_{\text{Cd}}(g = 1) = \frac{1 + rs}{r} \left[\frac{1}{1.97} \sqrt{\frac{T}{T_e}} + s \right]^{-1}$$

The figure 1.97 corresponds to a 0.5 ev Cd cut-off. A value 1.238 is taken for s and again a temperature of 250°C (523°K) is assumed for T .

Values of r for different distances from the top of the fuel plate are listed in Table II and are plotted in Fig. 71-2.

Next, a value of $\hat{\sigma}_{\text{calc}}$ was calculated for each point along the fuel element from the observed values of r . These values are also listed in Table I. Finally the "thermal contribution" curve was corrected by multiplying each point by the ratio $\hat{\sigma}_{\text{calc}}/41.17$. The final curve is plotted in Fig. 71-2. This plot represents the true thermal neutron flux along the length of the fuel elements.

In Fig. 71-1, the data as obtained are plotted and then smooth curves drawn through the points for both wires. The curve labelled "Thermal" is obtained by subtracting the smoothed episcadmium contribution curve from the smoothed total contribution curve. This of course is somewhat of an approximation since the two wires were in diametrically opposed positions in the core. It was assumed that the flux distribution in the two positions was similar.

The curve labelled, "Total Contribution" represents the activa-

Table II - Flux Data

Distance from top of fuel plate, inches	Curve 1 minus Curve 2 ($\times 10^{-12}$)	Cd ratio	r	ϕ calc barns	ϕ ($\times 10^{-12}$) n/cm ² sec
0	0.47	10.4	0.054	39.29	0.49
2	0.71	15.2	0.036	38.85	0.75
4	0.64	10.0	0.056	39.74	0.67
6	0.48	5.8	0.101	41.59	0.48
8	0.55	4.7	0.129	42.96	0.53
10	0.75	4.3	0.143	43.88	0.71
12	0.95	4.2	0.147	43.88	0.89
14	1.18	3.8	0.166	44.57	1.09
16	1.50	3.7	0.174	<div style="display: flex; align-items: center; justify-content: center;"><div style="margin-right: 5px;">↑</div><div style="border-top: 1px solid black; width: 100%;"></div><div style="margin-left: 5px;">↓</div></div>	1.38
18	1.80	3.6	0.177		1.66
20	2.14	3.6	0.177		1.97
22	2.56	3.7	0.174		2.36
24	3.00	3.9	0.161		2.76
26	3.35	3.8	0.166		3.08
28	3.56	3.7	0.174		3.28
30	3.75	3.7	0.166	44.57	3.45
32	3.96	3.8	0.161	<div style="display: flex; align-items: center; justify-content: center;"><div style="margin-right: 5px;">↑</div><div style="border-top: 1px solid black; width: 100%;"></div><div style="margin-left: 5px;">↓</div></div>	3.64
34	4.07	3.9	0.161		3.74
36	4.14	3.9	0.156		3.81
38	4.13	4.0	0.156		3.80
40	3.95	4.0	0.156		3.63
42	3.64	3.7	0.174		3.35
44	3.01	3.8	0.166		2.77
46	2.37	3.6	0.177	44.57	2.18
48	1.96	4.1	0.151	43.88	1.84
50	2.45	7.2	0.080	40.66	2.47
52	2.55	13.8	0.040	38.85	2.70
54	1.64	12.5	0.044	38.85	1.74
55	1.30	10.8	0.051	39.29	1.37

Fig. 71-2. Values of "r".

Conclusions. The shape of the flux plot in Fig. 71-1 is much as would be expected. The maximum flux occurs at about 36 in. from the top (18 in. from the bottom) of the fuel plate, in agreement with γ -ray flux plots run after shutdown. The peaking of the thermal neutron flux at either end of the fuel element is due to the effect of the reflectors.

The maximum value of the thermal neutron flux at this position at the edge of the core is $3.8 \pm 0.3 \times 10^{12}$ neutrons $\text{cm}^{-2} \text{sec}^{-1}$. No attempt will be made in this report to extrapolate this value over the remainder of the core.

From an examination of the plot of r vs distance (Fig. 71-2) it was hoped that an estimate of the point of inception of boiling could be determined. Theoretically, in the region away from the reflector, r should be constant and then should increase, because of less moderation, as boiling started and voids increased. Unfortunately Fig. 71-2 shows the data are not precise enough to give the required effect and within experimental error, r is constant over the central 28 in. of the fuel element.

Other factors contributing to the lack of effect are the fact that an unenriched element at the edge of the core was used. Thus, voids would not be great anyway and the fuel element is not surrounded by others in which boiling is occurring.

By graphical integration of the curve in Fig. 71-1, the average flux was found to be 2.22×10^{-12} . Thus, the maximum to average ratio is 1.72.

RELATIVE EFFECTIVENESS OF Sb-Be SOURCE AND CORE-INDUCED NEUTRONS

(Test Reports No. 102, 102A and 102B, by A. Hirsch)

In order to do critical experiments at low power levels, it was necessary to establish high efficiency counting channels and to reduce the source strength

Procedure The two fission counters and their associate channels used for the core reloading were re-established. They were installed in the core diametrically opposite the source, counter no. 1 in the 0-90 location and counter no. 2 in the 0-112 location. It was necessary to remove the stainless steel dummies from these locations. The AID amplifiers and high voltage supplies were set as follows.

	<u>Function</u>	<u>Bandwidth</u>	<u>Gain</u>	<u>PHS</u>	<u>HV</u>
No. 1	USE	.5 mc	64 x 1	170	500
No. 2	USE	.5 mc	64 x 1	200	500

Data as taken. (Counts reduced to c/m) shown in Table I.

Table I - Counting Data

Condition	No. 1	No. 2
Source in place	3073 ± 29	3284 ± 31
Tool on antimony	3070 ± 55	3262 ± 57
Removed antimony	1581 ± 40	1780 ± 42
Reinsert antimony	3083 ± 55	3243 ± 57
Removed antimony	1566 ± 40	1647 ± 41
Reinsert antimony		
Moved source 2/3 out	1466 ± 27	1533 ± 28
Moved source full out	1438 ± 38	1505 ± 39

A check was made with an oscilloscope to observe the character of pulses. They appear to be neutron pulses. A pulse height plateau was taken with the source removed. (See Fig. 102-1.)

Pulse height of the counter was reset as follows:

No. 1 PHS 170 No. 2 PHS 200

Table II - Counting Data

Condition	No. 1	No. 2
Source full out	1497 ± 38	1511 ± 39
Source raised to top of shroud	1460 ± 38	1444 ± 38
Antimony in dummy 0-333. Source Be at top of shroud	1408 ± 37	1504 ± 39
Source re-established in original position	3150 ± 40	3269 ± 41

Counting rate of plant counters is negligible.

Fig. 102-1. Counts per minute vs PHS.

Conclusions. The two counter channels are counting neutrons. Their sensitivity is many-fold over the plant BF³ counters. They will probably jam if a critical is run.

The source induces only a doubling of the background neutron count. A study of the photo fission and spontaneous fission of the core is in order. These data were taken after 170 Mw hours of operation and approximately 70 hours of decay time.

Source Level at Startup Counters and Sensitivity of Counters. Since the reloading of the core, the low counting rate of the plant BF^3 counters had been a disturbing condition. Though the counters were checked out prior to the loading, it was thought that their efficiency might have degenerated since the original installation in 1956. If the counting efficiency has not degenerated, then the source strength must have decayed appreciably.

With the rods banked at 6 in., the plugs were removed from the no. 1 counter hole. A Health Physics neutron counter was placed into the hole next to the no. 1 counter. Its count rate was negligible. Next, a Po-Be neutron source was put into the counter hole to present the same flux field to both counters. A counting rate was taken with the source in place and then a longer count with the source removed.

Table III - Counter Data

	Counter No. 1	Health Physics Counter
Source in	8950 counts/1 min	2300 counts/1 min
Source removed	20 counts/5 min 4 c/m	1 count /5 min .2 c/m
Ratio of sensitivity of $\frac{\#1}{\text{H.P.}} = \frac{8950}{2300} = 3.89$		
Sensitivity of H.P. counter = 134 c/m/nv		
Sensitivity of no. 1 counter = 521 c/m/nv		

The 4 c/m of the no. 1 counter with the source removed is certainly on upper limit. Usually the counting rate is less. Using 4 c/m and the 521 c/m/nv sensitivity, an upper limit of:

$$\frac{4 \text{ c/m}}{521 \text{ c/m/nv}} = 0.0077 \text{ nv in the hole,}$$

and due to the source, can be set.

A new replacement counter tube was purchased. A check of the relative sensitivity of no. 1 to the new tube was made, using the H.P. tube as a standard. The Po-Be source was used to establish the flux.

New tube	4913 c/m
H.P. tube	1594 c/m
Ratio of sensitivity	$\frac{4913}{1594} = 3.14$

Hence, the sensitivity of the new tube was less than the sensitivity of no. 1, and no gain would be made by using it.

Conclusion The plant startup counter channels were working. The sensitivity of the no. 1 counter was normal. The source flux was reduced from the original level by a factor of 10^3 .

Instrumentation for Startup. During Criticals 57-8-2 and 57-8-3, data were taken on various instruments to observe the approach to criticality and to note the information available to the operators. From these data it is also possible to assign flux levels at the various instruments.

Instruments available:

1. Westinghouse compensated ionization chamber in a thimble set into core location 0-16. High voltage + 300 volts, compensation -67 $\frac{1}{2}$ volts. Collector output to Keithley micro-microammeter. Output recorded on Brown recorder. This channel is referred to as "in-core." Manufacturer states a sensitivity of 4×10^{-14} amps/nv.
2. The panel mounted Keithley instrument connected to the Linear Flux Recorder. This instrument is referred to as the "Keithley" ANL, and has a chamber sensitivity of 10^{-14} amps/nv.
3. Plant startup counters no. 1 and no. 2. Sensitivity of no. 1 is set at approximately 520 cm/nv; no. 2 sensitivity is slightly higher.
4. The panel mounted L & N micro-microammeter. Chamber sensitivity is about the same as the "Keithley."

Since the in-core instrument would not be in use for startup at pressure, the operators were not permitted to use this instrument. It was used to monitor the startup so that no dangerous conditions could go undetected.

Figure 102-2 shows the plot of in-core chamber current and 1 over the sum of the counting rates of no. 1 and no. 2 counters versus the rod position. From the curve of $1/\Sigma$ cr, it was possible for the operators to assign the critical rod position within 0.1 in. while still three in. below critical. Plotting throughout the approach to critical reduced the error. After critical was reached a change in rod position to 0.5 in. more than critical gave an approximate 35 second period. The in-core instrument showed multiplication nicely, but did not offer any additional needed information.

Readings of chambers and counters were taken at about 200 w. From the data taken and sensitivity figures, it is possible to evaluate flux levels.

In-core

$$\frac{.372 \times 10 \times 10^{-5} \text{ amps}}{4 \times 10^{-14} \text{ amps/nv}} = 9.3 \times 10^8 \text{ nv}$$

Keithley

$$\frac{.758 \times 10 \times 10^{-10} \text{ amps}}{2 \times 10^{-14} \text{ amps/nv}} = 3.76 \times 10^4 \text{ nv}$$

L & N micro-microammeter

$$\frac{.55 \times 2 \times 10^{-9} \text{ amps}}{2 \times 10^{-14} \text{ amps/nv}} = 5.5 \times 10^4 \text{ nv}$$

Counter No. 1

$$\frac{745.967 \text{ c/m}}{521 \text{ c/m/nv}} = 1.43 \times 10^3 \text{ nv}$$

(The counter has undoubtedly exceeded its limit of resolution.)

Fig. 102-2

In-core chamber current
vs rod position for
criticals 57-8-2 and 3.

Conclusions. Though the source strength is low, a safe approach to critical can be made using the plant counters. The difficult task of establishing in-core instruments for criticals at 600 psi could hardly be justified, since the approach to critical is safe. An effort is being made to irradiate a stronger source. This will reduce the statistical error of the counting rates, but will not offer any other great advantage.

SECTION IV CONTROL AND CONTROL RODS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
2	Graphs of Rod Position Versus Time of Operation, J. O. Schuyler	235
9, 9B	Control Rod Positions and Power Calibration, J. A. DeShong	236
31, 42D, 77	Control Rod Data, A. J. Lizasoain, J. A. Thie, J. F. Pohlman	243

GRAPHS OF ROD POSITION VERSUS TIME OF OPERATION

(Test Report No. 2, by J. O. Schuyler)

Figures 2-1, 2-2 and 2-3 are plots of rod positions versus time of operation from February to April, 1957.

Fig. 2-2. Rod position vs time of operation

Fig. 2-3. Rod position vs time of operation

CONTROL ROD POSITIONS AND POWER CALIBRATION

(Test Reports No. 9, 9B, by C. B. Zitek and J. A. DeShong, Jr.)

Positions and Power Levels. While the reactor was being operated to obtain additional data to evaluate the condenser performance, evaluate feedwater pump performance, and obtain radiation levels at various power levels, the control rod position data at decreasing reactor power levels was obtained and is presented in this report. The conditions under which the reactor was operating were normal:

1. Four corner rods (2, 4, 6, and 8) at 47 in.
2. No. 2 reactor water purification system in service at about 3 to 4 gpm.
3. Decreasing condensate temperature.
4. Turbine not in service, all steam being by-passed to the condenser.
5. Feedwater temperature not constant.

The reactor had just been brought up to a 20 Mw power level after a 25 hour shutdown period. The xenon concentration was in a transient condition and this rod position data cannot be used unless corrected for current xenon concentration.

The data obtained during this run is given in the attached table and graphs. (Table I, Figs. 9-1, -4.)

As shown in Fig. 9-1, the step decrease in rod position is not exactly linear. The change from 13.3 Mw to 10 Mw was only 0.72 in. compared to an average 0.98 in. This difference was due primarily to changing the Po setting to reduce reactor pressure from 620 psig to 600 psig when the power level was changed. There was no visible change in reactor pressure on the TV pressure gauge during the two previous decreases even though the gauge was tapped locally. The rod worth over this position range should be about constant; however, the effect of xenon buildup should be taken into consideration, but would be quite small.

Using the two end points for the straight line, the extrapolated control rod position at zero power would be 17.15 in. The worth of the 5.65 in. of the five hafnium rods at this position in the core should be about 2.0 to 2.5 percent Δk according to nine rod bank operation at 20 Mw and the calibration curve.

Table I-CONTROL ROD POSITION vs VARIOUS REACTOR POWER LEVEL DATA

Time	Reactor Power Level Mw	Steam Flow lbs/hr	By-pass Steam Flow lbs/hr	Feedwater		Condensate Temp. F	Cooling Water Temp. F	Control Rod Positions						Neutron Flux Instruments				
				Flow lbs/hr	Temp. F			2-4 6&8		1	3	5	7	9	Micro-micro- ammeter		Galvanometer	Log Flux Recorder
Date: 4-11-57														Scale		Scale		
1700-1800	20.0	60,000	60,000	61,000	118	97	61	47	23	23	23	22.75	22	.495	5x10 ⁻⁵	22	50	88
1810-1900	16.6	50,000	50,000	50,000	116	91	59	47	22	22	21.5	21.5	21.5	.410	5x10 ⁻⁵	18	50	75
1910-2000	13.3	40,000	40,000	40,200	113	84	57	47	21	21	20.8	20.5	20.5	.33	5x10 ⁻⁵	14.2	50	60
2010-2100	10.0	30,000	30,000	31,000	112	73	52	47	20	20	20	20	20	.26	5x10 ⁻⁵	10.9	50	50
2110-2200	6.6	20,000	20,000	20,500	112	62	47	47	19	19	19	19	19.5	.185 .415	5x10 ⁻⁵ 2x10 ⁻⁵	7.4	50	38
2210-2300	3.3	10,000	10,000	9,500	125 avg.	51-47	40	47	18	18	18	18	18.5	.105 .230	5x10 ⁻⁵ 2x10 ⁻⁵	3.7	50	21
2300-2315	3.3	10,000	10,000	9,500	63	47	39	47	18	18	18	18	18.5	.097	5x10 ⁻⁵	3.7	50	21
2315-2330	6.6	20,000	20,000	20,000	70	55	39	47	19.5	19.5	19.5	19.5	19.5	.185	5x10 ⁻⁵	7.6	50	39
2330-2345	10.0	30,000	30,000	30,000	78	66	43	47	20.5	20.5	20.5	20.5	21.0	.25	5x10 ⁻⁵	10.8	50	51
2345-2400	20.0	60,000	60,000	60,000	97	90	50	47	23.5	23.5	23.5	23.5	24.0	.50	5x10 ⁻⁵	22.4	50	89

Fig. 9-1. Control rod position vs reactor power level

Fig. 9-2. Neutron flux reading

Fig. 9-3. Calculated xenon curve

Fig. 9-4. Control rod calibration with equilibrium xenon

The feedwater temperature changed after each change in power level due to decreasing cooling water temperature and decrease in flow with a constant clean-up flow at about 306°F. After each change in power level the feedwater temperature overshoot its final equilibrium temperature in increasing amounts until at 3.3 Mw operation the temperature oscillated with about a 20°F spread.

The overshooting of equilibrium temperature was due to the lag in feedwater response in the newly initiated three element water level control. The steam flow is the major control and when it changed the lag in feedwater control response permitted an increase in reactor water level which then called for a decrease in feedwater flow over and above that called for by the steam flow signal.

The final oscillating temperature condition at 3.3 Mw could have been due to two factors:

1. Oscillation of feedwater flow which had a greater amplitude at 3.3 Mw than at the higher power levels (50% vs 6.6%).
2. Fluctuation of clean-up loop flow back to the reactor.

At the time the effect of the feedwater flow variations was not thought to be large enough and the frequencies were not the same. Therefore, the purification loop was shut off. The oscillations stopped and the feedwater temperature decreased from an average 125°F to 63°F. Actually, both factors were responsible for the condition.

With this large change in feedwater temperature, an increase in reactor power output was expected. However, there was no indication of an increase, if anything, the neutron flux recorder showed a slight decrease from 0.105 to 0.097 (5×10^5). With the low flow (20 gpm) the effective sub-cooling would be small.

The power level was increased in the same step changes to check control rod positions. Five hours had elapsed since the start of the run at 20 Mw and the xenon buildup would have changed rod positions at the various power levels, therefore, a complete set of data was not obtained.

The neutron flux instrument readings were plotted and Fig. 9-2 shows the galvanometer to exhibit better linearity than the micro-microammeter except in the higher range.

It appears advisable to repeat this test under more controlled conditions to get more and better data for cross checking the worth of control rods (bank of nine vs bank of inner five). Two runs should be made, one with low xenon concentration and another at equilibrium xenon. Both runs should be made with no purification system flow and with flow to show the effect of decreased feedwater temperature at each reactor power level.

Rod Power Calibration With Equilibrium Xenon. The objective of this test was to determine not only the rod power calibration, but also the percent k in voids. Table II presents the data taken. It was necessary to correct for xenon buildup, because of elapsed time between readings. Figure 9-3 presents the theoretical buildup, together with an adjusted curve, which was arrived at from the data in Table II.

Figure 9-4 shows the calibration for both 9 and 5 rod banks. The 5 rod data were taken for correlation purposes only, since k calibrations are available on only the 9 rod bank. The 5 rod worth, at least through the region shown, may be obtained by adding about 2.7 in. to the 5 rod reading, and then looking up the 9 rod calibration for this value. The k values are shown on the right hand ordinate. Note that the k in voids at 20 Mw is approximately 1.7 percent when obtained in this way.

Table III presents data taken for the purpose of determining the most effective rod position for the rod oscillator, which was designed to operate at a nominal position of 10 to 13 in., with adjustable peak deviation from 0 to 2 in.

The 13 in. nominal position gives the greatest change, and a 17 in. position probably would give even more for the full xenon case which existed when these data were taken. Since the oscillator will be used under many conditions where low xenon is present, the design will not be altered. Note that the center rod is worth about 0.09 percent k /inch and about 5.0 percent w /inch, while the no. 5 rod is worth about 0.06 percent k /inch and about 3.3 percent w /inch.

Table 11

										Test Quantities						
Time	Bank	Feedwater Temp.	Total Steam Flow	Bypass Steam Flow	Feedwater Flow	Temp. in °F	Condensate Outlet Temp. in °F	Condensate Inlet Temp. in °F	Level	Micro-microammeter		Galvanometer		P Recorder	P Console	P TV
										Reading	Mult.	Reading	Mult.			
1:53	9 Rods	127	60,000	55,500	59,000	464	92	80	0	0.19	5×10^{-7}	20.6	50	570	582	600
1:36	5 Rods	127	60,000	55,500	59,500	465	92	79	0	0.5	5×10^{-7}	22.2	50	570	582	600
2:00	9 Rods	122	40,200	36,500	39,500	465	89	80	0	0.725	5×10^{-7}	13.5	50	565	578	590
2:15	9 Rods	120	19,000	16,000	18,200	466	81	76	0	0.67	1×10^{-6}	6.0	50	559	570	582
2:30	9 Rods	122	8,500	4,500	11,000	466	76	74	+0.1"	0.36	1×10^{-7}	3.2	50	555	570	582
2:49	5 Rods	123	8,500	4,000	8,000	466	74	71	+0.5'	0.39	1×10^{-7}	3.45	50	555	570	582
3:07	9 Rods	123	18,800	15,000	19,500	464	76	71	-0.1'	0.36	2×10^{-7}	6.3	50	558	571	584
3:26	9 Rods	123	40,000	36,400	37,000	463	82	74	0	0.305	5×10^{-7}	13.3	50	563	578	590
3:40	9 Rods	125	60,000	56,000	59,500	465	88	76	0	0.49	5×10^{-7}	20.8	50	567	581	600
3:47	9 Rods	125	60,000	56,000	57,500	457	91	78	0	0.46	5×10^{-7}	20.5	50	555	565	580

Control Rod Position

Total Flow	1		2		3		4		5		6		7		8		9	
	5	9	5	9	5	9	5	9	5	9	5	9	5	9	5	9	5	9
60,000	27.3	29.8	47	29.8	27.3	29.8	47	29.8	27.3	29.8	47	29.8	27.3	29.8	47	29.8	27.3	29.8
40,200		27.5		27.5		27.5		27.5		27.5		27.5		27.5		27.5		27.5
19,000		25.1		25.1		25.1		25.1		25.1		25.1		25.1		25.1		25.1
8,500	21.5	24.0	47	24.0	21.5	24.0	47	24.0	21.5	24.0	47	24.0	21.5	24.0	47	24.0	21.5	24.0
18,800		25.6		25.6		25.6		25.6		25.6		25.6		25.6		25.6		25.6
40,000		28.2		28.2		28.2		28.2		28.2		28.2		28.2		28.2		28.2
60,000		30.6		30.6		30.6		30.6		30.6		30.6		30.6		30.6		30.6
60,000		30.4		30.4		30.4		30.4		30.4		30.4		30.4		30.4		30.4

Table III

	All rods in bank at 20-Mw	Rod no. 5 position with rod nos. 1, 2, 3, 4, 6, 7, 8, 9 in bank at 20 Mw				Center rod position with rod nos. 1, 2, 3, 4, 5, 6, 7, 8 in bank at 20 Mw	
		14 in.	12 in.	11 in.	9 in.	14 in.	12 in.
Galvanometer	21.4	22.8	21.2	23.3	22.2	25.4	22.7
L & N	.485	.53	.500	.555	.525	.565	.505
Reactor steam	60,000	60,000	56,000	60,000	57,100	60,000	54,000
Bypass steam	30,000	30,000	26,000	30,000	27,000	30,000	24,100
Rod bank position, in.	30.2	35.2	35.2	36.3	36.3	40.5	40.5
<u>Reactor Pressure</u>							
TV	600	600	600	600	600	-	-
Console	585	585	585	585	585	585	585

CONTROL ROD DATA

(Test Reports No. 31, 42D, 77, by A. J. Lizasoain, J. A. Thie and J. F. Pohlman)

Rod Drop Time Versus Reactor Steam Pressure. Figure 31-1 shows variation of rod drop time with reactor steam pressure for a boron stainless steel and a hafnium rod.

Fig. 31-1. Rod drop time vs steam pressure

Control Rod Calibration. The December, 1958 control rod calibration curve, Fig. 42D-1, from loading 46 experiments, should remain valid until the rods are changed or spikes are loaded into the core. The curve shows the differential control rod worth for core loading number 46 at room temperatures.

Fig. 42D-1. Control rod curve

Control Rod Activity. The gamma activity of the control rods in the unloaded core was measured. The gamma probe used was the same one previously employed in gamma-flux measurements in the EBWR core. (Test Reports 49, 49A).

Rod no. 2 (boron; corner position)	3,300 r/hr	} ± 20 percent
Rod no. 3 (hafnium)	8,500 r/hr	
Rod no. 9 (hafnium; center position)	15,000 r/hr	

The maximum occurred as a broad plateau extending over the middle half of the absorber section. No attempt was made to measure the Zircaloy follower.

Since the probe cannot be relied upon for accurate measurement in terms of r/hr, due to a sensitivity affected by radiation, one of the elements (H-7) in the storage pit was probed as a check on the control rod data. This element exhibited a maximum of 20,000 r/hr which is quite reasonable for an element in an outer edge position (27-113) in the core in the light of past measurements.

Thus, the center rod is about as radioactive as one of the outer fuel elements, and the corner rods are about one-fifth as radioactive.

SECTION V PHYSICS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
41C	EBWR Void Reactivity Versus Power, J. A. DeShong, W. C Lipinski	247
42, 42B, 42C	Reactivity Changes and Irradiation Effects, D. E. McFeron, J. A. Thie	247
100C	Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth, K. K. Almenas	256

EBWR VOID REACTIVITY VERSUS POWER

(Test Report No 41C by J. A. DeShong and W. C Lipinski)

Figure 41C-1 shows the variation of void reactivity with power based on control rod position data.

Fig. 41C-1. EBWR void reactivity vs power

REACTIVITY CHANGES AND IRRADIATION EFFECTS

(Test Reports 42, 42B, 42C, by D. E McFeron, and J. A. Thie)

Reactivity Changes in Core Loading No. 47. Scheduled changes¹ in the core loading of the EBWR constituting loading No. 47 were completed. The purpose of these changes was to provide more reactivity permitting higher power runs with operation at 35 Mw or higher continuously, and to allow a detailed inspection and analysis of a central natural element.

The above objectives were accomplished by replacing two central thin natural uranium elements with thin enriched elements. The natural thin elements were transferred to the storage pit. Critical tests were run before and after the loading change to determine the effect on reactivity. Both tests were carried out with the reactor at atmospheric pressure and temperature (69°F) and in the absence of soluble poisons or Xe. The first shutdown from 20 Mw(th) operation and the last test was performed 25 days later.

¹W. J. Kann, Private Communication

A summary of the results of these critical tests is given below:

Date	Critical test no.	Rod period position (9 rod bank)	Integral rod worth % ρ	Period, seconds	% ρ from inhour curve	ρ , %
1/2/58	46-53	14.11 in.	5.40	27	.176	5.224
1/27/58	47-1	12.64 in.	6.17	26	.180	5.990
						0.766%

Thus the change in core loading resulted in an increase of 0.766 per-cent ρ . This compares quite favorably with an estimated¹ increase of $\frac{3}{4}$ percent ρ . Most of the change (0.712%) occurred in replacing two natural thin fuel elements at central core locations (129-53, 129-264) with enriched thin fuel elements transferred from the outer periphery of the core. The addition of two fresh, natural, thick fuel elements to core locations (27-43, 27-255) resulted in the remainder of the reactivity increase (0.054% ρ).

Effect of Plutonium on EBWR Kinetics. During the operation of a thermal reactor such as the EBWR which is initially fuelled with slightly enriched uranium, a conversion of U^{238} to Pu^{239} and other isotopes takes place. Because the delayed neutron fraction, which has a major effect on thermal reactor kinetics, is quite different for Pu^{239} as compared with U^{235} and U^{238} , the transient behaviour of the reactor changes as irradiation of the fuel proceeds. Experimental evidence of the effects of these changes has been noticed in routine physics tests performed on the EBWR at various operating times.²

Discussion. The delayed neutron parameters for six delayed groups from thermal fission of U^{235} and Pu^{239} and from fast fission of U^{238} are shown in Table I.

Table I. - Delayed Neutron Parameters for Thermal Fission³ of U^{235} and Pu^{239} and Fast Fission of U^{238}

Group index	U^{235} (thermal)		Pu^{239} (thermal)		U^{238} (fast)	
	β_i	λ_i	β_i	λ_i	β_i	λ_i
1	.000210	.01244	.000072	.01277	.0002061	.01323
2	.001401	.03051	.000625	.03008	.0021526	.03212
3	.001253	.11144	.0004433	.12378	.0025457	.13863
4	.002526	.30137	.0006838	.32542	.0061029	.35915
5	.000737	1.13631	.0001787	1.12160	.0035381	1.41459
6	.000267	3.01370	.0000928	2.69708	.0011794	4.02994
$\Sigma_i \beta_i =$	.006396		.0020956		.01572	

¹J. A. Thie, Private Communication.

²EBWR Test Procedure No. 42, J. A. Thie, Dec. 9, 1957.

³ANL-5800 "Reactor Physics Constants," Section 1, Sept., 1958.

Because of the small yields under the irradiation conditions investigated, fast fission in U^{235} and in Pu^{240} or Pu^{241} have been neglected.

A quantity which is necessary in the calculation of weighted values of delayed neutron parameters in the reactor is F , the ratio of fast fissions to thermal fissions. An approximate expression for F is given by

$$F = 2(\epsilon - 1)$$

which is correct within 10 percent for $1 < \epsilon < 1.05$. It was assumed that the change in ϵ for reasonable short irradiations (in terms of total burnup) was negligible. The distribution of thermal fissions between the U^{235} and the Pu^{239} at various irradiation times was calculated and (the distribution of fissions between the three principal isotopes) is shown in Table II.

Table II - Distribution of Fissions in EBWR

Irradiation time, Mwd./element	Operating condition	Fast fission factor, k	F , fast fissions thermal fissions	n fission fraction in U^{238}	$(1-n) \frac{N^{25}}{N^{th}}$	$(1-n) \frac{N^{49}}{N^{th}}$
					fission fraction in U^{235}	fission fraction in Pu^{239}
0	Rm. temp.	1.0367n	0.0734	.0685	.9315	.0000
36.3	Rm. temp.	1.0367	0.0734	.0685	.8511	.0804
36.3	488°F 15% void	1.0453	0.0906	.0831	.8378	.0791
91.4	Rm. temp.	1.0367	0.0734	.0685	.7579	.1763

As shown in Eq. (1), the fission fraction values shown in Table II were used as weighting values of the delayed neutron parameters shown in Table I, in order to obtain values of these parameters at various irradiation times and operating conditions.

$$\rho_i = \beta_i^{25} (1-n) \left(\frac{N^{25}}{N^{th}} \right) + \beta_i^{49} (1-n) \left(\frac{N^{49}}{N^{th}} \right) + \beta_i^{28} n \quad (1)$$

where

$$n = \frac{F}{1+F}, \text{ ratio of fast fissions to total fissions}$$

$$N^{th} = N^{25} + N^{49} \text{ No. of thermal fissions}$$

$$N^{49} = \text{No. of thermal fissions occurring in } Pu^{239}$$

$$N^{25} = \text{No. of thermal fissions occurring in } U^{235}$$

$$\beta_i(-) = \text{Absolute yield of delayed neutrons for isotope } (-) \text{ in the } i^{\text{th}} \text{ group.}$$

A similar expression was used for evaluating λ_i .

Results of these calculations for the irradiation times and operating conditions considered are shown in Table III. In addition, similar quantities used in the calculation of the original EBWR inhour curve are shown for reference. The marked discrepancies between these values and those computed for the zero irradiation time condition in this study results from the use of different values^{1,2} for the delayed neutron yields and decay constants.

Table III - Weighted Values of Delayed Neutron Parameters for EBWR

Irradiation time, Mwd/element	Operating condition	$\bar{\beta}_i$	λ_i
0*	Room temp.	.0002487	.01247
		.001713	.03154
		.002181	.1536
		.002716	.4568
		.001065	1.613
		.000324	14.28
		$\Sigma \bar{\beta}_i = .0082477$	
0	Room temp.	.002097	.01249
		.0014525	.03067
		.0013434	.11330
		.0027710	.30533
		.0009289	1.15537
		.0003295	3.08331
		$\Sigma \bar{\beta}_i = .007035$	
36.3	Room temp.	.001986	.01252
		.0013901	.03059
		.0012782	.11429
		.0026229	.30726
		.0008840	1.1542
		.0003155	3.0578
		$\Sigma \bar{\beta}_i = .0066893$	
36.3	488°F + 15% void	.001988	.01253
		.0014021	.03061
		.0012980	.11467
		.0026775	.30807
		.0009256	1.15827
		.0003290	3.0731
		$\Sigma \bar{\beta}_i = .006831$	
91.4	Room temp.	.0001858	.01255
		.0013178	.03055
		.0012025	.11544
		.0024512	.30950
		.0008320	1.15281
		.0002993	3.02835
		$\Sigma \bar{\beta}_i = .0062886$	

*Values used in the calculations of the EBWR-1 inhour curve

¹ANL 5800. op. cit.

²D. J. Hughes et al. "Delayed Neutrons from Fission of U²³⁵,"
Phys. Rev. 73, 111 (1948).

In obtaining the proper value of the delayed neutron fraction, β , one usually includes a term to account for the difference in fast leakage for delayed neutrons as compared with prompt fission neutrons. It has been assumed that τ delayed $\approx \frac{2}{3}\tau$ prompt and thus the effective value of β is increased somewhat. In the present calculation a value of ϵ was chosen which corresponds to a core loading consisting of 50 percent thick fuel elements and 50 percent thin elements uniformly mixed. Since ϵ is lower for thin elements, the effect would be to lower the value of $F = 2(\epsilon - 1)$ for a loading of thin fuel elements which would decrease the relative fissions in U^{238} and thus decrease β . These effects are small, of opposite sign, and approximately cancel each other for a thin core. The values computed herein thus apply most directly to a core loading consisting of thin fuel elements entirely. The present loading (No. 49) consists of 56 thin elements in the central region of the core where their effect is greatest.

In addition to a change in the delayed neutron fractions with irradiation, the prompt neutron lifetime also changes. This is primarily caused by an increase in the reactor absorption cross section as fission products build in, U^{235} burns out, etc. As a zero order approximation, the expression for the prompt neutron lifetime is:

$$\ell_t = \frac{1}{\Sigma_{at} v(1 + L_t^2 B^2)}$$

where

Σ_{at} is the reactor averaged macroscopic absorption at time t

v is the average neutron velocity at the conditions specified for Σ_{at}

L_t^2 is the thermal diffusion area at time t_0

B^2 is the buckling.

In the original calculation for the prompt neutron lifetime for the EBWR, a value of $\ell_0 = 47.1 \times 10^{-6}$ seconds was obtained as the zero order approximation. An importance function correction to this calculation yielded a prompt neutron life time, $\ell'_0 = 62.6 \times 10^{-6}$ seconds. Since this correction value remains relatively unchanged for this reactor during irradiation, it was applied as a normalization factor to the calculated zero order prompt neutron lifetimes. The various values calculated are shown in Table IV.

Table IV - Prompt Neutron Lifetime in EBWR

Irradiation time, Mwd/element	Operating condition	$\ell \times 10^6$ seconds
0	Room temperature	62.6
36.3	Room temperature	60.3
36.3	488°F + 15% void	67.4
91.4	Room temperature	58.1

The reactivity-inhour relationship (inhour curve) was calculated for the different conditions listed by means of the RE-130 code¹ on the IBM 704 computer.

$$\rho = \frac{\ell}{\ell + T} + \frac{T}{\ell + T} \sum_{i=1}^m \frac{\bar{\beta}_i}{1 + \bar{\lambda}_i T}$$

where

ρ is the reactivity

ℓ is the prompt neutron lifetime

T is the desired period

$\bar{\beta}_i$ is obtained as indicated in Eq. (1)

$\bar{\lambda}_i$ is obtained by an equation similar to Eq. (1)

Results. The results of the computer calculations for the conditions investigated and for periods of interest are shown in Table V.

Table V - Inhour Calculations for the EBWR Reactivity

Period, seconds	Reactivity				
	0*	0	0	36.3	91.4
	room temp.	room temp.	room temp	488°F 15%V	room temp.
1.00	6.1522-03	5.5199-03	5.2466-03	5.3448-03	4.9307-03
2.00	5.2359-03	4.7466-03	4.5102-03	4.5864-03	4.3272-03
4.00	4.2337-03	3.8712-03	3.6778-03	3.7336-03	3.4547-03
8.00	3.2433-03	2.9783-03	2.8298-03	2.8682-03	2.6587-03
16.00	2.3485-03	2.1562-03	2.0495-03	2.0744-03	1.9265-03
32.00	1.5958-03	1.4609-03	1.3892-03	1.4045-03	1.3067-03
64.00	1.0086-03	9.2005-04	8.7525-04	8.8410-04	8.2369-04

*Original EBWR data

An examination of Table V discloses that as irradiation proceeds a given period is produced by less reactivity change. Or, putting it another way, a given reactivity change causes a shorter period. Primarily this effect is caused by the decrease in the effective delayed neutron fraction. The decrease in prompt neutron lifetime as irradiation proceeds also contributed to this effect although the contribution is slight for the periods of interest here. For periods less than 0.1 second, a shorter prompt neutron lifetime leads to significantly shorter periods for a given reactivity increase.

¹C. Cohn, "IBM-704 Codes for Reactivity Step Calculations," ZPR-V Memo 15. Q. B. Toppel, ANL Internal Memo, April 16, 1958.

For some time it has been noticed that room temperature calibrations of control rod worths which are made during the physics tests performed periodically on the EBWR¹ show values of differential control rod worths which are increasingly greater than the original control rod calibration as irradiation proceeds. This result is obtained using the original, no plutonium, inhour curve to convert from the measured period to reactivity. Actually, if the delayed neutron fractions and prompt lifetime did not change with irradiation it would be expected that the control rod worth would decrease slightly during irradiation due to burnup of the control rods and decreased absorption of the control rods relative to the entire core. Rod burnup should not be too significant thus far in the EBWR due to relatively short exposure (about 5000 Mwd on the reactor) and the fact that five control rods are composed of hafnium with the other four being 2 percent boron in stainless steel. That the net observed effect of all of these changes is an observed apparent increase in control rod worth as irradiation takes place demonstrates the greater relative contribution of the delayed neutron effect.

The delayed neutron parameters shown in Table III for the original EBWR inhour curve are in sharp disagreement with values obtained in this study using similar calculations but more recent data² for delayed neutron yields. However, the inhour curve based on these older values was in ex-

Fig. 42-1. Effect of irradiation on the kinetic equation.

cellent agreement with rod calibration tests using known boric acid additions. For this reason the values calculated here have been normalized to these earlier values in order to provide better agreement with experiments. No explanation for this discrepancy has yet been obtained. A graph showing the correction factor to be applied to the original (unirradiated) EBWR inhour curve to obtain a reactivity worth for a given measured period in the range of 10-30 seconds is shown in

Fig. 42-1. Reactivity values ob-

tained with this correction applied, when used to obtain differential control rod worths, agree within ± 3 percent for values obtained to date (4630 Mwd irradiation).

Conclusions. The effect of plutonium buildup on the reactor kinetics during the irradiation of a low enrichment uranium fuel has been estimated.

¹EBWR Test Procedure No. 42, op. cit.

²ANL 5800, op. cit.

Both the magnitude and the direction of the estimated effect agree with experimental values obtained in the EBWR after irradiation for various times. It appears as if this change in reactivity effect, as determined by changes in the differential control rod worth, is a fairly sensitive indication of uranium conversion to plutonium in this reactor. This conclusion includes an assumption that the burnup of the control rods is negligible during the irradiation period observed.

Irradiation Effects - Reactivity Ion. Table VI shows the results of room temperature criticals during the reactor life. The reactivity due to burnup was obtained by correcting the overall reactivity loss for the individual fluctuations due to minor loading changes.

Table VI - Reactivity Loss at Room Temperatures Due to Samarium, Fission Products, U^{235} Loss, and Pu Buildup

Loading No.	Mwd at end of loading's operation	% reactivity above loading 46 due to loading change	% burnup reactivity loss
46	0	0	0
46	410	0	.62
46	2120	0	1.4
46	3060	0	1.82
47	3856	.76	2.18
48	4629	.609	2.55
49	6667	.629	3.47
50(47)	6987	.76	3.63
51		.48	

The rate of reactivity loss is seen to be 0.46 percent per 1000 Mwd at room temperature. It is of more practical interest to know the rate of loss at 489°F. A single experiment at 6987 Mwd shows 2.93 percent has been lost at this temperature, ~7 percent less than at room temperature. The rate of loss at 489°F is believed to be 0.36 percent per 1000 Mwd. Presumably this shows the positive contribution of plutonium to temperature coefficient.

Reactivity in Voids. According to theory this quantity is more sensitive to control rod position than to burnup per se or minor loading changes. Table VII summarizes results obtained in this quantity, defined as the reactivity present in a period if all the voids should collapse. The control rod position effect was explored exactly by water boration during early experiments,¹ but was confirmed approximately by equilibrium xenon with 20 Mw rod positions corrected for burnup during reactor life.

¹J. A. Thie. "EBWR Physics Experiments," ANL-5711 (July 1957).

Table VII. Reactivity in Voids

Conditions	Results
19.6 Mw, all rods at 29.8 in., ~ 0 Mwd, equilibrium xenon	2.2 %
13.5 Mw, all rods at 26 in., 6087 Mwd, no xenon	1.65%
20.4 Mw, all rods at 28.5 in., 6987 Mwd, no xenon	2.2 %
Reduction in reactivity in voids by all rods moving out one inch at constant power:	
By water boration experiments	.0278%
By burnup experiments	.036 ± .036%

Temperature Coefficient. Four definitions of the reactivity loss between room temperature, 68°F, and 489°F have been found useful:

1. Difference in all rods out reactivities at 68°F and 489°F.
2. The amount of subcriticality at 489°F with all rods at their 68°F critical position
3. The amount of supercriticality at 68°F with all rods at their 489°F critical position.
4. An integral of small reactivity differences measured by maintaining criticality with the rods at all points between 68°F and 489°F.

The contribution of plutonium toward reducing the magnitude of the temperature coefficient is seen in Table VIII. According to definition (1) the coefficient will be near zero in another 7000 Mwd. It should be pointed out, however, that it has been necessary to base these results on only one measurement at 6987 Mwd.

Table VIII - Percent Reactivity Loss from 68°F to 489°F

	1	2	3	4
At 0 Mwd	.91	3.13	2.41	2.94
At 6987 Mwd	.26	1.55	1.19	1.45

Value of the Dollar. For initial experiments¹ the dollar was taken to be .825 percent reactivity. Calculations show that the effect of (uniformly distributed) plutonium is to change this to .825 (1 - Mwd × 10⁻⁵) percent.

¹J. A. Thie, op. cit.

Although experiments at 4000 Mwd agreed with this calculation, results at 6987 Mwd indicate the dollar to be .825(1 - .13) percent. The non-uniform distribution of plutonium may account for the discrepancy. As with the falling temperature coefficient, the dollar deflation does not appear to offer any real hazard in the near future.

Core Life. In its design phase no particular lifetime was assigned to the EBWR's first core. Therefore its life cannot be compared to expectations. To predict core life from the above experiments it will be defined as the number of Mwd at which 20 Mw operation with equilibrium xenon occurs with all control rods out. The result is a life of

(percent reactivity added to Loading 51) $\times$ 2780 + 9200 megawatt days.

EFFECT OF U^{235} CONTAINING STAINLESS STEEL FOLLOWERS ON EBWR REACTIVITY AND CONTROL ROD WORTH

(Test Report No. 100C by K. K. Almenas)

Description: The Zircaloy followers of the initial control rod bank have been replaced by U^{235} containing stainless steel followers. Each follower is made up of four plates 5 in. wide, 36 in. long and $\frac{1}{4}$ in. thick which have lapping edges bent at an angle and seam welded to form a cross $\frac{1}{4}$ in. thick with 10 in. wide blades. Each plate contains 50 g U^{235} as UO_2 in a stainless steel matrix.

Reactivity Effect. Since stainless steel has an appreciable absorption cross section, the reactivity increasing effect of the U^{235} is overshadowed by the stainless steel and the net effect of the followers on reactivity is negative. Thus, the new control rods of the EBWR are essentially composed of two control materials of widely different strength. Since these two control materials are movable in the core, the question of the amount of reactivity absorbed by the control rod followers depends on the control rod, and the control rod follower position.

The theoretical upper limit of that reactivity would be reached with the control rod followers evenly distributed over the whole core. This is about 2.3 percent ρ since for the predominating thin enriched (TE)(ET) elements the addition of control rod followers results in the given β_{∞} :

$$\beta_{\infty}(ET) = 1.174$$

$$\beta_{\infty}(ET + \text{control followers}) = 1.147.$$

Practically, this limit cannot be reached since the followers are only 36 in. long and also do not cover the whole core radially.

A point of interest and one for which data naturally became available is the reactivity change at the natural poison free control rod position at 8800 Mwd burnup (the burnup when the control rods were changed). The 9 rod bank had to be moved from 18.7 in. (dial) to 24.05 in. to compensate for the additional absorption introduced by the followers. This corresponds to a reactivity change of 1.4 percent ρ .

Control Rod Worth. A 2-dimension PDQ calculation simulating these conditions was performed giving a theoretical reactivity change of 1.42 percent ρ . The effect of the control rods and the control rod followers were represented by "homogenized" cross sections over a 56.4 cm radial central area of the core.

The maximum practical reactivity change would be the integrated change in the control rod worth over the whole 48 inches of rod travel. This was determined experimentally for the central control rod and is shown in Fig. 100C-1. Though the two rods in question are of different materials (Hf and B-SS), their total worth is practically identical and thus the marked difference in the worth curves is due almost entirely to the effect of the stainless steel followers. Since the integrated worth of the central follower is 0.29 percent ρ over the whole 48 in. of rod travel and 0.192 percent ρ up to 24 in. we can use the experimental worth of the 9-follower bank at 24 in. to extrapolate it to the all out condition:

$$(.29) \left(\frac{1.4}{.192} \right) = 2.11\% \rho$$

Independently from this, the worth of the control rod followers at operating conditions can be obtained from some available criticality data. Since, depending on method of approach, a sizable discrepancy exists, the data and the corrections on it will be quoted in full.

Fig. 100C-1. Differential worth of center rod - based on $k = 6.27 \times 10^{-3}$.

The basis of comparison is the operating reactivity of loading No. 55 (loading after control rod changes having 8813 Mwd burnup). It was found that this loading was capable of 6.6 Mwd at equilibrium Xe with rods all out.

This amounts to an excess reactivity of:

$$\begin{aligned}\text{reactivity in Xe} &= (1.18)(.87) = 1.027\% \rho \\ \text{reactivity in voids} &= (.66)(.87) = \underline{.574} \\ \text{Total} &= 1.60 \% \rho\end{aligned}$$

where the .87 factor corrects for the high β used in initial control rod calibrations.

This can be compared to the last previous hot critical which was made at 8458 MSC burnup and 467°F. Conditions of the critical were:

$$\begin{aligned}\text{all rods at 21.52 in.} &= (1.3)(.87)(2.82) = 3.19\% \rho. \\ \text{reactivity in period} &= (.16)(.83)(.87) = .115\% \rho \\ \text{reactivity in Xe} &= \underline{.024\% \rho} \\ \text{Total} &= 3.329\% \rho\end{aligned}$$

where

1.3 is the rod temperature coefficient

2.82 is obtained from the rod calculation curve

.83 is the Pu factor.

This reactivity must be corrected for the 355 Mwd burnup and the 21°F temperature difference from operating temperature.

$$\text{Burnup correction} = \left[-.36\% \frac{\% \rho}{1000 \text{ Mwd}} \right] 355 = -.128\% \rho$$

$$\text{Temp. correction} = \left[-.0105 \frac{\% \rho}{^\circ\text{F}} \right] 21^\circ\text{F} = -.22\% \rho$$

Therefore the total corrected reactivity equals 2.98 percent ρ , giving an operating follower worth of $2.98 - 1.60 = 1.38$ percent ρ .

A comparison can also be made to the previous operating criticality. Before the control rod change was made the reactor was operating at 20 Mw with equilibrium Xe, with the central rod at 33 in. and all other rods full out.

The reactivity based on these conditions is:

$$\begin{array}{rcl}
 \text{reactivity in Xe} & = & (2.1)(.87) = 1.827 \\
 \text{reactivity in voids} & = & (2) (.87) = 1.74 \\
 \text{reactivity in No. 9 rod} & = & .31 \text{ (from Fig. 100C-1)} \\
 \text{Total} & = & \underline{3.877\% \rho}
 \end{array}$$

giving an operating follower worth of

$$3.877 - 1.60 = 2.28 \% \rho$$

This last number is almost equal to the theoretically possible maximum and is therefore somewhat suspected. Basing our estimate on the more consistent room temperature measurements and calculations, we have a loss of approximately 1.6 - 1.8 percent ρ for a rod position of 26 to 32 in., dial.

Hf versus B-Stainless Steel Rod. A relatively "clean" experimental comparison of the two types of control rods was obtained. Experimental conditions are shown in Table I.

Table I - Experimental Conditions

Hf rod	B-Stainless Steel Rod
Critical conditions:	Critical conditions:
No. 9 rod (Hf) all in.	No. 9 rod (B-stainless steel) all in.
Rods No. 1-8 all at 42.5 in.	Rods No. 1-8 at 43.4 in.
Period conditions:	Period conditions:
No. 9 rod all in.	No. 9 rod @ 4 in.
Rods No. 1-8 at 48 in. (all out).	Rods No. 1-8 all out.
Reactor on a 24 sec. period	Reactor on a 35 sec. period.
Poison conc. .865 g H_3BO_3 /gal. in both cases.	Poison conc. .847 H_3BO_3 /gal. in both cases.

Previous to the experiment the rod bank indicated a reactivity of 3.24 percent in the reactor. Since H_3BO_3 is worth 1.23 percent per g per gal., the total worth of the central rods can be obtained as follows:

$$\text{Worth of rod} = 3.24\% - \text{worth of } \text{H}_3\text{BO}_3 - \text{reactivity in period}$$

$$\text{Worth Hf rod} = 3.24\% - 1.064 - .154 = 2.02\%$$

$$\text{Worth B-stainless steel rod} = 3.24 - 1.042 - .123 = 2.075\%.$$

The slight experimental edge of the B-stainless steel rod is almost within the error of the H_3BO_3 concentration determination. The concentrations quoted here are averages of two values obtained by independent methods, one based on a material balance, the other on a chemical analysis. A difference of a factor of 1.03 existed between these measurements. In any case, the rods are shown to be quite comparable with the B-stainless steel rod probably the stronger.

This was indicated also by a different theory cell calculation where the boundary conditions at the rod surfaces were obtained by transport theory approximations. Under otherwise identical conditions a ζ_∞ of .9902 and .9967 was obtained for the B-stainless steel and Hf rod containing cells respectively. This gives a factor of about 1.0065 to the B-stainless steel rod.

Calibration of Control-Rod Bank. The experimental data obtained during the calibration, and the integrated worth of the control rod bank in the upper half of the core are shown in Figure 100C-2. The data was obtained with two different control rods in the center, and as can be seen the points for the B-stainless steel rod with the stainless steel follower in the center tend to be somewhat lower. The flattening of the differential control rod worth in the upper 8 inches of the core is due to the fact that at this point the 36 in. long stainless steel followers have started to move out of the lower regions of the reactor, thus releasing a part of the reactivity they had absorbed.

Fig. 100C-2. Rod worth for 9 rod bank

Summary. A series of rod calibrating experiments using the H_3BO_3 addition method were made. The effect of the stainless steel control rod followers was determined and a comparison between a Hf and a B-stainless steel center control rod were obtained.

Summarized results are as follows:

1. Under most operating conditions (control rods in upper regions of core) the stainless steel followers will absorb approximately 1.6 - 1.8 percent.
2. The integrated worth of the center control rod follower is 0.29 percent. Extrapolating this to the 9 control rod follower bank gives about 2 percent.
3. The integrated worth of a Hf rod in the center is 2.02 percent.
4. The integrated worth of a B-stainless steel rod in the center is 2.07 percent.

SECTION VI DYNAMICS TESTS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
26	Some Kinetics Experiments in EBWR, J. A. Thie	265
41, 41A, 41B	Reactor Transfer Function Measurements, A. Hirsch, W. C. Lipinski	270
43A	Analysis of EBWR Control Rod Ejection Experiment, S. Jacobs, J. A. Thie	273
54B	Operation at 62 Mw, E. A. Wimunc	279
54C	Stability Analysis of 60 Mw Run, J. A. Thie	282
72A	Demonstration of the UTB Voidmeter, J. A. Thie, J. Beidelman	284

SOME KINETICS EXPERIMENTS IN EBWR

(Test Report No. 26, by J. A. Thie)

Low Pressure Oscillations. The first experiment on self-sustained reactor oscillations, Run 46-4, has been reported.¹ Figure 26-1 is a plot

Fig. 26-1. Buildup of EBWR oscillations with power.

the reactor during the experiment were:

All control rods	47 in.
Mean power	20 Mw
Δk in voids	.75 percent
Boric acid content	9 kg
Mean water level	11 ft 7 in.
Pressure (televised)	
a) in constant pressure part of experiment	620 $\pm$ 2 psig (by manual adjustment)
b) in constant steam valve position part of experiment	(mean value varies)

Figures 26-2 to 26-5 show the results of the experiment, relating all quantities by their values of amplitude of oscillation divided by the mean value of the parameter.

¹J. A. Thie, "EBWR Physics Experiments," ANL-5711, July 1957.

from the largest of these oscillations, and demonstrates quantitatively the threshold and amplitude variation. Division of this figure's ordinate by the reactor's zero power transfer function shows the oscillatory reactivity does not exceed 0.06 percent. However, since the Δk in voids is believed to be < 0.4 percent, a substantial fraction of the voids could have been oscillatory. Existing data is too meager to warrant any further experimental inferences from these oscillations.

Feedwater Oscillations. Approximately square wave oscillations of the feedwater flow were obtained by hand manipulation of the valve in Run 46-32. The conditions of

Fig. 26-2. Transport functions during constant pressure feedwater oscillations.

Fig. 26-3. Maximum deviation of the reactor from criticality during constant pressure feedwater oscillations.

Fig. 26-4. Transfer functions during constant steam valve position feedwater oscillations.

Fig. 26-5

Transfer functions during constant steam valve position feedwater oscillations

Low frequency feedwater oscillations caused a long term drifting of the power, presumably due to a slow concentration and dilution effect of the boric acid, as shown in Fig. 26-6. This long term effect was ignored in obtaining power amplitudes in the constant pressure case. The reactivity oscillations are seen from Fig. 26-7 to be well under the Δk in voids.

Fig. 26-6. Power response to feedwater oscillation at constant pressure.

Fig. 26-7. Reactivity corresponding to the power shown in Fig. 26-6.

Reactivity Excursions. The purpose of these experiments, Runs 46-39 to 46-44, was originally to get the threshold of oscillations as a function of reactor conditions. Although the experiment was terminated before this was obtained, information was had on the reactor's response to slow periods. Figure 26-8 shows peak power as a function of reciprocal initial period, when a period was generated by moving all 9 rods from their critical position and the reactor was allowed to rise to its peak power. During these excursions the reactor conditions were as follows:

All control rods -	
When critical at 68°F	11.95 in.
During excursions	12 in. to 13 in.
Boric acid and Xe content	0
Water level	9 ft 11.5 in. to 11 ft 11 in.
Pressure	0 psig

It was found that the peak power attained is insensitive to water level and temperature. Figure 26-9 which shows a typical excursion, also shows that the reactivity lost is at all times very nearly proportional to the power, the proportionality constant being 13 Mw per percent of reactivity

lost, and period independent in this period range. This is in contrast to faster excursions on Borax in which the reactivity lost was proportional to the energy. For these calculations the reactivity was had by AVIDAC analysis of the power. An independent check on the power coefficient of reactivity was Run 46-43 in which rods were pulled at 188°F in an attempt to keep the period constant. This gave 15 ± 7 Mw per percent reactivity.

Fig. 26-8 Reactivity excursions in EBWR.

Fig. 26-9. Reactivity excursion of Run 46-44 at 188°F.

Relevant to the onset of oscillatory power behavior it was found from these excursions that heating of water below 212°F with powers as high as 14 Mw is possible without oscillations, whereas heating of water at 212°F has an oscillation threshold below 1.19 Mw.

REACTOR TRANSFER FUNCTION MEASUREMENTS

(Test Reports Nos. 41, 41A, 41B, by A. Hirsch and W. C. Lipinski)

Experiment No. One. The object of this test was to obtain transfer function data under equilibrium Xe conditions at ~ 40 Mw and 550 psi. Experiment No. 1 was started and the reactor had reached a near equilibrium level of 34 megawatts at 550 psig with the center rod at 41 in. and rods no. 1 through no. 8 at 47 in. During the sinusoidal testing, automatic bypass and feedwater control was used and the turbine was operating. Manual bypass and feedwater control were used for the steady state step test.

Experiment No. Two. For Experiment No. 2, pressure was reduced to approach comparable conditions simulating core voids of 50 megawatts. As the pressure reduced, the flux and power dropped off to a value of 26 megawatts at 300 psig. Automatic bypass and feedwater control was used during the sinusoidal testing, and manual control during the steady state step test.

Figures 41-1 through 41-4 present the data of reactor frequency response for Experiments Nos. 1 and 2.

The reactor noise level was quite high throughout both tests making it difficult to obtain accurate data in the low frequency spectrum.

Fig. 41-1. Experiment no. one, phase vs ω .

Fig. 41-2. Experiment no. one, gain vs ω .

Fig. 41-3. Experiment no. two, phase vs ω .

Fig. 41-4. Experiment no. two, gain vs ω .

Experiments nos. Three and Four. These experiments are comparable to Experiments nos. 1 and 2. The reactor conditions differed only slightly in that the equilibrium power for Experiment no. 3 was 32.6 Mw compared to 34 Mw for Experiment no. 1. Manual bypass and feedwater control was used. The turbine was not operating during Experiment no. 3. During Experiment no. 4 the turbine was operating because the bypass valve could not pass the required steam at 300 psig. Figures 41-5 through 41-8 present the data of reactor frequency response for Experiments nos. 3 and 4.

Fig. 41-5. Experiment no. three, phase vs ω .

Fig. 41-6. Experiment no. three, gain vs ω .

Fig. 41-7. Experiment no. four, phase vs ω .

Fig. 41-8. Experiment no. four, gain vs ω .

Experiment No. 5. This experiment was made at 50 Mw and 600 psig. Figures 41-9 and 41-10 present the data of reactor frequency response measurements performed, with automatic bypass control and automatic feedwater control. The turbine was in service.

Fig. 41-9. Experiment no. five, phase vs ω .

Fig. 41-10. Experiment no. five, gain vs ω .

ANALYSIS OF EBWR CONTROL ROD EJECTION EXPERIMENT

(Test Report No. 43A, by S. Jacobs & J. A. Thie)

Purpose and Conditions. A method of evaluating a reactor's reactivity to power transfer function through the use of a sudden reactivity change has been proposed by Bethe¹ and employed for a calculation of the transfer function of Borax IV.² In this report a trace of the EBWR power following a rapid control rod ejection is analyzed to yield the transfer function, $1/n \, dn/dk = G(\omega)$, and this value compared with the results of rod oscillator measurements. The data is employed for a calculation of the rod worth, which can also be compared with a known value. Damping constants of similar rod ejection experiments involving different initial conditions are indicated in the results.

The experiment consisted of stable operation at some power followed by a sudden control rod ejection, with the result shown in Fig. 43A-1. Table I gives the initial conditions of the experiment.

Fig. 43A-1. Flux and rod position vs time.

¹H. A. Bethe, APDA 117, Appendix.²E. Fabricius and J. Thie, Private communication.

Table I - Experimental Conditions before Ejection

Pressure	600 psig
Power	40 Mw
Steam bypass control	automatic
8 Shims	45.5 in.
Center rod	24.0 in.
Amount of reactivity to be injected	.09 percent

The reactivity worth of the rod has been measured at .045 percent/in.

Theory. The formula for the transfer function in terms of power, generalizing Bethe's development, is:

$$G(\omega) = \frac{1}{N(0) I} \left\{ \frac{1}{2\pi} \int_0^{\infty} [N(t) - N(0)] e^{-i\omega t} dt - \frac{1}{2\pi i\omega} [N(0) - N(\infty)] \right\} \quad (1)$$

where

$$I = \lim_{\gamma \rightarrow 0} \left[\frac{1}{2\pi} \int_0^{\infty} K(t) e^{-\gamma t} e^{-i\omega t} dt \right] \quad (2)$$

Here

$N(t)$ = reactor power

and

$K(t)$ = reactivity insertion starting at $t = 0$.

For a step input at $t = 0$ one obtains

$$G(\omega) = \frac{1}{N(0) K_0} \left\{ N(\infty) - N(0) + i\omega \int_0^{\infty} [N(t) - N(\infty)] e^{-i\omega t} dt \right\} \quad (3)$$

where

K_0 = reactivity step.

For the more realistic case of a ramp reactivity insertion,

$$K(t) = at \text{ for } 0 \leq t \leq t_1 \text{ and } K(t) = at_1 = K_0 \text{ for } t \geq t_1 \quad ,$$

$$G(\omega) = \frac{i\omega}{aN(0)[1 - e^{i\omega t_1}]} N(\infty) - N(0) + i\omega \int_0^\infty [N(t) - N(\infty)] e^{-i\omega t} dt \quad (4)$$

where

t_1 = time of duration of ramp .

Method. The integral $\int_0^\infty [N(t) - N(\infty)] e^{-i\omega t} dt$ was computed

numerically, taking $t = 2.80$ as the upper limit. $N(\infty)$ was evaluated by integration of $N(t)$ from $t = 2.80$ to $t = 6.40$. As a check of the numerical integration, $N(t)$ was fitted with a damped sine wave and the integral evaluated analytically. This analytic approximation of the power trace, $N(t) = A \sin \omega' t e^{-\alpha t} N(\infty)$, has the constants indicated as follows.

A	ω'	α	$N(\infty)$
52.5	8.98	1.48	291.6

The transfer function was calculated using both the analytic approximation and the exact numerical values.

Transfer Function Results. The results of the analysis assuming a step input at the start of the test are given in Table II.

Table II - Results Assuming Step Input

$\omega(\text{rad/sec})$	$G(\omega)$	
	Numerical	Analytical Approximation
5.9		187.0 $\angle -58.5^\circ$
6.8		265.2 $\angle -55^\circ$
7.0		281.0 $\angle -51.7^\circ$
8.98	624.8 $\angle -3^\circ$	530.0 $\angle -4.4^\circ$
10.46		408.1 $\angle -36.9^\circ$
12.0		272.1 $\angle -52.8^\circ$
12.1		265.2 $\angle -55^\circ$

The chamber output shown in Fig. 43A-2 was corrected as shown because of presumably malfunctioning equipment.

Fig. 43A-2
Transfer function ramp input.

These approximate results were modified by taking into account the ramp nature of the reactivity input. These results are shown in Table III.

Table III - Results Assuming Ramp Input

ω (rad/sec)	$G(\omega)$	
	Numerical	Analytic Approximation
5.9		193.9 $\angle -31.4^\circ$
6.8		279.6 $\angle -23.7^\circ$
7.0		304.8 $\angle -19.6^\circ$
8.98	683.3 $\angle 38.2^\circ$	579.7 $\angle 45.6^\circ$
10.46		460.1 $\angle 11^\circ$
12.0		318.6 $\angle 2.2^\circ$
12.1		312.7 $\angle .4^\circ$

These results are shown in Fig. 43A-3. Figures 43A-4 and 43A-5 show these results and those of rod oscillator measurements.

Fig. 43A-3. Rod oscillator measurements.

Fig. 43A-4. Phase of transfer function.

Fig. 43A-5

Damping constant vs power.

Damping Constant Results. Many "ringing" experiments were done, but only one was analyzed for a transfer function. The others can be compared as to relative stability of their damping constants.

The damping constant, α , of the expression $Ae^{-\alpha t} \sin \omega t$ is computed from the formula

$$\alpha = \frac{2}{T} \ln \left(\frac{P_1 - P_2}{P_3 - P_2} \right),$$

where T is the period of oscillation, and P_1 , P_2 , P_3 are respectively the first maximum, first minimum, and second maximum of the above expression. The results for the different initial reactor conditions are shown

Fig. 43A-6. Initial reactor conditions.

in Fig. 43A-6. The interpretation of such a graph is that as the damping constant approaches zero the reactor is becoming less stable.

Control Rod Calibration.

A rod calibration was obtained by taking the ratio of the slopes of the K_{ex} vs time curve and the rod position vs time curve. The rod worth computed is

$$\frac{\Delta K_{ex}}{\Delta \text{in.}} = .060 \text{ percent/in.},$$

as obtained from the quotient of the slopes, which is somewhat higher than the .045 percent/in. obtained by calibration during a rod oscillator experiment.

Conclusion. In view of the fact that data from a malfunctioning chamber was used, plus the possibility of the transfer function being amplitude dependent at high powers, it is believed that ringing and rod oscillator transfer function are in accord. Also to obtain accuracy from the ringing method it is necessary to average the results of many tests when the pile noise is high. The ringing method of obtaining the transfer function is therefore a useful one because of its experimental simplicity, and the rapidity with which an approximate answer can be had using the analytical integration approximation.

EBWR OPERATION AT 62 Mw

(Test Report No. 54B, by E. A. Wimunc)

The EBWR reactor was operated at 61.7 Mw with momentary powers possibly reaching 65 Mw. In order to operate the plant to a power level between 60 Mw and 65 Mw, two natural thin elements in the center of the core were replaced with two thin enriched (1.44 percent enrichment) elements. The balance of the core remained as originally loaded.

In the first attempt to reach a high power level the plant was shut-down by a "Radiation Level High" shutdown circuit just as 50 Mw was attained. The radiation chamber located in the air exhaust duct monitors all air and gases that are discharged from the plant. The high radiation level was attributed to a shorter holdup time in the newly installed bypass located at the hydrogen recombiner. The scale range of this monitor was raised from 100 mr/hr full scale to 1 r/hr full scale so that the test could be resumed. The range of the monitor was lowered after the test was completed.

The plant was started again and three hours and ten minutes after initial criticality was achieved it was operating at a power of 61.7 Mw thermal heat with the turbine loaded to generate 5.5 Mw electrical power, and the balance of the steam dumped into the condenser through the bypass valve. The system was not at equilibrium Xe conditions during the power run.

It is believed that steady power greater than 61.7 Mw could have been attained but the reactor feedwater pumps were delivering their maximum under these operating conditions. The pump discharge pressure was 710 psig; about 34 psi was due to the pressure drop across the full flow condensate filters and the balance of about 75 psi was due to the pressure drop in the valves and piping.

The water level in the sight glass was maintained at 11 ft - 6 in. instead of the usual 11 ft - 11 in. so that water carryover might be minimized.

Fast recorders were used to monitor the flux and pressure so that any irregularities would be quickly detected. From all observations nothing unusual was noted in reactor behavior.

Table I contains a list of the more pertinent data taken about 15 minutes after the 61.7 Mw equilibrium operation was obtained.

Table I

Generator output	5.5 Mw
Reactor steam flow	192,000 lb/hr
Reactor FW pressure	710 psig
Pump no. 1	151 amps
Pump no. 2	142 amps
Reactor pressure	600 psig
Reactor temperature	489°F
Condenser pressure	
Indicator	25.5 in. Hg (vac.)
Recorder	3.5 in. Hg abs.
Reactor water level	11 ft - 6 in. (TV)
Micro-microammeter	$4 \times 1 \times 10^{-4}$
Rod positions no. 1 - no. 8	47 in.
Center	45 in.
IX no. 2 flow	9.5 gpm
Conductivity	
IX no. 2 inlet	.82 μ mho
IX no. 2 outlet	.14 in.
North half condenser hotwell	.14
South half condenser hotwell	.14
Radiation levels	
Turbine	20 mr/hr
Steam drier	720 mr/hr
Air ejector output	7000 mr/hr
Condenser hotwell	580 mr/hr
Feedwater filters	100 mr/hr
Plant air exhaust	120 mr/hr
Temperatures	
Secondary vent condenser	50°F
Makeup air cooler	40°F
Plant air	90°F
Lower bearing FW pump no. 1	92°F
Lower bearing FW pump no. 2	90°F
Air ejector discharge	141°F
Desuperheater temperature	318°F
Reference temperature	89°F
Hotwell	141°F
Feedwater	153°F
Condenser water inlet	72°F
Condenser water outlet	102°F
Water from oil cooler	75°F
Water from air cooler	82°F

Table I (cont'd.)

Top bearing FW pump no. 1	101°F
Top bearing FW pump no. 2	118°F
Water from regen. cooler no. 2	174°F
Water to IX no. 2	100°F
Water to reactor from purifier	410°F
Water from shield cooling coils	117°F
Water to shield cooling coils	115°F

Items not listed were normal or did not have a bearing on this run.

For the record:

Outdoor temperature	29°F
Relative humidity	100 percent

Both air ejectors, cooling tower pumps and fans were in operation. A few days prior to this run, the water side of the main condenser had been chemically cleaned and therefore performed much better than in the previous higher power operation.

STABILITY ANALYSIS OF 60 Mw RUN

(Test Report No. 54C, by J. A. Thie)

Oscillatory Threshold. The threshold of spontaneous reactor oscillations may be defined as that power at which two or more successive cycles of the same frequency appear without noticeable phase shifting by boiling noise. Figure 54C-1 shows this threshold to be 13 Mw, and also shows the growth of the oscillatory amplitude with power. Figure 54C-2 shows the correlation of the latter with the maximum value found for the transfer function for the same power. This figure makes possible the prediction of spontaneous oscillations from transfer function predictions.

Fig. 54C-1. Spontaneous EBWR oscillations as a function of power.

Fig. 54C-2. Correlation of spontaneous oscillations with transfer function.

Resonant Frequency. Figure 54C-3 shows the well-known trend of the resonant frequency with power. The similarity in trends of data from this run and that previously found in transfer function measurements is also worth noting.

Oscillation Envelope. The oscillatory trains shown in Fig. 54C-3 are typical of those found at 60 Mw. If a function of the form

$$f(t) = \exp. (-\zeta \omega t)(\cos \omega t)$$

is fitted to the onset of typical oscillatory trains, it is found that ζ is generally no worse than -.04.

Fig. 54C-3
Effect of power on resonant frequency.

Reactivity of Voids. Due to a departure from the standard programming of rods and omission of criticals it was not possible to evaluate the reactivity in voids at 60 Mw.

Pressure Transfer Function. During the peaks of the oscillatory trains at powers above 50 Mw it was possible to observe steam dome pressure fluctuations. The transfer function between pressure, p , and fractional power, n/N^0 , is

$$\frac{p}{n/N^0} = 1.315 \text{ psi} / 180^\circ$$

DEMONSTRATION OF THE UTB VOIDMETER

(Test Report No. 72A, by J. A. Thie and J. Beidelman)

Theory. In multigroup diffusion theory the equation relating the thermal flux, ϕ_3 and the epithermal flux, ϕ_2 , is

$$-\nabla D_3 \nabla \phi_3 + \Sigma_{a3} \phi_3 = \Sigma_{R2} \phi_2 \quad (1)$$

In typical light water power reactors the first term is small and the thermal to epithermal flux ratio is the removal to absorption cross section ratio corrected for thermal leakage:

$$\frac{\phi_3}{\phi_2} = \frac{\Sigma_{R2}}{\Sigma_{a3} (1 + L^2 B^2)} = K(z, \alpha) (1 - \alpha) \quad (2)$$

Equation (2) introduces the almost constant $K(z, \alpha)$, by taking advantage of Σ_{R2} being very nearly proportional to 1 minus the channels' void fraction. Other effects contributing a slight z and α dependence to K came from α , Σ_{a3} , L^2 , and B^2 .

The cadmium ratio of a material having a thermal cross section, σ_{a3} (.0253), and an epithermal cross section,

$$\sigma_{a2} = \frac{\int_{E_c}^{E_0} \sigma_a \frac{dE}{E}}{\int_{E_c}^{E_0} \frac{dE}{E}} \quad (3)$$

is given by

$$CdR = 1 + \frac{\phi_3}{\phi_2} \frac{\sigma_{a3}(.0253)}{\sigma_{a2}} A(z, \alpha) \quad , \quad (4)$$

where $A(z, \alpha)$ is a factor which takes the shape of the local thermal spectrum into account. Substitution of (4) into (2) makes it possible to determine α from the measured cadmium ratio:

$$1 - \alpha = \frac{(CdR - 1) \sigma_{a2}}{K(z, \alpha) \sigma_{a3}(.0253) A(z, \alpha)} \quad (5)$$

$$1 - \alpha = \frac{(CdR - 1)}{(CdR - 1)_0} \frac{K(z_0, \alpha) A(z_0, \alpha)}{K(z, \alpha) A(x, \alpha)} \quad (6)$$

The subscript zero refers to a point, z_0 , in the subcooled region away from the boiling boundary and also the bottom of the fuel, $z = 0$.

Figure 72A-1 was obtained from rigorous 3 group, multiregion calculations of EBWR at 42.7 Mw, 8000 Mwd, and all control rods out,

Fig. 72A-1. Calculated relationship between the cadmium ratio and the local void fraction.

essentially the conditions of the experiment in which bare and cadmium covered cobalt wires were irradiated. The fluxes from rz PDQ problems were used to evaluate the cadmium ratio according to (4) at various heights and voids. Since the voids are not experimentally known, two cases having different values of

$$\bar{\alpha} = \frac{1}{H} \int_0^H \alpha dz$$

for an element at $r = 9$ in. were assumed. It is seen that (6) can be approximated by a straight line of unit slope if one wishes to bypass the complexities of these calculations.

Experiment. Bare cobalt wires were irradiated in four elements in Run 52-2 at an average power of 42.7 Mw for 5.88 hours. In addition, cadmium ratios were obtained in three elements, two of which were at $r = 9$ in. Figures 72A-2 and 72A-3 show the results of extensive high precision counting of these wires. It was necessary to be able to detect small variations in the cadmium ratio, and Fig. 72A-3 shows the desired precision has been achieved. A priori calculations of cadmium ratios made above are shown in Fig. 72A-3 to be in agreement with experiment.

The wires were counted in one inch increments by an automatic counting system. A sodium iodide scintillation detector was used, and a single channel pulse height analyzer was set for the counting of only cobalt gamma rays. This was necessary since there were also other elements present in the wires. To obtain the desired statistical accuracy, each of the increments was counted for no less than 10,000 counts, and to reduce non-statistical counting rate variations, an average was taken of at least three such counts of each increment. In cases where time and the activity of the wire permitted, a greater number of counts was accumulated and an average of more repetitions was taken. Timing errors were in all cases less than one percent.

Fig. 72A-2. Flux distributions in EBWR at 42.7 Mw and all control rods out.

Fig. 72A-3. Cadmium ratios in EBWR at 42.7 Mw with all control rods out.

Figure 72A-2, showing the shift in the flux peak toward the bottom of the reactor, is believed to be the first experimental evidence that voids distort the flux. Previous attempts at investigating this have shown principally the effects of control rods on the flux. The slight bump in these flux plots just below their maximum is not due to counting statistics. Though not understood, it is probably significant that this distortion occurs near the boiling boundary.

The reactivity in voids was measured as part of this experiment from the position of the control rods at zero power, and the rod calibration curve at zero power, 600 psig conditions. A result of 3.38 percent reactivity at 42.7 Mw was obtained.

Incidental to the purpose of the experiment a calibration of the L and N recorder(hitherto uncalibrated), commonly used to indicate re-

actor power, was obtained. Figure 72A-4 combines data of Run 52-2 with earlier experiments at other pressures and control rod positions.

Fig. 72A-4. Calibration of the L and N recorder.

and 33.5 percent respectively. There is reason to believe these values will be in agreement with a priori hydraulics calculations. The curve computed with $\bar{\alpha} = .2125$ was used, as it was found to be more consistent with the α 's determined. It should be pointed out that no "vicious circle" exists, for the assumption of a particular void distribution in computing Fig. 72A-1 is only for purposes of refining the method, as pointed out above.

Results. Since the boiling boundary was estimated to be at $z = 12$ in., z_0 was chosen as 8.6 in. in order to be sufficiently distant from the bottom reflector and the boiling boundary. Some nucleate bubbles may have been present at this point, but this effect was ignored in normalizing the cadmium ratio to zero voids. The experimental cadmium ratios of Fig. 72A-3 were then used in combination with Fig. 72A-1 to determine α at three locations, $z = 15.4, 23.7$ and 33.4 in. This gave $\alpha = 8.3$ percent, 19.5 percent,

The measured flux distribution has been converted to axial and radial power distributions in Figs. 72A-5 and 72A-6 by using the

Fig. 72A-5. Measured axial power distribution at 42.7 Mw.

and, as expected, the calibration number varies almost a factor of 10 in the range of reactor conditions encountered.

theoretical isotope distributions at 8000 Mwd in this element. The a priori calculation of a radial distribution in Fig. 72A-6 is not for purposes of comparison with experiment since its assumed voids were lower than the above measurement indicated. The product of the experimental maximum to average factors of these figures is 2.2. There is reason to expect that this will be in agreement with a priori calculations using the proper void distribution.

Summary. Using a method proposed by Untermeyer and perfected by the authors, the voids in a thin enriched assembly located at $r = 9$ in. have been measured at 42.7 Mw at heights of 15.4, 23.7 and 33.4 in. above the bottom of the fuel. The voids were measured as 8.3 percent, 19.5 percent, and 33.5 percent respectively. In addition, axial and radial distributions of thermal and epithermal flux were measured. Finally a calibration of the L and N recorder was obtained,

Fig. 72A-6
Radial power distribution
at 42.7 Mw.

SECTION VII FLUID CHARACTERISTICS

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
39	Preliminary Determination of Recirculation Rates in EBWR, S. A. Bernsen	293
39A	Recirculation Flow Rate Tests, W. A. Sutherland	295
39B	EBWR Recirculation Rate, A. Hirsch	297
39E	Further Recirculation Flow Rate Tests, W. A. Sutherland	298
39G	Measurement of Inlet Subcooling for the Determination of Recirculation Flow Rates, B. M. Hoglund	303
62, 62A, 62B	Water Level Measurements in the Reactor Vessel, R. L. Mittl, M. Theys	305

PRELIMINARY DETERMINATION OF RECIRCULATION RATES IN EBWR

(Test Report No. 39, by S. A. Bernsen)

A series of measurements were made on EBWR to determine the inlet subcooling at various pressures and power levels. It should be pointed out that the results are preliminary and that the present data were obtained from only one inlet temperature thermocouple. If the results of this test check theoretical data it would be difficult to say whether the result was a fortunate accident or a true check of existing theory.

The observed feedwater flow was corrected for the 2000 lb/hr flow from the ion exchange loop. During all the tests a rather large variation in inlet subcooling was observed. These variations amounted to as much as 30-40 percent of the average indicated inlet subcooling. From the equation for recirculation rate:

$$W_c = W_F \frac{(T_R - T_{FW}) \bar{C}_F}{\Delta T_1 \bar{C}_c} = \frac{\text{const}}{\Delta T_1}$$

we can see that large errors in ΔT_1 produce large and asymmetrical errors in W_c . For this reason the average temperature does not necessarily obtain the average flow rate. Nevertheless, the data are presented as average inlet subcooling and average circulation obtained therefrom. For the 600 psi data an attempt was made to indicate the maximum observed inlet subcooling. Table I summarizes the data which are plotted in Fig. 39-1.

TABLE I - Experimental and Plotted Data

Pressure, psia	Power, Mw	W_F ($\times 10^3$)	T_F (°F)	T_R (°F)	ΔT_1 (°F)	ΔT_2 (°F)	W_{c1} ($\times 10^6$)	W_{c2} ($\times 10^6$)
600	20	60	119	488	4.56	4.53	4.53	4.56
600	15	45	119	486	3.90	3.37	4.00	4.62
600	10	30	121	486	2.73	2.54	3.87	4.16
600	5.3	16	124	485	1.90	1.70	3.09	3.46
600	2	6	152	485	1.06	0.87	2.27	2.77
300	19	57	120	417	2.23	2.20	6.97	7.07
300	14.7	44	117	414	2.23	2.03	5.44	3.98
300	9.7	29	110	411	1.56	2.06	5.30	4.02
300	4.7	14	112	416	1.06	0.53?	4.06	8.13
300	2.7	8	139	416	.73	0.3?	3.37	8.20

Fig. 39-1. EBWR recirculation rate preliminary data. Test of Nov. 6-8, 1957.

Procedure. The following procedure was used for the purpose of estimating the recirculation ratio of EBWR. The measurements were made over a range of power levels and pressures. The determination was made by measuring the indicated average temperature difference between a thermocouple located in the steam dome and one located below the core support plate. A check can be obtained by measuring the temperature difference between a thermocouple located in the water above the core and the same thermocouple below the support plate. These differentials are denoted by ΔT_1 and ΔT_2 respectively. Measurement of feedwater flow rate and temperature, and reactor power complete the determination.

The recirculation flow can be defined as

$$W_c = W_F \frac{(T_R - T_{FW}) \bar{C}_F}{\Delta T_1 \bar{C}_c}$$

where:

- W_c = recirculation flow rate
- W_F = feedwater flow rate
- T_R = reactor temperature
- T_{FW} = feedwater temperature
- $\bar{C}_F$ = average specific heat over range $(T_F - T_R)$
- $\bar{C}_c$ = average specific heat over range ΔT_1

The first experiment was to determine the zero values of the sub-cooling. For this purpose the reactor power was reduced to about 1-2 Mw and, after reaching equilibrium, the feedwater flow was zero. Observations of ΔT and ΔT_2 were made and reactor power measured. The test was short enough so that reactor water level did not fall below the low level alarm.

The second experiment consists of determining ΔT_1 and ΔT_2 at equilibrium reactor conditions at the following points:

<u>Reactor power, Mw</u>	<u>Reactor pressure, psi</u>
2.5, 5, 10, 15, 20	300, 600

These data should be verified at a later date and should be extended to include higher power levels when obtained.

RECIRCULATION FLOW RATE TESTS

(Test Report 39A, by W. A. Sutherland)

An investigation has been conducted to determine recirculation flow rate in EBWR. The method of analysis was the same as had been previously used, with what is believed to be an improvement in the experimental technique. The results are presented here as recirculation flow rate vs reactor power. (Fig. 39A-1).

The twelve thermocouples in the reactor vessel were located as follows

2 below fuel element	27-255
2 below fuel element	27-43
2 below dummy element	0-173
1 below dummy element	0-278
1 below dummy element	0-67
1 below dummy element	0-384
1 center dummy element	0-384
1 top dummy element	0-384
1 steam zone - located in core index position	

Eight of the nine bottom thermocouples were wired in parallel to average out the inlet temperature measurement. This proved to be a satisfactory arrangement. The center thermocouple was of no use, and the top thermocouple was of value only as a checking device. The steam zone thermocouple, as calibrated against reactor saturation pressure, indicated 105 microvolts below standard value and caused some concern on the measurement of subcooling since this was in the order of the signal strength.

Investigation of transient or periodic variations of temperatures was not possible because only one amplifier which could handle the small signals involved was available. Data were taken using one thermocouple, or subcooling directly, against time.

It is believed that more precise results can be obtained by using a thermopile arrangement to measure subcooling. It is recommended that this method be included at such time as an instrumented fuel element is placed in the core

Fig. 39A-1.
Recirculation flow rate vs power

EBWR RECIRCULATION RATE

(Test Report No. 39B, by A. Hirsch)

A disturbance of the feedwater flow is reflected in the power and flux of the reactor. It was thought that if an impulse were impressed upon the feedwater flow, the flux would show when the impulse entered the core and again when the recirculation brought the impulse around to re-enter the core. From the time of recirculation and the area of the downcomer, it was hoped to get the recirculation rate. The procedure was as follows:

1. Turbine and generator shutdown
2. High flux trip set to 30 Mw.
3. Second feedwater pump started.
4. Open feedwater bypass valve and reclose immediately.
(Feedwater on manual)
5. Open feedwater bypass and close main valve; interchange so that flow is normal for power of reactor.
6. Close bypass valve and reopen immediately.

Step four (4) will give positive impulses and six (6) will give negative impulses.

Data was recorded on the Sanborn Recorder as follows:

1. Sub-cooling.
2. Flux of chamber in instrument hold.
3. Flux of chamber in side tube.
4. Differential pressure.
5. Feedwater flow.
6. Steam bypass valve position.

The recorded data showed the impulse as it entered the core. A second impulse was not evident. In retrospect, it can be seen that the power change would tend to wash out any temperature difference in the water leaving the top of the reactor and hence the impulse would not be evident as the recirculation returned it to the core.

However, there was a delay from the time the flow started to change to the time when the flux started to change. The delay times were averaged for all the impulses at each power setting. The data follows:

Power	5 Mw	10 Mw	15 Mw	20 Mw
Time delay	6.50 sec	5.55 sec	4.78 sec	5.09 sec

Using an average distance of eleven (11) feet from feedwater ring to core bottom, one can calculate a flow rate as follows:

Power	5 Mw	10 Mw	15 Mw	20 Mw
Flow	1.69 ft/sec	1.98 ft/sec	2.3 ft/sec	2.13 ft/sec

No explanation can be offered as to why the 15 Mw flow rate appears to be faster than the 20 Mw flow rate.

The suggestion is offered that injection of an amount of poison such as boric acid through one of the forced recirculation inlets might give the impulse recirculation time.

FURTHER RECIRCULATION FLOW RATE TESTS ON EBWR

(Test Report No. 39E by W. A. Sutherland)

The operation of the EBWR at powers up to 60 Mw has provided an extension of the data presented in Test Report 39A. Duplication of data to within plus or minus 5 percent gives confidence in the results, which are believed to be conservative. Extrapolation to very high powers is questionable.

Derivation of Equations.

$$x = \frac{W_g}{W_t} \quad \text{core exit quality}$$

$$y = \frac{W_{g1}}{W_0} \quad \text{reactor steam quality}$$

$$z = \frac{W_{g2}}{W_t} \quad \text{steam entrainment ratio}$$

To determine reactor power, Q_t , an energy balance is made of the system shown in Fig. 39E-1, within boundary no. 1.

$$W_{g1} h_g + (1 - y)W_0 h_f + W_I h_I = W_I h_0 + W_0 h_0 + Q_t$$

$$Q_t = W_0 (y h_{fg} + h_f - h_0) + W_I (h_I - h_0) \quad (1)$$

Fig. 39E-1.
Circulation schematic

It is assumed that all the energy produced, Q_t , is absorbed by the water in the core, and none is transferred to water in the downcomer. Heat losses from the reactor itself are negligible, and it is further assumed that $h_I = h_f$. Then making an energy balance on boundary no 2:

$$W_t h_{in} + Q_t = W_t h_f + W_g h_{fg}$$

$$Q_t = W_t (h_f - h_{in} + x h_{fg}) \quad (2)$$

Setting equation (1) equal to equation (2):

$$W_0 (y h_{fg} + h_f - h_0) + W_I (h_f - h_0) = W_t (h_f - h_{in} + x h_{fg})$$

$$W_0 (h_f - h_0) + W_I (h_f - h_0) = W_t (h_f - h_{in}) + (W_g - W_{g1}) h_{fg}$$

$$\frac{W_0}{W_t} = \frac{h_f - h_{in} + z h_{fg}}{(h_f - h_0) \left(1 + \frac{W_I}{W_0}\right)} \quad (3)$$

since

$$W_g = x W_t = y W_0 = z W_t$$

$$\frac{W_0}{W_t} = \frac{x - z}{y} \quad (4)$$

Equation (3) may be evaluated if z is assumed zero. How valid this approximation is, is unknown: W_{g2} as high as 3,000 lb/hr at 20 Mw would induce an error in W_0/W_t of less than 10 percent.

It has been assumed that reactor steam quality was unity (e.g. $y = 1$). Hence equation (4) reduces to:

$$\frac{W_0}{W_t} = x \quad (5)$$

EBWR Test Report No. 37 indicates that reactor steam quality is better than 0.99 up to 30 Mw, and drops to 0.77 at 40 Mw. It has been established that this 0.77 may be increased to 0.99 by operating with a lower reactor water level. In any event, this large change in y (in effect a step change) between 30 and 40 Mw would cause a similar change in W_0/W_t , and Figs. 39E-2, -3, and -4 should show this effect.

Fig. 39E-2. Steam quality curve No. 1.

Fig. 39E-3. Steam quality curve No. 2.

Since the curves have no pronounced changes in the 30-40 Mw range, and are indeed the shape one would expect, the assumption of $y = 1$ appears valid.

Thermocouple Calibration. Each of the twelve thermocouples was individually calibrated by the manufacturer. However, this calibration was of little value since the thermocouples were used in various combinations, and not in the same manner in which they were calibrated.

Fig. 39E-4. Steam quality curve No. 3.

It might be argued that they could be checked by shutting off the feedwater so that $t_{in} = t_f$. Then if the bottom thermocouples were bucked against the steam zone thermocouple, the error mentioned above would be directly measurable. This was done, indicating an error of from 25 to 40 microvolts compared to an error of 61 microvolts as determined by saturation pressure.

In the author's opinion, the difference between these two methods is constant, regardless of the shifting calibration. In using the saturation pressure method, the corrected subcooling measurement will be about 1°F higher than the no feedwater method. This is in the order of about 10 percent higher for quality and correspondingly, 10 percent lower for recirculation flow rate. Since some doubt exists as to the exact calibration of the thermocouple system, the data presented here are conservative, possibly by as much as 10 percent.

Empirical Results. Figure 39E-2 shows graphically the data presented in Table I. It appears that x is a linear function of power, and is relatively insensitive to feedwater temperature. Over the range shown:

$$x = 0.00828 + 0.000652 P \quad (6)$$

where P is in Mw

from equation (1), assuming $y = 1$

$$W_0 = \frac{Px \cdot 3.413 \times 10^6}{h_g - h_0} - W_I \frac{h_f - h_0}{h_g - h_0}$$

The steam zone thermocouple was adjusted in the reactor by direct comparison with saturation pressure, and hence saturation temperature. This thermocouple was found to read low by 105 microvolts. The value of this error varied from day to day, from 45 to 133 microvolts. The reproducibility of the data as shown in Fig. 39E-3, 5 percent for the most part, indicates that this shifting calibration is indeed the case.

The nine thermocouples at the bottom of the core were used in parallel to average out fluctuations in the inlet temperature. These thermocouples could not be calibrated in the reactor, and were therefore assumed to be collectively reading correctly.

TABLE I

Date	Run	P psia	Power, Mw	Δt , °F	x	W_0 $\frac{\text{lb}}{\text{hr}} \times 10^{-3}$	W_t $\frac{\text{lb}}{\text{hr}} \times 10^{-6}$
2-4-58 ○	1	616.5	21.0	7.9	.0228	63.3	2.79
	2	616.0	15.6	6.8	.0193	47.0	2.44
	3	616.0	11.0	5.7	.0161	33.0	2.05
	4	616.0	5.8	4.2	.0111	17.0	1.53
	5	617.0	11.0	5.7	.0154	35.0	2.27
	6	616.0	16.5	6.7	.0185	49.0	2.65
	7	616.0	21.2	7.9	.0222	63.5	2.86
2-6-58 □	8	615.0	21.7	7.6	.0219	65.5	2.99
	9	615.5	20.9	7.8	.0225	63.0	2.80
	10	615.5	20.7	7.7	.0223	62.5	2.80
	11	620.0	14.9	6.6	.0189	45.0	2.38
	12	620.0	14.9	6.6	.0188	45.0	2.39
	13	615.0	10.0	5.4	.0153	30.0	1.96
	14	614.5	10.0	5.4	.0152	30.0	1.97
	15	613.0	5.3	4.2	.0120	16.0	1.33
	16	613.0	5.3	4.2	.0118	16.0	1.36
	17	618.5	10.0	5.4	.0152	30.0	1.97
	18	618.0	10.0	5.5	.0155	30.0	1.94
	19	624.0	14.9	6.5	.0186	45.0	2.42
	20	616.5	19.8	7.5	.0216	60.0	2.78
2-10-58 ◇	21	615.5	21.8	7.0	.0202	66.0	3.27
	22	615.0	22.5	7.0	.0202	68.0	3.37
	23	615.0	22.4	6.8	.0196	67.5	3.44
	24	616.0	21.9	6.8	.0198	66.5	3.36
	25	616.0	22.1	7.0	.0204	67.0	3.28
2-10-58 △	26	618.5	26.0	8.2	.0245	79.6	3.25
	27	618.5	26.0	8.5	.0252	79.6	3.16
	28	618.0	26.0	8.3	.0248	79.6	3.21
	29	618.0	26.0	8.3	.0248	79.6	3.21
	30	616.5	32.0	9.7	.0293	98.5	3.36
	31	617.0	35.8	10.6	.0328	11.0	3.38
	32	616.5	38.4	11.0	.0346	119.5	3.45
	33	616.0	42.8	11.5	.0373	135.0	3.62
	34	616.0	42.8	11.5	.0374	135.0	3.61
	35	616.5	42.8	11.5	.0375	135.0	3.60
	36	616.5	44.9	12.0	.0392	142.0	3.62
	37	616.0	44.9	12.0	.0393	142.0	3.61
	38	616.0	44.9	11.9	.0390	142.0	3.64
2-11-58 ⬡	39	610.5	41.8	10.5	.0331	131.0	3.96
	40	610.5	41.8	10.4	.0328	131.0	3.99
	41	610.5	41.8	10.3	.0327	131.0	4.01
3-20-58 0	42	616.0	21.9	7.0	.0191	65.0	3.40
	43	616.0	30.9	9.7	.0273	93.0	3.41
	44	616.0	46.5	12.9	.0380	141.5	3.72
	45	616.0	56.0	14.0	.0434	173.0	3.99
	56	616.0	61.4	13.5	.0433	192.0	4.43

1f

$t_0 = 115^\circ\text{F}$ and pressure is 600 psig

$$W_0 = 3.05 \times 10^3 P - 0.350 W_I$$

$$W_t = \frac{W_0}{x} = \frac{3.05 \times 10^3 P - 0.350 W_I}{0.00828 + 0.00652 P} \quad (7)$$

The flow rate curves in Figs. 39E-2 and -4 are plots of equation (7) where $W_I = 0$.

MEASUREMENT OF INLET SUBCOOLING FOR THE DETERMINATION OF RECIRCULATION FLOW RATES

(Test Report No. 39G, by B. M. Hoglund)

The purpose of the test was to check the results of previous tests and, if possible, to evaluate the effect on the recirculation flow rate of the shroud that had been placed around the core.

The temperature measurement detectors consisted of nine, 40 gauge, asbestos covered, Fiberglas sheathed, iron-constantan thermocouples housed in three Inconel sheaths (three thermocouples per sheath). The individual thermocouples had been calibrated to within 0.1°F . Two of the probes were installed in the nose piece of a stainless steel dummy element and measured the core inlet water temperature. The third probe was fastened to the other two sheaths at a point ten feet above the top of the dummy, thus putting it in the steam zone. The thermocouples were connected to copper lead wire in an insulated junction box external to the reactor vessel. The copper lead wires were connected to a switch so that each thermocouple could be read against an ice junction or three of the thermocouples in the water could be bucked against the three steam thermocouples to form a thermopile that would read the subcooling directly.

It was impossible to obtain boron free Fiberglas insulation for the thermocouples. To check the effect of possible heating of the thermocouple bead due to the boron, three of the thermocouples had the insulation stripped back approximately one inch and the bare wires were insulated with Sauerisen cement. The other three water thermocouples had glass insulation almost up to the bead.

The results of the subcooling measurements were absolutely negative due to erratic behavior of the water thermocouples. It is believed that a breakdown of the dielectric strength of the thermocouple insulation due to irradiation was responsible for the inconsistencies in the data. The breakdown was particularly noticeable with this system because of the high

impedance of the 40 gauge wire. Mechlin¹ reported that the leakage resistance of 28-gauge thermocouples with similar insulation, irradiated in MTR, dropped from 2 megohms to 7000 ohms upon application of full reactor power but gradually rose back to approximately 2 megohms within one month. A similar trend was noted in our test data, i.e., the readings were bad (subcooling too high) immediately upon application of power but after the first week the measured subcooling began decreasing and appeared to be approaching reasonable values during the third week of running. The reactor was shut down after three weeks of running.

The thermocouples in the steam zone appeared to be unaffected as they read within one degree of the saturation temperature corresponding to the reactor pressure as indicated by the Norwood Precision Indicator, Model 850. These thermocouples failed just prior to reactor shutdown. This failure was probably due to corrosion of the iron wire by moisture in the sheath.

It appears that the heating effect of the boron in the glass insulation may have affected the readings of the thermocouples that did not have the insulation stripped back. They read from 5 to 12°F above the other water thermocouples. These same thermocouples failed after about a week and a half so that part or all of the discrepancy might be attributed to corrosion.

Another phenomenon observed during the test was an unexplained dc potential of about 100 millivolts between the thermocouple wires and ground. This potential varied slowly with time and was not a function of reactor power. While it had no effect on the potentiometer readings it rendered self-balancing indicators and recorders useless.

Conclusions.

1. If at all possible, heavier gauge thermocouple wire should be used to lower the system impedance and to reduce breakage. Breakages during insulation in the reactor caused a considerable number of problems.
2. An insulation other than asbestos and glass should be considered if accurate readings are desired during the first month of operation.
3. Efforts should be made to completely dry and seal sheaths containing iron-constantan wires because of the rapidity with which the iron wires are attacked by moisture.

¹Mechlin, G. F., "WAPD-1 Experiments in the Materials Testing Reactor; II. Bellows Extensometer Test-WAPD-1-2," WAPD-76, March 12, 1953.

- 4 Care must be exercised with glass covered thermocouples to prevent erroneous readings because of the heat generated by the boron in the glass.
- 5 A program should be initiated to discover the source of the dc potential obtained between these thermocouples and ground.

WATER LEVEL MEASUREMENTS IN THE REACTOR VESSEL

(Test Reports No. 62, 62A, 62B, by R. L. Mittl and M. Theys)

Purpose The purpose of the tests was to determine the true water level in the reactor vessel and the thickness of the interface between water and steam. The methods presently employed to indicate the water level are a sight glass and a differential pressure recorder. The sight glass does not give a true indication of water level due to the variations in the temperature and consequently the density of the water contained within the reactor and piping connecting the sight glass to the vessel. The presence of voids in the reactor water during operation increases this error. The water level indicated by the sight glass is always lower than the true level within the vessel during operation. The differential pressure recorder is only a means of determining the relative variations of the water level. It is not a method for an absolute measurement.

True Level. It is desirable to define what is called the true water level. Progressing downward from the upper region of the reactor vessel, first is found saturated steam with a constant moisture content. Then a turbulent region is approached where the moisture content increases rapidly with depth. Below this second region is found water containing an almost constant amount of voids

It is possible to define two surfaces between the top of the vessel and the core. These are the surface or steam interface above which there is steam with constant moisture content, and the second or lower surface below which there is water with a constant void percent. The thickness of the interface between steam and water is defined as the vertical distance between the steam and water surfaces. It is in this region that the steam and water are separated. The water level is defined as the position of the water surface and the steam level as the position of the steam surface.

True Level and Sight Glass Level. To verify the effect of the temperature difference between the reactor water and the sight glass water, the temperatures were measured at different points of the system (see Fig. 62-1) when the sight glass indication was 11 ft 6 in. Using these temperatures, it is possible to evaluate the error. The weight of a water column AD, in the reactor, must have the same weight as the column BD, in the sight glass system.

Fig 62-1 Temperatures at various levels in the reactor.

So that

$$P_{AD} = P_{BC} + P_{CD}$$

Where P = average specific weight of water times the height of the water column

$$BC = 716 \text{ ft } 6 \text{ in.} - 714 \text{ ft } 8.83 \text{ in.} = 21.12 \text{ in.}$$

$$\text{Average temperature} = 200^{\circ}\text{F}$$

$$\text{Specific weight} = 0.347 \text{ lb/in.}^3$$

$$P_{BC} = 7.3 \text{ lb/in.}^2$$

$$CD = 714 \text{ ft } 8.88 \text{ in.} - 708 \text{ ft } 5 \text{ in.} = 75.88 \text{ in.}$$

$$\text{Average temperature} = \frac{202^{\circ} + 310^{\circ}}{2} = 256^{\circ}\text{F}$$

$$\text{Average specific volume} = 0.339 \text{ lb/in.}^3$$

$$P_{CD} = 25.7 \text{ lb/in.}^2$$

$$P_{AD} = 33.0 \text{ lb/in.}^2$$

$$\text{Specific volume of water at } 600 \text{ psig} = 0.287 \text{ lb/in.}^3$$

$$\text{Hence, } AD = \frac{P_{AD}}{AD} = 11.5 \text{ in.} = 9 \text{ ft } 7 \text{ in.}$$

$$\text{Error: } 708 \text{ ft } .5 \text{ in.} + 9 \text{ ft } 7 \text{ in.} - 716 \text{ ft } 6 \text{ in.} = 18 \text{ in.} \pm 2 \text{ in.}$$

This evaluation of the error falls very close to the 17.5 in. which was measured experimentally.

Evaluation of Level Increase Due to Boiling. Between the moment a steam bubble is produced and the moment it leaves the water, it moves upward in the reactor water. The amount of voids in the water above the core decreases the water density but increases the water level, so that the weight of a water column remains the same and the sight glass indication is not changed. The steam content of the water is primarily a result of the time spent by the bubbles in the water and the amount of bubbles which are formed in a unit of time.

The steam flow is 90,000 lb/hr or 19.2 ft³/sec at 600 psig and 487°F, at 30 Mw

A velocity of the steam of 3 ft/sec is assumed. If a is the distance in feet between the core exit and the water surface, the time necessary for the steam to reach the surface is $t = \frac{a}{3}$.

(Neglect the amount of void in the core)

The volume of steam in the reactor water between the core and the water surface is then

$$19.2 \times \frac{a}{3} \text{ ft}^3$$

The section which the steam defines as it leaves the core and shroud has roughly a conical shape with the core surface as lower area and the vessel section as the higher area. The average surface is 25 ft.

The height of this volume without bubbles is

$$718 \text{ ft} - 713 \text{ ft } 5 \text{ in.} = 4 \text{ ft } 7 \text{ in.}$$

The increase of volume is such that

$$25 (a - 4 \text{ ft } 7 \text{ in.}) = 19.2 \times \frac{a}{3}$$

or

$$a = 73.5 \text{ in.}$$

and the increase of height $a - 4 \text{ ft } 7 \text{ in.} = 18.5 \text{ in.}$

The total error of the sight glass at 30 Mw, is then 36.5 in. which is in good agreement with the experimental measurements.

However, no importance should be given to the accuracy of this calculation. It shows that the explanations proposed for the error of the sight glass indication are reasonable.

Experimental Water Level Measurements. In order to determine the water level in the reactor vessel $\frac{3}{4}$ in. stainless steel tubes were placed in the vessel and samples of steam and water were extracted. The following properties of the samples were studied to determine if the ends of the sampling lines were in water or in steam:

1. N^{16} activity
2. Enthalpy
3. Flow rate through the line
4. Non-condensable gas content

N^{16} activity. N^{16} is produced by a fast neutron reaction in the core ($E = 12.5$ to 18 Mev)

N^{16} is an unstable nuclide which decays with the emission of a β particle and γ rays. The γ rays are mainly γ_1 and γ_2 with $\gamma_1/\gamma_2 = 12.5$.

Experimentally it is found that the activity of N^{16} in the steam and in the water per unit mass is different, being many times greater in water than in steam. In the interface, the activity is a combination of both the water and steam activity levels.

Enthalpy of the sample. The $\frac{1}{8}$ in. probe lines were insulated through a concentric tube heat exchanger. If we neglect the heat losses, we can assume that the expansion from 600 psig to atmospheric pressure is done at constant enthalpy.

Since the enthalpy of steam at 600 psig is 1203.2 Btu per lb and of water 471.6 Btu per lb., it was easy to determine whether the probe was in water or in steam.

Flow rate through the line. The sample water flow rate is such that the pressure drop in the sample line is equal to the driving pressure of 600 psig. However, the specific volumes of the samples differ greatly if the line is in water rather than steam, as shown:

	<u>Inlet</u>	<u>Outlet</u>
Pressure, psig	600	14.69
Specific vol. of water ft ³ /lb	0.0201	8.084 (30.2% steam)
Specific vol. of steam ft ³ /lb	0.7698	32.94

It is to be expected that the sample water flow rate will be different if the sample is from water or if it is steam. Experimentally a flow rate from the water line double that from the steam line was found to exist.

Noncondensable gas content Because of the neutron flux in the core, the water is decomposed in the reactor into noncondensable gases. A small portion of the gases remain dissolved in the reactor water. Most of the gas is carried away with the steam. According to measurements performed (Test Report No. 22) the amount of gas at various power levels is as shown.

<u>Condensable Gases</u>	
<u>Power level Mw</u>	<u>cc gas/liter of condensate</u>
5	26.1
10	32.0
15	35.8
20	55.1

The composition is 30.7% O₂, 66.4% H₂ and 2.9% of inerts. As these gases appear only in the steam, the presence of bubbles in the condensed sample indicates that the sampling line is in steam.

Water Loops. Nine sampling lines of $\frac{1}{8}$ in. OD were installed in the reactor vessel at different radial positions and different elevations chosen according to previous experiments. The positions of the probes are indicated in Fig. 62-1 and in Table I. Each set of probes is fastened to the top of a fuel element. A schematic of the test loop is shown in Fig. 62-2.

TABLE I - Elevation of the Probes
(Explanation of Fig. 62-1)

Probe No.	Location through reactor head	Fuel element	Position*
1A	d	77-191	40
2A	a		61
3A	a		64
4A	a		67
5A	d		100
1B	c	129-264	69
2B	d		75
1C	c	0-301	62
2C	c		68

*Positions are in inches above fuel element hole which is at elevation 713 ft 5.5 in.

The sampling water or steam passes through a high pressure valve and is cooled in a cooling coil. The condensate flows through a flat coil placed in front of the crystal of a scintillation counter. The ion exchangers which can be connected in the circuit are used for N¹⁶ studies. A rotameter placed after the coil gives an indication for the flow rate.

N¹⁶ Measurement. The counting coils are made from a $\frac{1}{8}$ -in. tube bent in a spiral shape and forming two flat layers. The diameter is 3 in. and the length of the tubing comprising the coils is 90 in. This coil is placed 3 in. above a crystal of NaI of 2 in. height and 2.45 in. diameter. The pulses generated in the photoelectric cell are analyzed in a single channel pulse height analyzer and the activity of N¹⁶ is separated. The single channel pulse height analyzer is an Argonne model. The settings used were:

High voltage - 750 v
Window level - 75/100
Window width - 1%

Amplifier gain - 2
Integral
Normal time constant

Fig. 62-2. Experimental setup.

The delay times obtained in the actual experiment are from 5 to 8 sec depending on whether the probe is in water or steam. The fact that there is a difference of 3 sec between the delay time for steam and water yields a counting rate of the steam N^{16} activity 25 percent lower than if the delay time were the same as water. This increases the observed difference of N^{16} activity in water and in steam, but does not affect the water level determination. The activities were not corrected. To eliminate the effect of fluctuation in the counting rate, counts were taken during 100 to 300 sec periods.

Enthalpy Measurements. The enthalpy was measured by a heat balance in the counterflow cooling coil. For this purpose, all sample lines, valves and the cooling coil, were covered with insulation. The temperatures were measured by iron-constantan thermocouples. Sampling water and cooling water flow rates were measured by a collecting method. The enthalpy of the sampling water approximately is the same at the entrance of the sampling line and at the inlet of the cooling coil.

If we neglect the heat losses:

$$W_C (t_a - t_b) = W_S (h_1 - h_2)$$

The pulses counted were all those corresponding to an energy higher than 4.8 Mev and were due to N^{16} activity only. As N^{16} has a decay period of 7.35 sec the time delay between the inlet of the sampling line and the counter must be reduced to a minimum. The minimum time delay would be obtained by condensing the sample in front of the counter. The measure of the mass of sampling water using this method becomes very inaccurate and absolute concentrations are difficult to determine. To circumvent this the counting coil was connected after the cooling coil. It is made of the same $\frac{1}{8}$ inch pipe to retain a high velocity of the sampling water up to the counter.

where,

W_c = cooling water flow rate

W_s = sampling water flow rate

t_a = outlet temperature of cooling water

t_b = inlet temperature of cooling water

h_2 = enthalpy of outgoing sampling water

$h_2 = t_2$ = temp. at the outlet of the sampling water

The enthalpy, h_1 , of the sample is then readily obtained.

Flow rates were already known from the enthalpy measurements and gas content was observed in the rotameter as bubbles. The presence of bubbles was also observed in the fluctuations of the N^{16} activity.

Fig. 62-3. N^{16} activity vs sight glass water levels.

Enthalpy measures presented deviations due to temperature fluctuations.

Procedure for Making the Measurements. The reactor was operated a few hours at a certain power. The water level was lowered until the line A_2 was in steam. The level was then raised step by step to make the measurements.

The power levels were 7, 10, 15, 20, 25 and 30 Mw thermal.

Results of Measurements. The measurements using line A_3 were plotted versus the indication in the sight glass for each power level.

Figure 62-3 is the N^{16} activity.

Figure 62-4 is the sampling water enthalpy.

Figure 62-5 is the sampling water flow.

The best curves are obtained with N^{16} activity and sampling water flow

Fig. 62-4. Sampling water enthalpy vs sight glass levels

Fig. 62-5. Sampling water flow vs sight glass levels.

Variation of the True Water Level with Power. From the previous three sets of curves and the elevation of line A_3 (13 ft 9.5 in.), curves between the sight glass indication and the water and steam level can be obtained. Curves 1 and 2 of Fig. 62-6 are those curves plotted versus power from the N^{16} curves.

Fig. 62-6. Difference between sight glass and true levels

The points considered as the limit of steam and of water are indicated on Fig. 62-3. The point at essentially zero power, when no bubble effect exists, was measured by slowly raising the water level at very low power (mostly decay heat) in order to compensate for only the difference in temperature between the feedwater and the reactor water.

Points considered as water and steam limits are shown in the flow rate set of curves. The water limit was found to be essentially the same as by N^{16} , but the steam limit was lower. (Curve 3 of Fig 62-6) This is explained by the larger effect of a small amount of moisture in the steam on the N^{16} activity than on the sampling water flow rate. The accuracy of these curves is ± 1 in.

Thickness of the Disengagement Layer. The thickness of the disengagement layer is determined by the N^{16} counting rate. N^{16} is the best way, because it is the most sensitive measurement of the average moisture at a certain level.

We shall consider as thickness the increase of level in the sight glass necessary to bring a sampling line from water to steam. This supposes that an increase of 1 in. on the sight glass corresponds to the same variation in the reactor vessel, which is only a poor approximation. To prove this, the N^{16} activity versus the sight glass indication for the two lines A_2 and A_3 at 10 and 30 Mw have been plotted in Fig. 62-7. These curves show the change in N^{16} activity from steam to water as the water level is raised in the reactor. To find the same effect on line A_3 as on line A_2 , the water level was raised 2.5 in. at 10 Mw and 2 in. at 30 Mw. The inlets of these lines are apart by an elevation of 3 in. This shows that when the water level is raised 1 in. in the sight glass, the true water level increases by a varying amount in the reactor vessel.

Calculations can be made proving that this 30 percent excess in the increase is due to a higher disengagement surface and to a longer time

Fig. 62-7. N^{16} activity of sampling lines A_2 and A_3 .

spent by the bubbles in the water before reaching the steam region. For a height of water without bubbles of 55 in. above the core, we have computed an increase in level of 18 in. due to bubbles at 30 Mw.

According to this, we have to restrict the value of the curves 1 and 2 of Fig. 62-6. They show only the error in the sight glass when the water surface (for Curve 1) and the steam surface (for Curve 2) are at the level of 13 ft 9.5 in.; i.e. elevation of probe A_3 , and this error increases with an increase of the true water level.

Curve 4, which is the difference between Curve 1 and Curve 2, is then an under-estimation of the interface thickness as the increase of level in the sight glass is smaller than the true increase in the vessel.

Curvature of the Water

Surface. Sampling lines were placed at different radii so that the water level could be determined at the center, at the middle radius and at the outer radius of the core. The water level given by the outer radius probes seems to be a little lower than at the center, but the difference is within the inaccuracy of the measurements. The only conclusion is that the level is essentially flat with a deviation less than 2 or 3 in.

Extrapolation to 100 Mw Operation. According to the previous discussion, the extrapolation of these results to a 100 Mw operation with the same vessel internals is very difficult. If we extrapolate the curves 1 and 2 of Fig. 62-6 to a power level of 100 Mw we find an error of 53 in. $\pm$ 5 in. above the sight glass for the steam level. This means that when the water level in the vessel is at 13 ft 9.5 in., the sight glass will indicate 9 ft 5 in. When the true steam level in the reactor is at 13 ft 9.5 in., the sight glass will indicate 7 ft 7.5 in.

It is much more difficult to calculate the water and steam level in the vessel, if the sight glass indicates the normal reading of 11 ft 6 in. It may be concluded that the reading error will be larger than 53 in. for water and 74 in. for steam.

Amplification Factor. Direct measurement of the existing water level error was limited, with few exceptions to a single and common reactor vessel water level for all powers. The water level errors reported at sight glass indications above and below the points of direct measurement were determined by employment of a theoretical amplification factor. This amplification factor is a function of void fraction in the riser section of the vessel and the water temperature in the sight glass and in the reactor vessel.

The amplification factor, F , is defined by:

$$F = \frac{\Delta h}{\Delta SG}$$

where:

Δh equals the increase in reactor water level for an increase, ΔSG in the sight glass indication.

It can be shown that:

$$\frac{\Delta h}{\Delta SG} = \frac{K}{1 - \alpha}$$

where:

K is the effect due to different water temperatures.

α is the average void fraction in the riser section.

From the above it may be assumed that the amplification factor is constant for a given reactor power and is independent of water level. Due to the uncertainty in values of void fraction and sight glass water temperature, the amplification factors could vary by at least ± 10 percent.

The water level errors at the following sight glass indications were determined by direct measurement (Table II).

TABLE II - Water Level Error

Reactor power, Mw	Sight glass indication	Water level error
0	12 ft 0 in	17 in.
5	11 ft 10 in	23 in.
10	11 ft 8 in	25 in.
20	11 ft 5 in	29 in.
30	11 ft 2 in.	31 in.
40	10 ft 11 in.	35 in.
44	10 ft 8 in.	37 in.

The steam-water interface thicknesses were obtained by applying the amplification factor to the change in sight glass indication corresponding to the change in reactor water level which is required to advance from the condition of a fixed sample probe exposed to steam to the condition of the probe exposed to water. The interface thickness was assumed to be independent of water level over the range of water levels considered. Figure 62-8 plots sight glass indication vs actual water level. Water level error as a function of reactor power is shown in Fig. 62-9. The interface thickness vs reactor power is shown in Fig. 62-10. A diagram of the reactor vessel is shown in Fig. 62-11.

Fig. 62-8. Sight glass vs actual water level.

Fig. 62-9.
Sight glass error vs reactor power.

Fig. 62-10.
Steam-water interface.

Fig. 62-11. Reactor vessel.

SECTION VIII FUEL

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
51	Evaluation of the Fuel Transfer Coffin, S. Meyers . . .	323
55	First Defect Test in EBWR, R. F. S. Robertson	325
64	Fuel Transfer Radiation Survey, C. V. Pearson, L. M. Rentschler	346
75	Probe of Fuel Channels with a 0.050 Inch Clearance Gage, D. J. Roy, E. A. Wimunc	356

EVALUATION OF THE FUEL TRANSFER COFFIN

(Test Report No. 51, by S. Meyers)

The purpose of this report is to compare calculated values vs. measured values of dose rate due to the decay gammas on the outside surface of the fuel handling coffin containing an irradiated natural fuel element.

Two different sets of calculations are considered. The first,¹ using all empirical data and relationships arrives at a dose rate of 16 mr/hr. The author notes that his method of calculation gives values of dose rate about a factor of 4 or 5 above measured values. The second and more recent calculation, uses a gamma source strength which was actually measured in the EBWR core.² Using this source strength the calculated value of dose rate is 3.4 mr/hr. Both calculated values compare very favorably with the measured value of 3 mr/hr obtained with a standard gamma survey meter.

Second Calculation. In this calculation the following assumptions were made:

- A. The energy distribution of the core decay gammas is as follows:
 - 1. 0.8 Mev - 75%
 - 2. 2.5 Mev - 25%
- B. The dose from a single natural element as determined from Test Report No. 25, 25A is 60,000 r/hr.
- C. There is linear buildup in the lead shielding.

Let S_A = total gamma source strength

then:

$$.75 S_A = \text{source from } .8 \text{ Mev } \gamma \text{'s}$$

$$.25 S_A = \text{source from } 2.5 \text{ Mev } \gamma \text{'s}$$

and

$$(.75S_A) (1.6 \times 10^{-6}) = (1.2 \times 10^{-6})S_A \text{ r/hr}$$

$$(.25S_A) (3.8 \times 10^{-6}) = (.95 \times 10^{-6})S_A \text{ r/hr}$$

¹ R. J. Rickert - Personal communication.

² EBWR Test Report No. 25, 25A

therefore:

$$(2.15 \times 10^{-6}) S_A = 6.0 \times 10^4 \text{ r/hr}$$

and:

$$S_A = \frac{6.0 \times 10^4}{2.15 \times 10^{-6}}$$

$$S_A = 2.8 \times 10^{10} \quad \gamma' \text{s/cm}^2 \text{ sec}$$

$$(.75) (2.8 \times 10^{10}) = 2.1 \times 10^{10} \quad 18 \text{ Mev } \gamma' \text{s/cm}^2 \text{ sec}$$

and

$$(.25) (2.8 \times 10^{10}) = 7.0 \times 10^9 \quad 2.5 \text{ Mev } \gamma' \text{s/cm}^2 \text{ sec}$$

In this case it is the 2.5 Mev gammas which determine the required shielding, which consists of 12 inches of lead and 1 inch of steel.

Determining the absorption coefficients:

$$\mu_{Pb} (2.5 \text{ Mev}) = .475 \text{ cm}^{-1}$$

$$\mu_{Fe} (2.5 \text{ Mev}) = .298 \text{ cm}^{-1}$$

$$t_{Pb} = 30.48 \text{ cm}$$

$$t_{Fe} = 2.54 \text{ cm}$$

and:

$$b_{Pb} = (\mu t)_{Pb} = (.475) (30.48) = 14.5$$

$$b_{Fe} = (\mu t)_{Fe} = (.298) (2.54) = .756$$

$$b_{total} = 15.3 \quad .$$

Assuming the fuel element to act as a plane source:

$$\Phi = \frac{BS_A}{Z} E_1 b_1$$

where:

B = Buildup factor

S_A = Gamma source strength

E_1b_1 = E function, determined graphically

$$\Phi = \frac{8 (7.0 \times 10^9)}{2} 3.2 \times 10^{-8}$$

$$\Phi = 895 \text{ 2.5 Mev } \gamma\text{'s/cm}^2 \text{ sec.}$$

And the resulting dose is:

$$D = (.895 \times 10^{-3}) (3.8 \times 10^{-3})$$

$$D = 3.4 \text{ mr/hr.}$$

FIRST DEFECT TEST IN EBWR

(Test Report No. 55, by R. F. S. Robertson)

PART I

A defected test specimen containing UO_2/ThO_2 ceramic fuel was irradiated in EBWR. The objectives of this test were:

1. To determine the rate of release of fission products from the defected specimen.
2. To determine the distribution of these fission products throughout the reactor system; both in the steam and liquid phase.
3. To carry out preliminary tests with the hope of developing a monitoring system which could locate the channel in which the defected specimen was situated.

Fuel Specimen. The fuel, which was prepared by the metallurgy division consisted of five pellets of compacted and sintered UO_2/ThO_2 , of composition 90.05 wt % ThO_2 and 9.95 wt % $\text{UO}_{2.1}$. The uranium was enriched to about 93 percent in U^{235} . The pellets were enclosed in a stainless steel tube, 0.305 in. ID, 0.375 in. OD, and about 3 in. long. A 0.010-in. hole was drilled through the tube at the center line. Measurements of the pellets are given in Table I.

Table I. Pellet Data

Pellet No.	Wt., g	Length, in.	Average diam., in.	Porosity	
				Closed vol. %	Open vol. %
2	6.6356	0.612	0.300	5.8	0.1
4	6.7823	0.628	0.299	5.7	0.1
6	6.6866	0.618	0.299	5.9	0
8	6.9415	0.635	0.300	5.7	0.1
9	6.6561	0.615	0.299	6.0	0.1
Total	33.70	3.11			

The bulk density of the pellets was 9.5 g/cc (94% theoretical). Data pertinent to the fuel specimen are given in Table II.

Table II. Fuel Specimen Data

Average diametral clearance, fuel-sheath	0.002 in.
End clearance, fuel sheath	0.020 in.
Volume of fuel	3.60 cc
Surface area of perpendicular surface	18.9 cm ²
Weight U ²³⁵	2.73 g
N ²³⁵	7.0 x 10 ²¹ atoms

The fuel specimen was mounted in one of the channels of a thin natural EBWR fuel element (T-7), at a position 18 in. from the bottom of the element; i.e., approximately in the position of maximum thermal neutron flux. The fuel element was irradiated in position 77-276.

Sampling Tubes and Monitoring System. Four stainless steel tubes (0.061 in. ID by 0.125 in. OD) were inserted through Swagelock fittings in the head of the reactor vessel. Each tube was placed to draw water samples from a specific location. These locations were:

Tube A	At top of fuel channel in which the defected specimen was situated.
Tube B	At top of fuel channel adjacent to that containing defect.
Tube C	Above fuel element adjacent to that containing defected specimen (i.e., above position 72-254).
Tube D	Above fuel element position 77-64, diametrically across the reactor.

Each of these tubes ended in a valve; care was taken that the length of tube to each valve was the same. The four valves were then joined to a manifold system which led to a cooler as shown in Fig. 55-1. From the cooler the tubing led to coils around the outside of a tank 1 ft in diameter and 2 ft high. This tank was filled with water and a BF_3 tube was inserted inside for delayed neutron monitoring. The BF_3 tube was connected to a linear amplifier, discriminator, and count rate meter.

Fig. 55-1. Fuel element channel monitoring system

From this tank the tubing led through a control valve to a column, 1 in. in diameter and 1 ft high, containing a strong acid cation ion-exchange resin. Valving was arranged such that the column could be bypassed if

necessary. Beyond the column there was a flat coil of tubing below which a 1 in. by 1 in. NaI crystal and photomultiplier tube was mounted. From this coil the water was returned to the delay tanks in the reactor basement. The photomultiplier tube was attached to a single channel pulse height analyzer.

Fig. 55-2. Reactor power

Irradiation Data. Figure 55-2 shows a plot of reactor power vs time for the duration of this test. The figures in brackets show reactor megawatt-hours (thermal) for each period of operation. Over the entire test the reactor operated for a total of 18,850 Mw-hr.

The parameters listed in Table III were assumed for purposes of calculation.

Table III. Test Parameters

Average thermal neutron flux in fuel	2×10^{13} n/cm ² sec
σ_f fission cross section U ²³⁵	575 barns
Heat release to reactor water	190 mev/fission
The following data may be calculated for this irradiation:	
Fission rate in specimen	8.05×10^{13} fission/sec
Power from specimen	2440 watts
Heat rate - fuel surface	129 watt/cm ² (410,000 Btu/hr ft ²)
- cladding surface	103 watt/cm ² (327,000 Btu/hr ft ²)
Core temperature*	1000°C
Total Irradiation	6.78×10^{19} n/cm ²
Burnup**	4.5% of U ²³⁵ atoms

*This is only a rough estimate based on thermal conductivity data of Kinger.¹

**This also must be considered only a rough estimate based on $\phi = 2 \times 10^{13}$ n/(cm⁻²)(sec⁻¹). A more accurate value must await dissolution and fission product analysis of the irradiation specimen.

Sampling Techniques. The following types of samples were taken during the test:

Reactor water: Samples of reactor water and of the effluent from the ion exchange resin were taken regularly for analyses for the nuclides, Cs¹³⁸, Mo⁹⁹, radio-iodines, Sr⁸⁹, and Ba¹⁴⁰. Decay curves were also determined by beta counting. Samples were also taken for gamma-ray energy analysis on a 200 channel gamma-ray spectrometer using a 3 in. by 3 in. NaI crystal.

Reactor steam: A few samples were taken for analysis on the 200-channel gamma-ray spectrometer.

Gases from air ejector: Air ejector gases were sampled regularly and were analyzed for Xe¹³⁸ and Kr⁸⁸. Methods of analysis were reported in a later publication.² Concurrently with each analysis the air ejector gases were analyzed for H₂, O₂, and inert gases, and the flow rate, as measured by a Brooks Rotameter and a Fischer-Porter flow meter, was recorded.

¹ Kinger, W. D., J. Am. Ceramics Soc. 37, 107 (1954)

² Robertson, R. F. S., ANL-6022, June, 1960.

The gas analysis was used to correct the reading of the flow meters for specific gravity of the gas. The rate of evolution of the gases was calculated as follows:

$$\text{Release rate} = 6.48 \times 10^{-10} \text{ AR (curies/day)}$$

Where A = activity of gas sample d/min/liter gas (STP)

R = flow rate of gas liter (STP)/min.

Metal samples in condenser:

Plates of aluminum, stainless steel, and copper were fastened to the uppermost tubes in the condenser prior to the start of the experiment. At the conclusion of the irradiation these specimens were removed and examined by the Chemical Engineering Division to determine whether there was any significant collection of fission or activation products. Further details on these samples must await a report from the Chemical Engineering Division.

Normal Fission Product Activities. Before describing the results of this test it is felt advisable, since they have not been reported elsewhere, to incorporate the results of analyses of the reactor water and air ejector gases carried out prior to this experiment. These data give the background fission product activities where no defected fuel is present.

Air ejector gases: Data are presented for 20-Mw operation in Table IV.

Table IV. Xe^{138} and Kr^{88} Production Rates.
No Defect

Date	Gas flow rate, l/min	Activity, d/min/l		Production Rate, curies per day	
		Xe^{138}	Kr^{88}	Xe^{138}	Kr^{88}
2-18-58	47	1.4×10^7	2.2×10^5	0.43	0.007
2-19-58	58		2.2×10^5		0.008
3-31-58	58	6.5×10^6	3×10^5	0.25	0.011
4- 3-58	~ 20	2.0×10^7	6×10^5	0.25	0.008
4- 4-58	~ 20	1.6×10^7		0.21	

Thus the average production rates were:

$$\text{Xe}^{138} \quad 0.3 \pm 0.1 \text{ c/da}$$

$$\text{Kr}^{88} \quad 0.009 \pm 0.002 \text{ c/da}$$

Reactor water samples: Samples of reactor water were taken and analyzed by conventional radiochemical techniques for various nuclides. Results are tabulated in Table V. The higher activities shown by the April 3 sample are undoubtedly due to the fact that the ion exchange flow had been turned off 60 hours previously. A typical beta decay curve is shown in Fig. 55-3 for a reactor water sample.

Table V. Fission Product Activities in Reactor Water.
No Defect

Sample	Time	Ion exchange flow rate, gal/min	Activity at sampling time, d/min/ml				
			Ba ¹⁴⁰	Cs ¹³⁸	I ¹³¹	Mo ⁹⁹	Sr ⁸⁹
3-26-58	0933	9.5	16		5	79	7
3-31-58	1009	9.5	16	148	7	89	2
4- 3-58	1010	0*	123		50	137	5

*Ion exchange flow off from 2240, March 31 to 1145, April 3, 1958, otherwise 9.5 gpm.

Fig. 55-3. β Decay curves
reactor water

Fission Product Activities with Defect. During the entire course of this test none of the monitors in the reactor system recorded any activities above normal background. These monitors included the fission product monitor on the air ejector exhaust, and gamma probes located on the steam dryer, at the air ejector, and in the condenser hot well, on the filters to the stack, and at the stack effluent.

Activity measurements carried out by the Industrial Hygiene and Safety Division showed no increase in general activity. They did find an increase in the activity of the stack gases (see following paragraphs). The conventional monitoring system for the reactor apparently could not detect the presence of this small defected fuel specimen.

Air ejector activities:

Table VI shows the measured production rates for Kr^{88} and Xe^{138} .

Table VI. Xe^{138} and Kr^{88} Production Rates
Defect Test No. 1

Date	Time	Reactor power (Mw)	Gas flow rate (l min)	Activity, d l min		Production rate, curies per day	
				Xe^{138}	Kr^{88}	Xe^{138}	Kr^{88}
4-24-58	1400	5	71	2.62×10^6		0.12	
4-24-58	1930	5	62	4.8×10^6		0.10	
4-28-58	1022	10	65	4.3×10^6		0.18	
4-28-58	1210	15	59	7.4×10^6		0.28	
4-28-58	1547	20	62	3.3×10^7		1.34	
4-29-58	1548	20	73	2.9×10^7		1.38	
5- 1-58	0925	20	65	2.6×10^7	2.4×10^6	1.10	0.10
6-11-58	0940	20.4	97	2.0×10^7		1.26	
6-11-58	1455	21.2	97	2.6×10^7		1.64	
6-11-58	1457	21.2	97	2.7×10^7		1.70	
6-12-58	1033	20.0	97	1.9×10^7	1.4×10^6	1.20	0.09
6-12-58	1400	20.9	97	2.4×10^7	1.8×10^6	1.51	0.11
6-13-58	1008	20.4	70	2.0×10^7	1.6×10^6	0.91	0.07
6-19-58	1103	20.9	61	2.6×10^7	2.3×10^6	1.03	0.09
6-19-58	1348	21.4	61	2.2×10^7	2.0×10^6	0.87	0.08
6-25-58	0933	20.6	62	1.5×10^7		0.61	
*6-26-58	1320	20.3	32	2.7×10^7	2.3×10^6	0.56	0.05
*6-26-58	2345	20.6	32	2.2×10^7	2.3×10^6	0.46	0.05
6-27-58	0837	20.6	77	1.6×10^7	1.0×10^6	0.80	0.05
6-27-58	0911	20.6	77	1.5×10^7	7.5×10^5	0.75	0.04
6-27-58	1237	20.4	77	1.4×10^7	6.5×10^5	0.70	0.03
6-27-58	1237	20.4	77	1.7×10^7	9.0×10^5	0.85	0.045
7- 1-58	1125	10.3	37	6.6×10^6	4.0×10^5	0.16	0.009
7- 1-58	1428	10.7	34	8.2×10^6		0.18	
7- 1-58	1700	11.8	48	7.7×10^6		0.25	
7- 2-58	1103	24.6	55	1.7×10^7	8.5×10^5	0.61	0.03
7- 2-58	1330	24.6	51	2.1×10^7	6.7×10^5	0.69	0.02
7- 2-58	1537	28.9	69	1.6×10^7	9.8×10^5	0.72	0.04
*7- 7-58	0921	20.2	30	2.3×10^7	1.2×10^6	0.45	0.02
7- 7-58	1550	18.9	50	1.6×10^7		0.52	
7- 8-58	0925	19.9	58	$\sim 1.2 \times 10^7$	4.2×10^5	~ 0.45	0.016
7- 8-58	1411	20.5	58	$\sim 1.0 \times 10^7$		~ 0.38	
7- 8-58	1526	20.2	55	1.6×10^7		0.57	
7- 9-58	0944	20.4	58	1.1×10^7	3.8×10^5	0.42	0.014

*Recombiner in operation.

Figure 55-4 shows a plot of the production rate of these gases as the irradiation increased. It is obvious that after about 5000 Mw-hr the production rates of both Xe^{138} and Kr^{88} fell continuously until at the end of the irradiation they were not far above the levels observed when no defect was present.

Figure 55-5 shows the production rates of these gases as a function of reactor power, the two curves shown for each nuclide are the extremes of the irradiation, i.e., after only short irradiation and after prolonged irradiation. It can be seen that, with the fresh sample, up to powers of 15 Mw, the defected sample provided but little additional activity to the system. Above 15 Mw the production rate of these gases was significantly higher, but this rate fell continuously after an irradiation of about 5000 Mw-hr.

Fig. 55-4. Production rates of Xe^{138} and Kr^{88} 20 Mw power

Fig. 55-5.

During the initial part of the test, the production rates of Xe^{138} and Kr^{88} were significantly higher than normal. This is confirmed by the Industrial Hygiene and Safety Division report (see PART II). Activity release rates shown in Table XI of this report are in general agreement.

Reactor water samples:

Table VII shows activities of various fission product nuclides determined throughout the test.

Table VII. Fission Product Activities in Loop Water, d/min/ml

Date	Ion exchange flow rate, gal/min	Activities measured				Activities corrected to 10 gal/min ion exchange flow			
		I ¹³¹	Mo ⁹⁹	Sr ⁸⁹	Ba ¹⁴⁰	I ¹³¹	Mo ⁹⁹	Sr ⁸⁹	Ba ¹⁴⁰
6- 9-58	5	19.3	96.3	8.4	30.4	9.7	49	4.4	15.5
6-13-58	7.2	85.5	82		30.1	61.5	60		22
6-19-58	6.9		90.4	5.9	30.9		64	4.1	21.3
6-24-58	6.5	24.3	138	7.5	32.2	15.8	91	4.9	21
7- 1-58	5.8	17.0	92.8		44.9	9.7	56		29.4

Since the ion exchange flow rate was dropping throughout the test, all values have been corrected to an ion exchange flow of 10 gal/min by the formula:

$$\frac{\text{Activity at } r_1 \text{ gal/min}}{\text{Activity at } r_2 \text{ gal/min}} = \frac{\lambda + \frac{r_2 \epsilon}{V}}{\lambda + \frac{r_1 \epsilon}{V}}$$

where

λ = decay constant of nuclide per hr.

V = volume of water in reactor - 3000 gal.

r_1 = flow rates to ion exchange resin

r_2 = flow rates to ion exchange resin

ϵ = efficiency of resins. Assumed to equal to 1 in all cases.

(This is justified because of the high D.F. across the resins - see Table VIII.)

In deriving this relationship, steady state activities in the reactor water are assumed.

Figure 55-3 shows a typical beta decay curve of a reactor water sample. As the test progressed the flow rate to the ion exchange resins continuously decreased and the specific beta activity of the effluent from the resins also decreased, i.e., the D.F. across the resins progressively increased. This is illustrated by the data of Table VII. Beta activities 2 hours after the sample was taken were used to obtain a common basis for comparison between the samples.

Table VIII. Decontamination Factor Across Resin Bed

Date	Time	Activity of reactor water, c/min/ml	Activity of ion exchange effluent, c/min/ml	D.F.
6-11-58	0942	4.5×10^5	1.7×10^3	265
6-11-58	1725	4.5×10^5	1.9×10^3	235
6-23-58	0926	5.2×10^5	5×10^2	1000
6-25-58	0939	5.2×10^5	6×10^2	870
6-30-58	1412	5.0×10^5	2.5×10^2	2000
7- 7-58	0918	5.0×10^5	$\sim 1 \times 10^2$	5000

Gamma ray spectra of reactor water samples were taken. Analysis of these spectra by photopeak energy, intensity, and decay showed that over 90 percent of the gamma emitting nuclides present at the time that the sample was taken could be accounted for by:

Nuclide	Half-life	Corrected to 10 gpm IX flow
Na ²⁴	15 hr	75
Mn ⁵⁶	2.6 hr	20
Co ⁵⁸	72 da	2
Cu ⁶⁴	12.6 hr	3 (not yet confirmed).

From the decay curve of Fig. 55-3 a very rough estimate of the Na²⁴ activity at sampling time of $\sim 2 \times 10^5$ counts/min/ml may be obtained. The efficiency of the counter used was about 60 percent, hence the activity of Na²⁴ was roughly 3×10^5 d/min/ml. Using the figure the activities of the other gamma emitters in the reactor water may be calculated. Results are shown in Table IX. The flow from the ion exchange resins was about 5 gal/min, having slowly fallen to this figure from the original value of 10 gal/min over the course of one month. In Table IX the observed activities are corrected to what the activity would be at 10 gal/min using the formula derived earlier.

Table IX. Activation Product Activities in Reactor Water

Sample of July 11, 1958		
Nuclide	Obs. activity, d/min/ml	Corrected to 10 gpm IX flow d/min/ml
Na ²⁴	3×10^5	1.6×10^5
Mn ⁵⁶	8×10^4	6×10^4
Co ⁵⁸	8×10^3	4×10^3
Cu ⁶⁴	1.2×10^4	7×10^3

These figures must be considered very rough but are probably accurate to ± 50 percent. Some confirmation may be obtained from the long lived tail of the decay curve of Fig. 55-3, which was taken at about the same time as the data of Table IX (ion exchange flow 5 gal/min). It will be noted that the only long lived gamma emitter of any consequence is the Co^{58} . The beta decay curve shows a long lived activity of about 5×10^3 counts/min/ml or, correcting for counter efficiency, about 8×10^3 d/min/ml. This specific disintegration rate compares well with that obtained for Co^{58} by gamma counting.

Reactor steam samples:

A sample of reactor steam was taken and gamma ray spectra were obtained. The only significant activities were:

<u>Photopeak</u>	<u>Half-life</u>	<u>Probable nuclide</u>
510 kev	114 min	F^{18}
510 kev	~ 10 min	N^{13}

The F^{18} isotope has been identified previously in the steam. N^{13} identification must be regarded as tentative as yet. The method of production is perhaps $\text{N}^{14} (n, 2n) \text{N}^{13}$, the N^{14} coming from the small amounts of N_2 present in the reactor water.

Decontamination factors between the reactor water and steam were also obtained from the gamma ray spectra. Values quoted are approximate because steam samples were not counted for a sufficiently long period. However, Na^{24} values are in good agreement with those obtained previously.

Decontamination Factors

<u>Nuclide</u>	<u>Reactor water-steam photopeak counted</u>	<u>D.F.</u>
Na^{24}	1.37 Mev	3.2×10^3
Co^{58}	810 kev	2.6×10^3

Experiments with Monitors. In none of these experiments was the activity, whether neutron or gamma ray, in any one channel significantly different from that in any other, under fixed conditions. For completeness, however, a brief description of the experiments will be included.

Delayed neutron monitoring:

Signals indicating the presence of delayed neutrons were obtained for all four channels. The counting rate, however, showed a marked dependence on flow rate, increasing sharply as the flow increased. No significant

difference in the counting rate was observed from one channel to another at a given flow rate. This would indicate that the main contribution to the neutron counting rate was from N^{17} (half-life 4.1 sec) and that any contribution from the fission product halogens was negligible.

Gamma ray scintillation counting:

For given flows, the total gamma counting rates for certain discriminator settings were recorded. Figure 55-6 shows typical results. It can be seen that there is no significant variation from one sampling value to the next; in fact water samples taken diametrically across the reactor from the fuel element containing the defected specimen appear to show the highest counting rate. This was also true for the delayed neutron measurements.

Fig. 55-6. Counting rate vs discriminator setting

At a flow of 120 cc/min and less the counting rate was independent of flow rate, but above this value it rose markedly as the flow rate increased. This is undoubtedly due to the presence of short lived N^{16} . The cation resin removes a greater proportion of high energy components than of lower energy. Again this is probably due to removal of Na^{24} (1.37 Mev and 2.76 Mev gamma energies) and also to removal of N^{16} or perhaps by allowing sufficient delay for N^{16} to decay.

Gamma ray spectra were taken of the flowing water but did not show any apparent difference from those taken on reactor water samples and counted on the 200-channel analyzer.

One interesting effect was observed when the gamma counting rate was measured with water flowing from a given valve. No resin was in the circuit. When the reactor was shut down the counting rate peaked sharply and then decayed with a short half life to a lower level. Figure 55-7 illustrates results obtained with water from valves A and D. No explanation of this phenomenon is offered, but because it occurred in water samples from widely different

points in the reactor it is not associated with the defect. It also appears to be quite a reproducible effect. Apparently some short lived activity is building in and then decays with a half life of approximately 8 to 10 sec.

This is suggestive of N^{16} but it is hard to see why it builds in on shut-down unless it has a surprisingly high neutron capture cross section and results from the decay of some shorter lived species.

Interpretation of Test Data.

The most significant feature of this experiment was that by general monitoring no rise in activity level could be detected due to the presence of the defected fuel specimen.

Activation Products:

This lack of effect is understandable, when the activities of the activation products are considered. In the reactor water, by far the most abundant activity is Na^{24} which most probably comes from $Al^{27}(n,\alpha)Na^{24}$. The Al is, of course, present in the dummy fuel elements and in addition Al_2O_3 is the main component of the corrosion product deposit on the

Fig. 55-7. Gamma counting rate after shutdown

fuel plates. Na^{24} could also come from thermal neutron activation of sodium ion in solution in the reactor water. However, to give the observed activities the amount of a sodium salt present in solution would lead to far higher specific conductivities of the reactor water than those observed during this test ($\sim 0.5 \mu mho cm^{-1}$).

The Mn^{56} undoubtedly comes from $Mn^{55}(n,\gamma)Mn^{56}$, the Mn^{55} being present in the stainless steel.

Co^{58} comes from $Ni^{58}(n,p)Co^{58}$. There is some Ni present in the Al-Ni dummy fuel plates, but the main source of the Co^{58} activity is probably from activation of the NiO present in the corrosion product film on the fuel plates. The source of this Ni is probably the Ni plate in the turbine which has shown distinct signs of erosion.

The presence of Cu^{64} in the reactor water has not yet been definitely established, but that present probably comes from thermal neutron activation of Cu in suspension in the reactor water.

The activities of activation products are rather different to previous results quoted by Goslovich *et al.*¹ A comparison is given below, for an ion exchange flow of 10 gal/min.

	Activity, d/min/ml	
	Test Report No. 55 July 11, 1958	Goslovich April-May, 1957
Na ²⁴	1.6×10^5	3.3×10^4
Mn ⁵⁶	6×10^4	1×10^5
Co ⁵⁸	4×10^3	2×10^3

It thus appears that during the year the Na²⁴ content of the reactor water has increased by a factor of 5 and the Co⁵⁸ has increased by a factor of 2.

It should be stressed that although the activities of these corrosion products in the reactor water seem fairly high, the actual concentrations of impurity they represent are negligibly small and would be hard to detect by ordinary chemical means.

Fission product release:

When the data of Table VII are compared with those of Table IX, it is obvious why the fission products released from the fuel specimen were not detected by conventional means. In the water their activities were several orders of magnitude below those of the activated corrosion products. In the steam the activities due to fission product gases were masked by the N¹⁶ activity.

We will now examine the various fission product release rates and compare them with those observed from similar tests carried out on sintered and compacted ceramin fuel.^{2,3} Our comparison must be with UO₂ since no data on the behavior of ThO₂ exists.

¹ S. J. Goslovich, *Nucleonics*, XVI, May, 1958, p. 94, 86-7.

² J. D. Eichenberg *et al.*, WAPD-183, October, 1957.

³ R. F. S. Robertson and G. M. Allison, AECL Report LCRDC-718, May, 1958.

Fission Product Gases:

If all fission produced in the test specimen were released to the reactor, it may be calculated that the release rates of Xe^{138} and Kr^{88} would be:

	Max. possible release rate, curies/day	Observed rate, curies/day	% of total released
Xe^{138}	7300	1.0	0.014
Kr^{88}	465	0.08	0.017

A very small fraction of the total available fission product gases escape through the defect.

As a basis for comparison with other tests, "escape coefficients" may be calculated. The significance and calculation of these escape coefficients may be found in the references given below.^{1,2} Calculated values are:

	Escape coefficient (sec^{-1}) this test
Xe^{138}	1×10^{-7}
Kr^{88}	1.2×10^{-8}

These values are in line with those found for defected UO_2 specimens. Values found in references 1 and 2 vary widely depending on such factors as density, surface heat rate, core temperature, etc., but for high density fuel operating at a high heat rate the values are about in the order of 10^{-8} sec^{-1} for both gases.

The reason for the continued decrease in evolution rate after 5000 Mw-hr operation is not clear. It may be due to the defect hole becoming plugged with corrosion product or it may be due to a change in the physical nature of the oxide under irradiation. If the thermal conductivity increased the average temperature would decrease leading to lower diffusion rates. Only after examination of the defect, examination of the oxide, and further experimentation can this question be answered.

Fission products in reactor water:

Table X shows the activities in the reactor water after correction for the amounts of radio-nuclide normally present in the water with no

¹ Eichenberg, op. cit.

² Robertson, op cit.

defective fuel. It is clear that the only isotope examined which showed any significant increase over background was I^{131} and perhaps Ba^{140} . Escape coefficients calculated for these two nuclides are,

<u>Escape coefficient (sec^{-1})</u>	
I^{131}	2×10^{-10}
Ba^{140}	1×10^{-11}

Uranium oxide tests have in general shown the escape coefficients for radio-iodines to be roughly the same as for the fission product gases, (i.e. $\sim 10^{-8} \text{ sec}^{-1}$) while for Ba they have been in the order of 10^{-10} to $10^{-11} \text{ sec}^{-1}$.

Table X. Net Steady State Activities in Reactor Water
All Values at 10 g/min Ion Exchange Flow

Date	I^{131}			Mo^{99}			Sr^{89}			Ba^{140}		
	Defect	Bkgd.	Net	Defect	Bkgd.	Net	Defect	Bkgd.	Net	Defect	Bkgd.	Net
6- 9-58	10	6	4	49	75	nil	4	5	nil	16	15	1
6-13-58	62	6	56	60	75	nil	-	-	-	22	15	7
6-19-58				64	75	nil	4	5	nil	21	15	6
6-24-58	16	6	10	91	75	16	5	5	nil	21	15	6
7- 1-58	10	6	4	56	75	nil	-	-	-	24	15	9

Iodine is apparently released at a lower rate from ThO_2 fuel than it is from UO_2 type fuel. This same type of behavior was seen at the BORAX reactor.

Conclusions. The results of this test lead to the following conclusions:

1. In this specimen, fission products were released to the reactor at a very low rate. These rates were in fair agreement with rates found for UO_2 fuel under similar conditions, with the exception that apparently ThO_2 retains radio-iodine to a greater extent.
2. Under continued irradiation the rate of evolution of fission product gases decreased steadily. This may be due to physical plugging of the defect hole or it may be due to a change in the physical properties of the oxide under irradiation.
3. Even when the evolution rate of fission was at its highest, it was too small to affect any normal operation activities in the EBWR plant. This was because any contribution from fission

products was overshadowed by the contribution from activation products such as N^{16} , Na^{24} , Mn^{56} , and Co^{58} .

4. The apparatus used was not capable of locating the channel in which the defect sample was located.

Further tests are now continuing in which a much larger defected specimen, 48 inches long as opposed to 3 inches, is being irradiated in EBWR.

PART II (Industrial Hygiene and Safety Division, J. J. Hartig, D. J. Sreniawski, S. J. Goslovich, J. Sedlet, and C. L. Cheever)

Reactor Engineering personnel conducted a simulated fuel defect test at the EBWR during the period April 24 through May 1, 1958. The defect consisted of a .010-in. diameter hole in a stainless steel tube which contained five $UO_2 - ThO_2$ pellets. The total weight of the pellets involved was 33.7 grams; the total weight of the U^{235} in the pellets was 2.73 grams.

The initial plans were to gain information on the stack release and the shell activities while the reactor was operating at various power levels. The duration of the test was expected to be six weeks. However, the operation was interrupted when turbine difficulties occurred.

As a result of IHS monitoring operations, information was obtained on the general shell activity, the activity release via the stack, the outdoor activity in the neighborhood of the reactor building.

Shell Activity. During the simulated defect operations, the general radiation levels from the components of the EBWR system were found to be very similar to the levels encountered during the normal operations at each power level.

The shell airborne activity was monitored by two means, by filter samples taken with a silverman sampler, and by the Nuclear Measurements Corporation Model AM-2 Continuous Air Monitor located on the fourth level of the shell. A second recorder was connected in series with the Continuous Air Monitor and placed in the control room so that the airborne activity could be observed by the operating personnel from the control room. The highest indication of airborne activity was found to be $3 \times 10^{-9} \mu c/cc$, which compares with concentrations found with normal operations. Decays followed the 32 minute half-life associated with Cs^{138} .

Stack Activity. The stack discharge was continuously monitored by a $\frac{3}{4}$ in. x 1 in. NaI crystal located approximately six in. below the attic exhaust duct. The crystal feeds a scaling and recording unit located in the EBWR control room whose discriminator level is set such that gammas above

0.5 Mev are counted. Direct calibration of the crystal (in $\mu\text{c}/\text{volume}$) was not possible; however, this detector has been useful in indicating relative stack activities.

The stack monitor response at the indicated power levels prior to and during the simulated defect operations was as follows:

Table XI. Stack Monitor Response to Various Power Levels

Thermal power	Stack monitor response in gross counts/minute		Factor increase
	Prior to defect	During defect	
5 Mw		1020	--
10 Mw	1400	1980	1.4
15 Mw	2000	3480	1.75
20 Mw	2800	7200	2.6

For additional information concerning the stack concentrations, gas samples from the shield vent system's 4 in. discharge line were obtained at each power level. The samples were taken with a two liter brass sample can and counted utilizing a NRD count rate meter and an Esterline-Angus recorder. In this manner a continuous beta decay curve was drawn from the various half-lives and activity levels were determined.

The beta decay analysis indicated half-lives of 9 minutes, 30 minutes, 90 minutes, 2.8 hours and 9.5 hours. The 30 minute activity is attributed to a mixture of Xe^{138} plus Cs^{138} (which may not be separated by decay methods alone); the 2.8 hour activity is attributed to Kr^{88} and the 9.5 hour activity to Xe^{135} .

The 9 and the 90 minute activities were attributed to the following:

The presence of the Cs^{139} has not been identified by previous experimenters due to the fact that there are no gammas reported with its decay. The isotope Ba^{139} has been previously identified.

The following tables show the stack discharge rates which have been observed prior to and during the simulated fuel defect runs. The results during the defect run represent only one sample at each power level.

Table XII. Activity Released During the Simulated Fuel Defect Runs.

	Curies per day from stack			
	5 Mw	10 Mw	15 Mw	20 Mw
Cs ¹³⁹	0.47	2.4	3.4	4.05
Xe ¹³⁸ + Cs ¹³⁸	0.27	0.64	0.93	1.59
Xe ¹³⁵		0.03	0.06	0.07
Kr ⁸⁸	0.07	0.19	0.23	0.25

Table XIII. Activities Released Via the EBWR Stack, at 20 Mw Operations, During and Prior to the Simulated Fuel Defect

	Curies per day prior to defect	During defect	Factor increase
Cs ¹³⁹	1.5	4.05	2.7
Xe ¹³⁸ + Cs ¹³⁸	0.5	1.59	3.2

During the period of the test a total of twenty-three millipore filter and charcoal samples were collected in or near the EBWR exhaust duct. A description of these samples is listed as follows:

Position	Type
No. 1	From 4" pipe - Millipore filter and charcoal U-tube in alcohol and dry ice cold trap.
No. 2	In 2' x 2' duct - Millipore filter and charcoal U-tube in alcohol and dry ice cold trap.
No. 3	In 2' x 2' duct - Millipore filter and bubbler.
No. 1A	From 4" pipe - Millipore filter and charcoal U-tube (no cold trap).

Collection was accomplished by drawing the reactor exhaust air through the filters and traps at a rate of about one liter per minute.

The samples were analyzed by chemical separation and beta counting techniques and by gamma spectrometry methods with a 200-channel analyzer. The following activities were detected in relatively low concentrations:

1. Xe¹³³ (5.2 days)
2. Xe¹³⁵ (9.2 hours)
3. Xe¹³⁸ + Cs¹³⁸ (33 minutes)
4. I¹³¹ (8.0 days)

Significant amounts of barium, strontium, cesium and cerium activities were also found. The strontium fractions constitute the bulk of the activity found thus far and also appear to contain small amounts of strontium-90.

Because of the difficulty involved in preventing xenon from escaping when the charcoal was removed from the cold trap, it was not possible to determine quantitatively the amounts of xenon isotopes which had been trapped. The charcoal sample which indicated a small amount of I^{131} activity was collected on the last day of the test. Assuming that approximately 150 ml of charcoal could be considered as a point source, the I^{131} concentration in the exhaust duct was determined to be $3.4 \times 10^{-9} \mu\text{c}$ per cc of exhaust air (which is about 1 x MPC for I^{131}).

Outdoor Activity Measurements. To assist the Background Group in the location of filters downwind from D331, smoke pots were set off in the vicinity of the stack. The pots were hoisted to the stack by means of a pulley arrangement. Instead of gradually diffusing downwind the smoke travelled directly along the sides of the dome and hit ground level at the base. Apparently this is due to the dome's shape. With southerly winds, the smoke was pulled into the control room via the annex air intake. Since the radioactive discharge would follow the same route as the smoke, this indicates that a certain amount of the stack release is drawn into the control room and shell, depending upon the wind. An air sample from the control room under these conditions showed a maximum total of $5 \times 10^{-10} \mu\text{c/cc}$ at 20 Mw. As radon and the major particulate stack activity, Cs^{138} , both decay with similar half-lives, it is difficult to say how much of this activity was due to the stack discharge and how much to the natural radon in the air. An estimate may be made by using the immediate alpha count and the alpha to beta-gamma factor as follows:

$$\text{Total beta d/m} = \text{beta d/m from } \text{Cs}^{138} + \text{alpha d/m} \times 2.5.$$

Using this in the above case the concentration of Cs^{138} is estimated to have been $4 \times 10^{-10} \mu\text{c/cc}$.

Under the present conditions the levels of airborne activity drawn into the control building represent no personnel hazard; however, in the event of a serious release this may not be the case. An extension to the stack should alleviate this situation.

Three air samplers were operated near Building 331 during the day. These samplers were placed 300 to 400 ft downwind from the reactor. One air sampler was also operated outside Building 808 at the same time as those outside of Building 331. In addition, the regular outside air samplers were operated continuously. None of the samples collected near Building 331 contained any particulate activity which can be attributed to the reactor.

Conclusions. Although the simulated fuel defect test was interrupted by turbine difficulties, the monitoring efforts of IHS personnel resulted in the accumulation of adequate information concerning activity levels in the EBWR environment. The information obtained indicated that radioactive levels inside the shell were affected very little by the added defect. Stack gas samplers and the stack monitor indicated that the radioisotopes leaving the stack were those found during the past operations and differ only in quantity. At twenty megawatts the stack discharge was found to have increased by approximately a factor of three over previous runs. In general, the activity levels at EBWR during the fuel defect test represented no personnel hazard.

FUEL TRANSFER RADIATION SURVEY

(Test Report No. 64, by M. Grothenhuis, C. V. Pearson, L. M. Rentschler)

During the November 1958 shutdown a gamma radiation survey was made of the EBWR fuel transfer process. This experiment was conducted under the supervision of Industrial Hygiene and Safety Division personnel. The reactor had been operating with a power output of 10 Mw thermal just prior to the shutdown.

The experiment was conducted for three reasons. The first was to qualitatively appraise the fuel transfer process. This necessitated flux measurements during all stages of the standard procedure. However, to fully appreciate the possible hazards, deviations from this procedure were made purposely and cautiously. Index head ports were left open, the loaded coffin was raised with the trap door open, and the fuel transfer slot was bypassed.

The second purpose was to determine areas of over and under design in the transfer equipment. Flux measurements were taken around, above, and under the loaded coffin as well as streaming surveys during fuel removal from the reactor.

Finally it was desired to gather data in a systematic and prescribed form to permit a qualitative and quantitative evaluation of shield design methods and techniques. In addition to the data compiled to fulfill the first two objects, measurements were taken in the vessel water during fuel removal, inside the loaded transfer slot, and in the fuel storage pit.

Direct reading instruments used were two Geiger-Muller counters, three "Juno" ion chambers and one Jordan radiation meter. Also utilized were three current reading chambers; a $\frac{1}{4}$ in. diameter 4 in. long ion chamber, a 2 in. diameter 10 in. long fission chamber coupled to a Keithley microammeter, Model 410, and a 2 in. diameter 12 in. long ion chamber coupled to an amplifier.

These latter three instruments were later calibrated at the Argonne high level gamma irradiation facility. None of the data from the $\frac{1}{4}$ in. chamber could be used because of a process of breakdown and recovery inside the chamber. The calibration of the other two chambers was as follows:

Fission chamber with Keithley	1.3×10^{-10} amps/r/hr
2 in. ion chamber with the amplifier (no buckling current)	4.0×10^{-10} amps/r/hr

The two fuel assemblies removed are designated as T-15 and T-13. Each contained 6 thin plates of natural uranium. The Zircaloy 2 cladding had nominal thicknesses of 0.020 in. over the faces, 0.125 in. over the edges, and 3 in. over the ends. The thickness of the uranium metal was 0.172 in.

Test Report No. 25, 25A was used as the basis for the contribution figures in the following build-up calculations:

Table I. Test Data

Element	T-15	T-13
Position	77-276	27-43
Comparable position in N.W. Quadrant	77-276	27-149
Percent contribution to power in quadrant assuming enriched-thin (ET) element	5.07	1.80
Grams of U^{235} in ET element	582.2	
Grams of U^{235} in T element	291.9	
Correction for the T element	.50	
Correction for total reactor power	.25	
Percent contribution to total power	.634	.225
Total reactor heat generation at time of installation of element (Mw-hr)	111,089.8	92,754.3
Reactor heat generation while element was in core (Mw-hr)	48,021.9	67,257.4
Total reactor heat generation at time of removal of element (Mw-hr)	160,011.7	160,011.7
Element heat generation (Mw-hr)	310	151
Element heat generation (Mev)	6.95×10^{24}	3.39×10^{24}
Mev per fission	190	
Fissions per element	3.66×10^{22}	1.79×10^{22}
Original number of U^{235} nuclei	7.46×10^{23}	
Percent burn-up	4.95	2.40

The results of the survey are contained in Figs. 64-4 through 64-10 and Table II. Figures 64-1 through 64-3, drawn to scale, are included to simplify interpretation and analysis. Convenience dictated the flux units, mr/hr and r/hr, but the location unit, the foot, was chosen to facilitate an appreciation of the distances involved. Included with the data is the date and time of the measurement along with the name of the instrument used.

Table II. Activity Levels

Element T-15 in coffin:	
Flux at top of coffin	0.5 to 1 mr/hr
Flux at side surface	3 mr/hr
Flux 1' from side surface	1 mr/hr
Element T-13 removed 16' from reactor	
Flux in air space	1400 r/hr
Flux in air space, shine from lid only	21 r/hr
Streaming under coffin	2.5 r/hr
Element T-13 in coffin:	
Flux at coffin entrance, trap door open	200 r/hr
Flux at coffin entrance, trap door closed	0.5 r/hr

Core activity traverses

Scale $1/4" = 1"$

Fig. 64-1. EBWR pressure vessel

*This is datum point from which measurements are referred

Fig. 64-2. EBWR fuel assembly

Scale $1\frac{1}{2}" = 1'$

T-15

T-13

Fig. 64-3. Index head top view

Scale $\frac{1}{2}'' = 1'$

JORDAN RADIATION COUNTER

Fig. 64-4. Radiation levels

Fig. 64-5. Core activity traverses

Fig. 64.6 Fuel removal survey

Fig. 64-9. Storage pit survey

Fig. 64-10. Storage pit survey

PROBE OF EBWR FUEL CHANNELS WITH A 0.050 IN. CLEARANCE GAGE

(Test Report No. 75, by D. J. Roy and E. A. Wimunc)

The end fitting of a probe, Fig. 75-1, which had been placed in the center channel of fuel element ET-51 (101-266) could not be removed after completion of the experiment described in EBWR test procedure No. 72. The width of this end fitting, approximately 0.375 in.* should have easily passed through the 0.438-in. channel of this element. However, a check of an unirradiated element indicated irregular channel widths. Further investigation revealed that the fuel element assemblies had not been dimensionally checked at the time of assembly. It was assumed that the welding fixture used for element assembly would assure proper channel width.

Fig. 75-1. Probe end fitting

Two rings, Fig. 75-2, were fabricated and used to probe all fuel channels on a "go-no-go" basis. The results of this probe are shown on Fig. 75-3, and Table I.

The results of the probing are inconclusive in that there is no definite pattern of narrow channels with respect to the high flux area of the core. It is therefore assumed that blistering or warpage of the fuel plates has not occurred, but rather that the fuel channel width varied at the time the elements were placed into the core.

* The exact width (0.360 to 0.375) of the probe ET-51 was measured in the cave where the element was undergoing dimensional and metallurgical checks.

A - 320 IN - HEAVY PLATE ELEMENTS
A - 388 IN - THIN PLATE ELEMENTS

Fig. 75-2. Channel probe

A LIST OF ELEMENTS AND THEIR RESPECTIVE LOCATION, IN WHICH ONE OR MORE CHANNELS WOULD NOT ALLOW FULL PENETRATION OF A 0.050" UNDERSIZE DISK, IS SHOWN IN TABLE I

Fig. 75-3. Loading chart for hood No. 52.

Table I

Element No.	Location	Narrow channels	Element No.	Location	Narrow channels
H-18	(27-219)	3	EH-41	(34-342)	3-5
EH-16	(34-250)	2	EH-47	(59-153)	4-5
EH-40	(23-188)	4	ET-42	(109-205)	4
EH-48	(54-241)	3	ET-46	(129-264)	1
EH-11	(59-276)	5	ET-54	(129-159)	1
EH-10	(44-287)	1-5	ET-33	(109- 42)	1-4
EH-27	(23-172)	2-3	ET-38	(77-402)	1
EH-44	(47-203)	1-5	EH-25	(59-381)	2-3
ET-20	(72-254)	2	EH- 5	(34-356)	5
ET-40	(77-276)	5	EH-32	(54-135)	2
ET-32	(72-308)	5	ET-12	(72-149)	1-4
EH-21	(63-312)	4	ET-53	(89-158)	3
EH-20	(47-308)	2	ET-15	(101-160)	1
ET-31	(80-237)	1	ET-56	(109- 99)	5
ET-27	(95-294)	2	ET-45	(72-414)	3
ET-23	(95-324)	5	EH-42	(54-390)	3
ET-29	(89-338)	1	H- 3	(27-361)	3
ET-36	(63-337)	3-5	ET-37	(63-126)	4
EH-37	(44-326)	3	ET- 4	(80-131)	2
EH-50	(54-179)	2	ET- 2	(89-127)	2
ET-47	(89-233)	3	ET- 3	(95-113)	2
ET-21	(109-311)	2-5	ET-41	(89- 53)	1
ET-52	(109-359)	5	EH-18	(63-418)	2-5
ET-11	(89-370)	5	EH- 6	(44-392)	1
ET-35	(72-360)	3-5	EH-13	(47- 97)	5
EH-30	(54-347)	1	ET-28	(72- 96)	5
EH-35	(27-325)	2-5	ET-34	(77- 85)	2
EH- 1	(34-145)	2	ET- 6	(72- 43)	5
EH-19	(59-170)	2-4	EH-36	(44- 75)	1-3
ET-49	(77-191)	2-5	EH-38	(59- 64)	4
ET-55	(109-254)	1	EH- 4	(59- 48)	4
ET- 5	(129-370)	4	EH-33	(44- 9)	1
ET-50	(109-417)	2	H- 9	(27- 43)	5
ET-22	(95-400)	4-5	EH-29	(34- 39)	1
EH-14	(59-365)	1	EH-26	(27- 7)	3

SECTION IX WATER CHEMISTRY AND CORROSION

<u>Test Report No.</u>	<u>Title</u>	<u>Page</u>
13A	Water Technology of EBWR During Initial Operation, C. B. Zitek	361
15	Ion Exchange Column No. 1 Resin Removal, V. C. Hall	376
22	Water Decomposition as a Function of Reactor Power, R. E. Bailey	377
22A	Water Decomposition at Varying Reactor Powers and Pressures, R. E. Bailey, E. W. O'Keefe	379
29	Deposits from a Fuel Element and Dummy Fuel Elements, C. B. Zitek	386
29A	Scale on Fuel Elements in EBWR, R. H. Leyse	389
29B	Test for Scale Removal from Fuel Assembly H-18, I. Charak, R. H. Leyse	392
29C	Post-irradiation Heating of Fuel Element H-18, R. H. Leyse	401
29D	Visual Inspection of Fuel Assemblies During Core Loading, I. Charak	408
30	Deposits on the Steam Side of the Condenser and Steam Dryer, C. B. Zitek	410
37	Reactor Steam Purity by Radioactive Sodium Tracer, J. Barrett, H. Cadoff	415
66	Effect of H ₂ Addition to Reactor Feedwater on Radiolytic Decomposition, G. E. Goring, A. P. Gavin	419
73A	Measurement of Suspended and Dissolved Solids in EBWR Water, N. R. Grant	427
79	Corrosion of EBWR Steam Line Samples, N. R. Grant	429
104	Radiochemical Analysis of Samples from the EBWR Shock Shield, R. J. Armani	438

WATER TECHNOLOGY OF EBWR DURING INITIAL OPERATION

(Test Report No. 13A, by C. B. Zitek)

During the initial period of operation the subjects of primary importance, such as reactor water quality, water decomposition rate, and activity carryover (decontamination factor), were investigated.

This report presents all of the data which has been obtained regardless of whether the results are considered satisfactory. The following subjects are covered:

1. Water quality
2. Reactor water purification system operation
3. Water decomposition
4. DF (decontamination factor) data
5. Dissolved oxygen in the system
6. Make-up demineralizer operation
7. Retention tank waste disposal

Radiation levels of the external components, which are a direct result of reactor water quality, has been covered. The air ejector gases will be discussed in a subsequent memo when identification of radioisotopes is completed.

The data presented in this report was obtained through the cooperation of many of the Reactor Engineering personnel. The water decomposition data was obtained by Reactor Engineering personnel.

Water Quality. Table I gives the EBWR water quality data from the critical runs to the present time. After washing and flushing the reactor vessel and flushing external piping the reactor vessel was filled to within 3 to 4 feet of the vessel flange with demineralized water (0.1 micromho conductivity) for the critical test. Daily checks were made to determine whether any boric acid had leaked in from the high pressure boric acid system. No boric acid was found, but the conductivity had increased to about 4 micromhos due to gas solution, (probably O_2). Then 11.250 grams of boric acid was added to get physics data and the reactor vessel was drained, flushed and refilled with demineralized water three days later. The reactor water was circulated through the reactor water purification system and no boric acid was determined.

At the time of startup, the water was about 600,000 ohms/cm³ by the addition of poor quality water from the condensate hotwell. The reactor water purification system brought it up to about 5 Meg ohms/cm³ during the run. At the start of the run (8 Mw) the steam had a conductivity of about 1 micromho but later when at 20 Mw it was about 2-3 micromhos (250,000 ohms/cm³). The activity in the steam dryer was increasing also and the subsequent decontamination factor determination was only 2.04, equivalent to steam quality of about 50 percent.

Table I - EBWR Water Data

Date	Time	Sample designation	pH	Conductivity		Dissolved oxygen, ppm	H ₃ BO ₃ Conc. g/l
				Measured *ohms/cm ³ X1000	Recorded **micromho		
11-29-56	1600	RW	7.3	250			none
11-30-56	1730	RW	7.25	540			none
12-1-56	0720	RW	7.2	470			none
12-2-56	1100	RW	7.2	510			none
12-3-56	1530	RW	7.2	540			1 ppm
12-17-56	0030	RW		300			none
	0030	AIX 01		350			
	0200	RW		530			none
	0200	AIX 01		650			
12-23-56	1745	RW	7.1	160			
		APF	7.0	160			
		St. A. D.	7.0	840			
12-23-56	1845	RW	7.0	180			
		AIX	7.05	280			
		Cond.	6.8	370			
		St. A. D.	7.0	250			
12-23-56	2000	St. A. D.	7.1	250			
12-24-56	1445	RW	7.1	570			
		AIX	7.0	980			
12-29-56	0420	RW	5.9	200			0.353
	0415	Cond.		1200			0.019
	0410	St. A. D.		900			0.019
12-29-56	2220	RW	5.2	90			0.748
	2215	Cond.	6.2	1500			0.038
	2210	St. A. D.	6.15	1200			0.038
1-8-57	2110	RW			20		0.735
1-9-57	2300	RW			20		0.548
1-10-57	0020	RW	4.8	38	20		0.534
1-10-57	0415	RW	4.4	38	20		0.371
	0420	AIX	6.9	1200			0.0058
1-10-57	0700	RW	4.8	65	20		0.217
	0700	AIX	7.4				0.0071
	0700	APF	4.8				0.213
1-10-57	1235	RW	6.9	45	13		0.0985
	1250	Cond.	6.9				0.012
	1250	St. B. D.					0.012
1-10-57	1700	RW	7.2	150	6.6		0.044
	1700	AIX	7.3	1870			
1-11-57	0825	RW	5.8(?)	410	3.3		0.0083
1-11-57	1300	RW	5.2(?)				
	1300	AIX	6.9				
1-11-57	1630	RW	5.0(?)				
2-11-57	1917	RW	7.0	760	1.1		
	1917	APF		900			
	1918	AIX		1570			
	1930	Cond.		1820			
	1933	St. A. D.		1150			
2-12-57	1200	RW	7.15	640	0.9		
	1200	AIX	7.05	1700			
2-12-57	1530	RW	7.10	(flow cell)	0.9		
	1530	AIX	7.22	(flow cell)			

Table I (Cont'd.)

Date	Time	Sample designation	pH	Conductivity		Dissolved oxygen, ppm	H ₃ BO ₃ conc. g/l
				Measured, *ohms/cm ³ X1000	Recorded **micromho		
2-12-57	2345	RW	7.0		0.8		
	2345	Cond.			0.1	0.005	
2-13-57	1030	RW	7.1		0.9		
	1030	Cond.			0.1	0.005	
2-13-57	1920	RW	7.08		0.9	0.37	
	1925	Cond.				0.005	
		Cond. St. Sx.				10.3(?)	
2-14-57	2110	RW	7.02		1.0	0.32	
	2120	Cond.				0.002	
	2120	Cond. St. Sx.				3.5	
2-18-57	1115	RW	7.3		0.8	0.30	
	1120	Cond.				0.001	
		Cond. St. Sx.				3.7	
2-20-57	1000	RW	6.9		0.9	0.30	
	1010	Cond.	6.9			0.004	
		Cond. St. Sx.				4.4	
		Cond. (South Half)			0.1		
		Cond. (North Half)			0.6		
3-7-57	1530	RW	7.0		0.9	0.29	
4-4-57	1245	RW	7.2	1800	0.55		
		Cond.	7.0	3200	0.312		

Sample designation abbreviation

- RW - Reactor water
- APF - After prefilter
- AIX - After ion exchange column
- Cond. - Condensate (from feedwater line)
- St. B.D. - Condensed steam - before steam dryer
- St. A.D. - Condensed steam - after steam dryer
- Cond. St. Sx. - Condensed steam during gas sampling

*Determined in the laboratory.

**Recorded on Conductivity Recorder.

The explanation of this condition was that reactor water was being forced up the 4 in. pipe between the startup heater and the steam dryer by the pressure drop through the steam piping to the dryer. The condition was immediately corrected by installing a valve at this point.

To simulate full poisoned conditions 19.8 pounds of cp boric acid were added to the water. The pH decreased to 5.2 and resistivity was 90,000 ohms/cm³. Originally it was planned to operate the reactor at 20 Mw for one or possibly two days, but the reactor boric acid remained in the system for about twelve days. During this time the cleanup loop was not operated and the water conditions were poor, pH of 4.4 and only 38,000 ohm/cm³(over 20 micromhos) with high crud content. No determination of the insoluble solids was obtained.

The boric acid was taken out by the mixed bed ion exchanger column a little at a time so that at the end of a three day period only 8.3 ppm remained of the initial 748 ppm. Total operating time of the reactor water purification system was about 20 hours at about 8.5 gpm. Boric acid removal efficiency of the ion exchanger column was about 98 percent. The designed capacity of the cleanup system is much greater than needed for continuous 20 Mw operation since the 38,000 ohm/cm³ water was increased to one megohm/cm³ while operating at full power. The flow rate required to maintain one megohm water has not been determined as yet. The filters and ion exchange column were bypassed for 11 hours to increase the activity of the reactor water for decontamination factor data and the conductivity increased from 0.9 to 1.2 micromhos

The question of boric acid carryover in the steam was brought up. Although it was surmised that there might be some vaporization the answer was somewhat of a surprise. The decontamination factor for boric acid was only 18 compared to an activity decontamination factor of about 4.5×10^3 . However, since the volume of the external system is only about 400 gallons (the condensate is returned to the reactor) the reduction in boric acid concentration in the reactor vessel was small (0.7%), and the effect on the physics data obtained was small.

Continued operation of the reactor water purification system brought the conductivity down from 3.0 to 1.0 micromho in about 30 hours. As can be seen from Table I, the conductivity was between 0.5 to 1.0 micromhos during reactor operation since that time. The pH has varied from 6.9 to 7.2 which agrees very well with the predicted conditions (pH 7.0 and < 1.0 micromhos).

The pH of the steam has been between 6.9 and 7.1 with a conductivity of < 0.1 micromho during normal operation. The conductivity of the north half of the condenser hotwell increased gradually to 0.6 micromho and a possible condenser tube leak was looked for. Chemical analyses of samples from both halves showed no difference in Na or Cl which would not be expected due to the high sodium phosphate content of the cooling water. Spectrographic analyses showed no great difference either and the trouble was finally found in the conductivity cells. The cells were replatinized and conductivities during the next period of operation were < 0.1 micromhos.

A great deal of trouble has been experienced with the conductivity cells in the reactor water purification system. It would be advisable to investigate polished stainless steel probes, and if new cells are put in, the replaceable type should be used so that the equipment would not have to be drained to take out the cells.

Work has been initiated to determine the total solids and insoluble solids in the reactor water. No one on the site was equipped to make this determination on high quality water so a continuous evaporation system is

being built. Another approach to getting total solids has been initiated. A reactor water sample was sent to Chemistry to get a quantitative spectrographic analysis. The rough calculation shows the total solids to be less than 0.5 ppm. This answer is questionable since anions were present and it was not known in what form the metal ions were. The individual metals were calculated to the oxides and added together.

A considerable amount of work had been done to determine the radioactive isotopes present in the reactor water. This work was started by determining decay curves to see if they corresponded with Borax III work. Instead of tailing out to primarily Na^{24} (15 hr $\frac{1}{2}$ T) they leveled out with a 22 to 26 hr half life. An investigation showed most of the activity to be strontium which did not decay as Sr^{89} or Sr^{90} . It was determined to be a mixture with 9 hour Sr^{91} being present.

It was learned that Chemical Engineering was doing work on identification of mixed radioactive isotopes with a gamma spectrometer and arrangements were made to examine a reactor water sample. The final report has not been received at the time of this writing, but a preliminary report is as follows:

99% of gamma activity is emitted by:

Mn^{56} 2.6 hr $\frac{1}{2}$ T - 820 kv greatest activity

Sr^{91} and Y^{91} - 500 kv next greatest

Na^{24} 15 hour $\frac{1}{2}$ T - 1.4 Mev next greatest.

It was known that Mn^{56} and some Na^{24} were present, but the mixture of Sr^{91} and Y^{91} shifted the decay curve. The Mn^{56} is formed from a n - p reaction on Fe^{56} , the Na^{24} from a n - a reaction on Al, and the Sr^{91} from a n - a reaction on Zr^{91} .

Soon after the original startup, higher than normal background was noticed by several low-counting laboratories on the site. Samples of gas were taken from the air ejector discharge and reactor vent system and counted. Cs^{138} was identified immediately and the fission gases $\text{Xe}^{138,135}$ and $\text{Kr}^{89,90,91}$ were subsequently identified. This, of course, meant fission products were in the reactor water. A series of four reactor water samples were checked for fission products. Cs^{138} , Sr^{89} , Ba^{139} , and Mo^{99} were found. From the results of the four tests it appeared that the production had reached equilibrium conditions.

Using data on neutron flux, volume of reactor water, and cleanup loop operation, it was calculated that between 30 and 35 mg of 1.4 enriched U^{235} was exposed to the reactor water (assuming 20% accuracy). The Cs, Sr, and Ba could be daughters of fission gases Xe and Kr, but Mo^{99} is a direct fission

product as could be the Ba^{139} . This indicated that uranium was somewhere outside the cladding. Reactor water samples were also checked. Mo^{99} was found in both samples and the uranium contents were the same (0.005 micrograms/ml). Using this figure the amount of uranium in the reactor water would have been 59 mg. No attempt was made to determine the activity of reactor water in terms of microcuries per liter.

The explanation of the uranium appeared to be in the surface contamination of the fuel plates with uranium oxide. During autoclaving of the fuel plates two or three failures were experienced. Although the autoclaves were decontaminated, subsequent plates could have been contaminated and it is not certain how well the plates which had been in the autoclave at the time of the failures were decontaminated. Another possible explanation which is being investigated at the present time is the uranium content of the Zircaloy itself. A report is that 1 ppm of U is present.

Reactor Water Purification System Operation. The efficiency and operation of the reactor water purification system from an actual cleanup viewpoint has been very good. However, considerable difficulty has been experienced in attaining temperatures below 110°F going to the ion exchange bed.

The reason for the temperature difficulty appears to be that the regenerative heat exchangers were piped for parallel flow rather than counter flow. Piping changes are being started to correct that condition. An interesting side light to the parallel flow in these coolers is that the heat transfer takes place as shown below:

	<u>Tube side</u>	<u>Shell side</u>
Temperature in, °F	488	85
Temperature out, °F	272	300

It was thought that the trouble was in the thermocouples but they were calibrated and found to be accurate. This phenomenon is possibly caused by a flow condition resulting from the helical type tubes in the shell.

During initial operation unit no. 1 was cooling the reactor water down to about 80 to 90°F, but unit no. 2 reduced the temperature to 125°F with about a half flow of 4 to 5 gpm. Unit no. 1 was used initially and has been used almost exclusively so that a clean ion exchange bed is available for use when no. 1 bed is exhausted. It appears that unit no. 1 cooling system is crudding up as it is difficult to maintain a temperature below 110°F with decreased flow.

Each ion exchange column in the reactor water purification system contains 6 cu ft of highly basic anion resin (1RA-400 Type) and 3 cu ft of cation resin (1RA-120). The mixing of these resins in the basket to obtain

high quality effluent presented a problem. The original mixed bed gave an effluent only slightly better than the influent. Both units had to be taken down and remixed carefully to form a good bed. The effluent was satisfactory as can be seen on Table I.

The reactor water purification system was put to a test when the boric acid and other dissolved solids had to be removed from the reactor water during 20 Mw operation. The water quality was increased from 38,000 ohms/cm³ to 410,000 ohms/cm³ resistivity in about 25 hours operation of unit no. 1. This unit has been in use most of the time since then and is still giving high quality effluent.

The pre-filter units have performed their part in the reactor water purification system very well. During the boric acid run, the amount of insoluble crud in the reactor water was high and most of the activity was deposited on the no. 1 pre-filter (see Table II). During this run the after-filter became radioactive also (2.0 r at 4 in.). This had not been anticipated. An explanation is that colloidal particles which passed through the pre-filter were agglomerated by going through the ion exchange column and picked up on the after-filter.

Table II - Radiation Levels in Filters and IX Column
in Reactor Water Purification System

Date	Time	Remarks	Pre-filter (thru part in shielding) r/hr	IX column (thru part in shielding) r/hr	After-filter r/hr
1-10-57	1300	Start of cleanup	6	2.9	
1-11-57	1300	during boric acid	2	1.4	
1-12-57	1300	run	1.6	1.0	2.0 at 4 in.
1-23-57	1450	Startup after	0.55	0.25	
1-23-57	1800	being down since 1-12-57	2.5	0.350	
3-5-57	2100	Startup after	0.09	0.04	
3-6-57	1200	being down since	0.24	0.12	
3-6-57	2400	2-20-57	1.8	0.31	
3-7-57	0200		1.9	0.34	
3-8-57	1400		1.0	0.64	2.0 at 4 in.
3-8-57	2400		0.5	0.63	
4-10-57		Normal run	0.45	0.40	0.1
		Average values	0.5 to 1.0	0.5 to 1.0	0.1

Both the pre-filter and after-filter of unit no. 1 reactor water purification system were reloaded with new filter cartridges because of the high activity and because a lot of insoluble crud had been taken out of the reactor water. The cartridges from the pre-filter were a dark red-brown color while those from the after-filter were a light brown color.

A radiochemical analysis was made on a sample from the pre-filter. Fe^{59} and Co^{60} were the major activities. Minor activities in decreasing amounts were Ni^{59} , Cr^{51} , Zr^{59} , Ce^{144} , and Zr^{65} . Small amounts of Sr^{89} , Ba^{140} , Mo^{99} , and Cs^{138} were found but not identified specifically. Nb^{95} was also identified.

The after-filter of unit no. 1 became radioactive during subsequent operation, but decayed out to about 100 mr. at the surface. During the startup operation it came up to 2r again. New filter cartridges were put in and a sample is to be analyzed by Chemistry Division to identify the activity. Another sample from a pre-filter cartridge is also being analyzed. It may be necessary to build a temporary concrete block wall around the filters if the activity pickup continues. The pressure drop across the filters was determined, and also across the ion exchange columns. When the reactor was down the filters were changed.

The pressure drop data was taken at atmosphere pressure and was as shown in Table III.

Table III - Pressure Drop Data

Pressure drop, pounds	Unit No. 1	Unit no. 2	Unit no. 2
	4-22-57	4-22-57 (dirty)	4-23-57 (clean pre-filter)
Across pre-filter	3.25	8.9	1.6
Across ion-column	6.75	4.7	6.2
Across after-filter	2.0	1.7	2.5
Water flow, gpm	5.5	5.7	6.8

A periodic check should be made of pressure drops across the units at each shutdown following an extended period of reactor operation.

The activity levels of these units under various operating conditions are as given in Table II. The activity of the ion exchange column was checked at the three ports in this vessel. The activities were 1.9 r at the top, 200 mr in the center, and 100 mr at the bottom port. This indicates that the bed was not exhausted during the cleanup of reactor water.

Only one good set of decontamination factor measurements across the pre-filter and ion exchange column was taken. The activity data is given in Table IV. The decontamination factor across these units will vary depending on the condition of the reactor water and this individual set of data should not be used as an overall criterion of their operation.

TABLE IV - EBWR Decontamination Factor Data
(Activity in Reactor Water/Activity in Steam or XSs)

Date	Time	Sample designation	cc evaporated	Decay time hours	Activity		Decontamination factor (DF)	Conditions of test			Remarks
					Geiger tube	Proportional counter		Power level Mw	Water data		
									c/m/ml	pH	
12-23-56	1745	RW	1	19	358				7.1	6.2	
	1745	St A D	11	19	0.14		2.5 x 10 ³	8	7.0	1.2	
	1845	RW	1	18	482			20	7.2	5.5	
	1845	St A D	1	18	235		2.1 x 10 ³		7.0	4.0	
	1845	Cond	1	18	197		2.4 x 10 ³		6.8	2.7	
12-29-56	0420	RW	1	36	131			20	5.9	5.0	Boric acid added Reactor started at 0005
	0410	St A D	8	36	0.29		4.5 x 10 ²			0.5	
	0415	Cond	8	36	0.36		3.6 x 10 ²			0.6	
	2220	RW	1	19	1432			20	5.2	12.0	Reactor on since 1640
	2210	St A D	10	19	0.10		1.4 x 10 ⁴		6.15	0.5	
	2215	Cond	10	19	0.28		5 x 10 ³		6.2	0.5	
1-10-57	1235	RW	4	21	4922			20	6.9	22.2	
	1250	St B D	10	21	1.06		4.9 x 10 ³				Steam quality 99.98
	1300	St A D	10	21	0.61		8.1 x 10 ³				Steam quality 99.99
	1250	Cond	10	21	2.95		1.6 x 10 ³		6.9	0.1	
1-12-57	0030	RW	4	23	1037			20	7.0	2.0	Boric acid out of reactor water
	0030	St A D	10	23	0.64		1.6 x 10 ³				
	0030	Cond	10	23	0.73		1.4 x 10 ³				
	0030	A P F	4	23	772		1.35				
	0030	A I X	10	23	8.49		1.2 x 10 ²				
2-18-57	1125	RW	4	37	452.3			20	7.3	0.8	
	1100	St A D	40		0.075		6.0 x 10 ³				
	1110	Cond	40		0.153		3.0 x 10 ³			<0.1	
	1125	RW	4	37		2372					
	1100	St A D	40	37		0.377	6.3 x 10 ³				
	1110	Cond	40	37		0.993	2.4 x 10 ³				
2-20-57	0042	RW	4	22	914			20	6.9	0.9	
	0033	St A D	80	22	0.187		4.9 x 10 ³				
	0040	Cond	80	22	0.362		2.5 x 10 ³			<0.1	
2-20-57	0945	RW	1	38	471.6			20	6.9	0.9	
	1030	St A D	60	38	0.373		1.3 x 10 ³				
	1000	Cond	60	38	0.828		5.7 x 10 ²			<0.1	
2-20-57	0945	RW	1	38		6479					
	1030	St A D	60			4.01	1.6 x 10 ³				
	1000	Cond	60			9.05	7.2 x 10 ²				
3-5-57	1700	RW	4		87.6						
	1700	St B D	80		0.087		1.0 x 10 ³				Steam quality 99.92
	1700	St A D	80		0.058		1.5 x 10 ³				Steam quality 99.93
3-10-57	2005	RW	4	74	384			20	7.0	1.2	
	2005	St B D	80	74	0.150		2.6 x 10 ³				Steam quality 99.96
	2005	St A D	80	74	0.081		4.7 x 10 ³				Steam quality 99.98
	2005	Cond	80	74	0.118		3.3 x 10 ³			<0.1	
3-10-57	2005	RW	4	74		2807					
	2005	St B D	80	74		1.27	2.2 x 10 ³				
	2005	St A D	80	74		0.504	5.6 x 10 ³				
	2005	Cond	80	74		0.806	3.5 x 10 ³				

Sample Designation (see Table I for Designation)

Water Decomposition. Water decomposition rate data from EBWR was of great interest to many people and an effort was made to get some data. Twelve of fifteen sets of determinations (those with low inert contents), were averaged to give 55.1 cc gas/liter of condensed steam. All of the data is given in Table V. The analysis was 30.7 percent O₂, 66.4 percent H₂O, and 2.9 percent inerts. This is about $\frac{3}{4}$ of that obtained at a pH of 7.0 from Borax III, 77 cc gas/liter analyzing 32 percent O₂, 60 percent H₂, and 8 percent inerts.

Table V - EBWR Water Decomposition Data

Date	Time	Power level	Reactor water quality				Gas yield cc/liter*	Condensed steam gas analysis			Air ejector gas flow cfm**	Air ejector gas analysis			Remarks
			Mw	pH	Determined ohms/cc	Recorded micromhos		% O ₂	% H ₂	% Inerts		% O ₂	% H ₂	% Inerts	
12-23-56	0115	20	7.0	180,000		50	22.7	47.0	30.3**		25.2	32.8	42.0	Steam by-passed to condenser	
12-23-56	2015	20	7.0	180,000		69	20.4	44.5	35.1**		22.6	27.2(?)	50.2	Steam going to the turbine	
12-29-56	2210	20	5.2	90,000		53	30.2	55.9	13.9		29.4	56.1	16.1	Boric acid in reactor water - .748 g/l	
1-11-57	1630	20	5.0(?)	410,000		47.5	29.7	68.6	1.7					Only 8.3 ppm of boric acid in reactor water	
1-30-57	1715	20	7.05	900,000		66	31.0	63.4	5.6		30.4	53.9	15.7		
2-11-57	1917	20	7.0	760,000		52.5	30.0	66.2	3.8		28.2	47.3	24.5		
2-11-57	2330	20	7.0	600,000		46	29.2	62.4	8.4		28.9	44.6	26.5		
2-12-57	1300	20	7.15	640,000		45.7	29.3	69.2	1.5						
2-12-57	2020	20	7.2		0.8	47.8	30.2	67.3	2.5		29.5	48.0	22.5		
2-13-57	1030	20	7.1		0.9	58.5	31.6	66.6	1.8						
2-13-57	1920	20	7.08		0.9	55	30.0	66.0	4.0		29.9	47.5	22.6		
2-14-57	1200	20	7.1		1.0	57.8	31.1	66.5	1.8		29.9	49.7	20.4		
2-14-57	2110	20	7.02		1.0	63.3	32.5	67.5	0.0	1.2	29.7	51.8	18.5		
2-18-57	1115	20	7.3		0.8	59.8	31.7	67.0	<1.3	1.0	30.5	56.0	13.5		
2-20-57	1000	20	6.9		0.9	58.0	31.7	66.2	2.0	0.92	30.9	56.1	13.0		
Average of samples of 1-11-57 thru 2-20-57						55.1	30.7	66.4	2.9		29.7	50.5	19.7		

*Not corrected for gases dissolved in the condensed steam.

**Values are high due to transfer of sample or actual nitrogen in system.

***Not corrected for gas density - value reported as cfm air at 80°F.

None of the data has been corrected for the gases O₂ and H₂ dissolved in the condensed steam sample. Additional work will be done to obtain the dissolved H₂ figures.

The decomposition rates were not constant, varying from 45.7 to 66 cc/liter. There was some question about the steam sampling probe used. Another reason for questioning the results is the lack of correlation between the decomposition rate and that calculated from the air ejector gas analyses and flow rate. According to these calculations the decomposition rate was about 50 percent lower than calculated from air ejector flow. The same difficulty in trying to balance out the gas flows was experienced in Borax III where the gas flow rates were in question. The flowmeter in EBWR had been calibrated and the actual air inleakage measured.

One determination was obtained which throws some light on water decomposition rate vs pH of the reactor water sample. With the boric acid concentration high (.748 gms/liter) the pH of the reactor water composition was 5.2 with a resistivity of only 90,000 ohms/cm³ (11 micromhos).

The decomposition rate was 53 cc/liter. This is very close to the average 55 cc/liter. According to the curve of Borax III data the rate should have been about 135 cc/liter. The change in Borax III data was probably due to the low Cl ion concentration (about 2 ppm) at a pH of 5.2; however, this does not explain a decrease in rate from a pH of 6.2 to 8.0 where no chlorides were present.

Decontamination Factor - Carryover in Steam. Determination of the decontamination factor difference between the reactor water and the steam was another item that was pushed during the initial start-up of EBWR.

Several problems encountered in obtaining decontamination factor data for Table IV were the low activity of the reactor water and obtaining proper decay curves of reactor water and steam and condensate samples to assure counting of the same radioisotope.

Based on six sets of counting data the average decontamination factor after the steam dryer was found to be 5.9×10^3 (4.7 to 8.1×10^3). This data is in close agreement with the decontamination factor data on Borax III at high power levels of 4 to 8×10^3 .

The decay curves of the reactor water, steam, and condensate have different initial slopes and do not attain the same slope until after about 20 hours decay. This has a definite effect on the decontamination factor and the change with time is shown in Table VI. Figure 13A-1 shows the slope during initial decay period and Fig. 13A-2 the decay during a three day period after sampling. This is similar to Borax work except that Borax samples leveled out with a 15 hour half-life (Na^{24}) while EBWR levels out with a 22 to 28 hour half-life. It was recently learned that this is probably a mixture of Na^{24} and Sr^{91} with Y^{91} building in.

Table VI - Change in DF with decay time

Date	Time	Decay time	DF of steam sample	DF of condensate sample
1-12-57	0030	Samples taken		
	0212	$1\frac{3}{4}$ hours	1.08×10^2	1.36×10^2
	0300	$2\frac{1}{2}$ hours	1.26×10^2	1.57×10^2
	0402	$3\frac{1}{2}$ hours	1.58×10^2	1.77×10^2
	0556	$5\frac{1}{2}$ hours	2.28×10^2	2.5×10^2
	0801	$7\frac{1}{2}$ hours	2.91×10^2	3.2×10^2
	1011	$9\frac{3}{4}$ hours	4.14×10^2	4.66×10^2
	1157	$11\frac{1}{2}$ hours	5.54×10^2	
	1615	$15\frac{3}{4}$ hours	1.73×10^3

Fig. 13A-1
Initial decay curves

Fig. 13A-2. Three-day decay

Performance data on the steam dryer was of interest and three sets of samples were taken to get this data. The average increase in decontamination factor across the dryer was a factor of 1.66. The performance guarantee is expressed in steam quality and the decontamination factors were used to calculate steam quality. To do this a number of assumptions had to be made:

1. That the activity counted was due to true droplet carryover.
2. The concentration of the activity in the droplets was the same as in the reactor water - no preferential carryover or concentration of solids (soluble or insoluble) near the steam-water interface.
3. The activity counted in the reactor water and steam samples were the same radioisotopes or a mixture of isotopes (see Fig. 13A-1).

The results of these calculations are given in Table IV. In each case the steam quality was improved by only 0.01 or 0.02 percent. The fact that the steam quality ahead of the dryer was above 99.92 percent in all cases indicates that a boiling reactor having the same free space between the water surface and steam outlet point and the same type of steam collecting ring would not need a steam dryer in the system. A steam collecting ring is not even needed since the Borax reactor vessel did not have one and the steam quality was 99-99.9. However, there is one point in favor of having a steam dryer i.e., the amount of activity is reduced by a factor of 1.66.

In all the data the decontamination factor of the condensate is lower than that of the steam. This is due to the fact that all drainage is dumped to the condenser. Because of this it is difficult to obtain data showing what proportion of the activity in the steam is deposited in the turbine or condenser.

Mention was made previously that the activity of the reactor water was too low to get good decontamination factor data. In an attempt to obtain better data the reactor water purification system was turned off for 11 hours and the activity was increased by a factor of two. The resulting decontamination factor value was 4.7×10^3 .

Three sets of samples were counted with both the proportional gamma counter and the open end Geiger tube counter and the resulting decontamination factor figures were in close agreement.

Dissolved Oxygen in the System. A few determinations of dissolved oxygen were made. The data was sufficient to indicate that the deaerating condenser met the performance guarantee. The average dissolved oxygen content of seven samples of condensate was 0.004 ppm (0.0028 cc/liter). The guaranteed dissolved oxygen figure was 0.05 cc/liter.

The dissolved oxygen content of the reactor water ranged from 0.28 to 0.37 ppm. Determinations of the dissolved oxygen content in the condensed steam samples showed 3.5, 3.7, and 4.4 ppm. Additional measurements should be made of dissolved gases during water decomposition runs.

Make-up Demineralizer Operation. Operation of the make-up demineralizers was started early under necessity of having demineralized water available for flushing the reactor vessel, piping, and external components in the system. The units did not meet the specified capacity output of 6000 gallons of water per regeneration and the caustic solution injection system was troublesome. The output of each unit between regenerations varied between 3500 and 5000 gallons. The pH of the effluents at the end of a run were 5.4 and 5.6 which indicated insufficient anion resin. Three cubic feet of anion resin was supplied by the vendor. Addition of $1\frac{1}{2}$ cu ft of anion resin was made to unit no. 1 and later the remaining $1\frac{1}{2}$ cu ft were added to unit no. 2. The units have been regenerated only once or twice since then and no definite capacity figures have been obtained. Each time difficulty was experienced on injecting the caustic solution.

Retention Tank Waste Disposal. The retention tanks in the reactor building have been emptied innumerable times since startup of the reactor. All of the samples taken prior to discharge have been below the maximum permissible level and all waste water has been discharged to the laboratory sewer system.

The reactor water purification system is maintaining a low level of activity in the reactor water. The activity present is not long lived and with normal decay time and dilution in the retention tanks no difficulty should be experienced with waste disposal.

Feedwater Filters. During the initial operation of EBWR, no. 2 feedwater filter was in service and because it was felt that a large quantity of insoluble matter (corrosion products) would have been flushed out of the turbine-condenser unit the filter cartridges were replaced. Unit no. 2 was reloaded fifteen days later and the amount of residue was considerably less and much lighter in color than the cartridges taken out previously. The cartridges in unit no. 1 were changed for the first time nine days after unit no. 2 was reloaded.

Unit no. 2 was changed again after about 230 hours of operation. A sample of one of the cartridges, picked from about the center, was sent to Chemistry for a quantitative spectroscopic analysis in an attempt to get some data on the amount of insoluble corrosion products being taken out of the turbine and condenser. A sample of one square inch of filter surface was analyzed for the metals expected to be present, Fe, Ni, Cr, etc.

To arrive at a value of the amount of material removed from the condensate the results from this one sample were assumed to be representative of all the cartridges. The amounts of insolubles were calculated as the metal itself and are:

Iron	34.7 grams
Nickel	15.7 grams
Chromium	3.1 grams
Manganese	0.82 grams
Molybdenum	0.31 grams

The activity level of the unloaded assembly of cartridges measured about 1.5 mr/hr and individual cartridges were below 1 mr/hr.

Conclusions. The amounts of metal pickup appear quite high and the question of having a representative sample was raised. After a long period of continuous reactor operation a group of cartridges taken from various locations in the feed water filter unit should be analyzed.

Pre-filter in Reactor Water Purification System. The short-lived half-life of the activity in the pre-filter was determined to be about three hours. This corresponds to the 2.6 hr half-life of Mn^{56} . This checks with the fact that the primary activity in the reactor water was Mn^{56} . The activity could be taken out on the pre-filter cartridges by a number of mechanisms such as:

1. Insoluble MnO_2 .
2. Complex permanganates (Fe, Ni)
3. Co-precipitated with original $Fe(OH)_2$ with subsequent oxidation.
4. Adsorbed on the filter cartridges from the soluble state.

Water Quality. The ion exchange column in unit no. 1 of the reactor water purification system was used to pick up the boric acid after the borax poison run. The boric acid capacity of the 6 cu ft of anion resin is about 27-30 pounds; since only 20 pounds were removed by unit no. 1, it was kept in service. A sample of influent and effluent was taken from unit no. 1. Analysis showed the boron content to be 1.3 and 1.4 ppm respectively, showing that at this low concentration the ion exchange bed is not effective. Since it is planned to make a full load (50 pounds) injection with the high pressure boric acid system, no. 1 ion exchange column should be replaced before the test is made.

ION EXCHANGE COLUMN NO. 1 RESIN REMOVAL

(Test Report No. 15, by V. C. Hall)

The resin bed from the no. 1 ion exchange unit was removed from the pressure vessel. The total flow through the bed had been 5,600,000 gal. Due to intermittent operation of the reactor during this period and intermittent operation of the ion exchange unit, a true history of the use of the resin column has little significance.

The reactor down time prior to removal of the resin column was 2 weeks. The activities along the side of the column were as follows:

Top of resin	490 mr/hr
1 ft down from top	125 mr/hr
2 ft down from top	35 mr/hr
Bottom of resin	20 mr/hr

These measurements were on the surface of the stainless steel resin basket.

Samples of resin were obtained for analysis. The activities measured within the basket above the resin were as follows:

Top of bed	1 r/hr at 6 in.
1 ft down from top	500 mr/hr at 2 in.
2 ft down from top	100 mr/hr at 2 in.

The activity was all hard gamma. Measurements lower in the bed were too low to obtain significant readings due to high background activity, 20 mr/hr.

The resin was removed from the column with a stainless steel scoop. The top resin was typical yellow anion for approximately 2 in. in depth. From 2 to 8 in. the darker cation resin was intermixed with the anion but in segregated pockets rather than as a good mixture. At approximately 1 ft down a white and gray sand colored mixture was dispersed through the bed. This material at times covered over half the cross section of the bed. Below $1\frac{1}{2}$ ft the anion and cation appeared to be uniformly mixed.

The samples of resin obtained for analysis were 1 ft down, 2 ft down, and 4 ft down from the top of the column.

A one-pint sample at each point will be used for radiolytic and chemical analysis. A one-gal sample at each point will be used for analysis of resin damage. The reports from these analyses will be submitted when they are completed.

WATER DECOMPOSITION AS A FUNCTION OF REACTOR POWER

(Test Report No. 22 by R. E. Bailey)

Water decomposition and the activity at the air ejector were determined at power levels of 5, 10, 15, and 20 Mw. The reactor was operating at 20 Mw with 3,000 pounds of steam per hour going to the turbine. Decomposition, oxygen in the condensate, oxygen in the feedwater, oxygen in the reactor water, and activity at the air ejector were determined. The power was then dropped to 5 Mw. The above mentioned items were then measured with time until the values became constant. One hour was required for this. This procedure was followed at 10 Mw and 15 Mw respectively.

During the 15 Mw run make-up water was added to the hotwell. There was no noticed rise in oxygen in the feedwater. The oxygen concentration in the condensate leaving the third to last stage of the turbine was found to be 12 ppb. This is in the same range as that found at full generating capacity (5 ppb).

The conductivity, oxygen in the reactor water, oxygen in the condensate, oxygen in the feedwater, and pH all remained constant throughout the experiment. They were 0.45 micromho, 0.23 ppm, 12 ppm, zero ppb and 7.0 respectively. The other data are tabulated in Table I and graphed in Figs. 22-1 and 22-2. Decomposition from Borax III at pH 7.0 has been included for comparison.

Table I - Decomposition Data

Date	Power level Mw	Pressure, psig	Steam flow, lb/hr	Decomposition,* cc gas/liter of condensate	Decomposition rate, cc O ₂ /sec	Dissolved O ₂ in condensate	Activity of the air ejector, mr/hr
2-20-57	20	600	60,000	55.1	157	3.5	-
5-19-57	20	600	60,000	40.0	148	12.0	270
5-19-57	15	600	46,000	35.8	113	12.0	200
5-19-57	10	600	30,000	32.0	69	12.0	110
5-19-57	5	600	13,000	26.1	29	12.0	40
Borax III	8	300	30,000	77.0	102	~3.5	-

*Composition of EBWR Gas - 30.7 percent O₂ 66.4 percent H₂ and 2.9 percent inerts

Composition of Borax III Gas - 32 percent O₂ 60 percent H₂ and 8 percent inerts

It should be noted that water decomposition is reported as total cc of O₂ per liter of condensed steam and/or cc of O₂ per second. This is valid since the water decomposed stoichiometrically. Dissolved gas in the condensate is then also accounted for.

The data shows that the decomposition rate (cc O₂/sec) is a quadratic function of reactor power for EBWR, and that the decomposition is a linear function of reactor power. The Borax III data point falls high, approximately a factor of two. The activity at the air ejectors does not seem to be linear with reactor power.

Fig. 22-1
Oxygen evolution rate as a
function of reactor power

Fig. 22-2
Activity level at the air ejector
as a function of steam flow

WATER DECOMPOSITION AT VARYING REACTOR POWERS AND PRESSURES

(Test Report No. 22A, by R. E. Bailey & E. W. O'Keefe)

The EBWR was operated at varying power levels, and at varying pressures at each power level. Measurements were made of the oxygen carried by the steam and the oxygen content of the reactor water. The total quantity of gas carried by the steam was measured by displacement of water. An analysis was made on the gas with an orsat. An analysis of the oxygen in the condensed steam associated with the gas was determined by the Winkler method. This gave the total oxygen evolved per liter of condensed steam. A typical gas analysis showed the following: 30 percent oxygen, 67 percent hydrogen, and 3 percent inerts. This in-balance of the oxygen and hydrogen is due to the greater solubility of the oxygen in the condensed steam over which the gas is collected. An analysis of the gas leaving the air ejector on the condenser contains a stoichiometric amount of hydrogen and oxygen.

During this test the carryover of nongaseous activity from the reactor water was studied as a function of the two variables, power and pressure. It was very difficult to obtain a sample of steam plated out on a planchet before the activity present had decayed out. During the entire run only one good sample was obtained and the main constituent identifiable was Co^{61} which has a 1.7 hr half-life. This data is presently being confirmed and when definite results are obtained a summary report will be written.

The data obtained during this testing period and others has been presented in graphic and tabular form. See Figs. 22A-1 through 22A-6 and Table I. A sample calculation has also been included and this is on the last page.

Fig. 22A-1

Net decomposition rate vs
reactor steam power

Fig. 22A-2. Decomposition rate vs pounds steam per minute

Fig. 22A-3. Decomposition rate vs reactor steam power

Fig. 22A-4. Decomposition rate vs reactor steam power

Fig. 22A-5
Decomposition rate vs
reactor steam power

Fig. 22A-6
Decomposition rate
vs reactor pressure

Table I - Water Decomposition Data

Power level Mw	Steam flow lb/min	Oxygen in condensate, ppm	Decomposition rate, cc O ₂ /min	Oxygen per lb steam, cc	Source of data
600 psig					
20	1000	3.5	9430	9.43	1
20	1000	12.0	8900	8.90	1
20	1000	10.4	11708	11.71	2
20	1000	10.4	11403	11.4	2
20	1000	10.4	11439	11.44	2
20	1000	13.0	8800	8.80	3
20	1000	11.7	8860	8.86	3
20	1000	12.4	8760	8.76	3
20	1000	12.9	9071	9.07	4
20	1000	13.0	8967	8.97	4
20	1000	13.2	9030	9.03	4
20	1000	13.3	8726	8.73	4
20	1000	13.0	8795	8.80	4
20	1000	12.0	8930	8.93	4
20	1000	11.0	9060	9.06	4
20	1000	11.0	8880	8.88	4
15	750	12.0	6780	9.05	1
15	750	12.9	7121	9.44	2
15	750	12.9	7076	9.44	2
15	750	12.9	7165	9.57	2
15	750	12.9	6902	9.23	2
15	750	12.9	6967	9.31	2
15	750	12.9	6946	9.25	2
14.6	734	12.5	6200	8.46	3
14.6	734	12.5	6410	8.75	3
14.6	734	12.5	6370	8.70	3
10	500	12.0	4150	8.3	1
10	500	13.3	4531	9.06	2
10	500	13.3	4490	8.97	2
10	500	13.3	4474	8.96	2
8.6	484	12.5	4230	8.74	3
8.6	484	13.9	4240	8.77	3
8.6	484	14.8	4310	8.91	3
5	250	12.0	1740	8.1	1
5	250	10.7	1933	7.74	2
5	250	12.2	1924	7.71	2
5	250	12.7	2115	8.46	2
5	250	13.1	2201	8.82	2
5	250	12.3	2149	8.60	2
500 psig					
19.4	954	11.7	9338	9.80	2
19.4	954	9.1	9403	9.85	2
19.4	954	12.2	10272	10.8	2
19.4	954	12.3	10263	10.8	2
15.2	750	13.1	7645	10.2	2
15.2	750	12.3	7497	10.0	2
15.2	750	11.7	7606	10.1	2
15.2	750	12.3	6957	9.85	2

Table I)Cont'd.)

Power level Mw	Steam flow lb/min	Oxygen in condensate, ppm	Decomposition rate, cc O ₂ /min	Oxygen per lb steam, cc	Source of data
500 psig					
9.6	475	11.0	4064	8.55	2
9.6	475	13.7	4709	9.93	2
9.6	475	13.4	4934	10.4	2
9.6	475	12.5	5241	11.0	2
7.1	350	13.5	3412	9.75	2
7.1	350	13.0	3325	9.58	2
7.1	350	11.4	3248	9.58	2
400 psig					
20.8	1042	10.7	12199	11.6	2
20.8	1042	10.6	12308	11.8	2
20.8	1042	12.9	12406	11.9	2
15.7	788	11.4	10694	13.5	2
15.7	788	11.9	9413	12.0	2
15.7	788	12.4	9374	11.9	2
10.6	529	10.9	5468	10.3	2
10.6	529	11.8	5911	11.2	2
10.6	529	10.4	6123	11.6	2
4.8	242	16.0	2794	11.5	2
4.8	242	16.4	2867	11.9	2
4.8	242	10.5	2677	11.1	2
300 psig					
20.3	1025	11.8	22647*	22.1	2
20.3	1025	11.8	18423*	18.0	2
20.3	1025	11.8	24948*	23.3	2
18.3	916	13.9	9750	10.6	3
18.3	916	13.8	11070	12.4	3
18.3	916	13.7	10800	11.8	3
15.4	775	11.3	11304	14.6	2
15.4	775	11.3	10612	13.7	2
12.8	642	13.9	6700	10.4	3
12.8	642	13.9	8000	12.5	3
12.8	642	13.9	7950	12.4	3
9.9	500	13.3	6806	13.6	2
9.9	500	13.3	6833	13.7	2
4.4	221	14.3	2913	13.2	2
4.4	221	14.3	2889	13.0	2
4.4	221	14.3	2622	11.9	2

*Corrected values for 20.3 Mw run

20.3	1025	11.8	14100	13.7	2
20.3	1025	11.8	12700	12.4	2
20.3	1025	11.8	14700	14.3	2

Data sources

1. Test report No. 22, dated 6/6/57
2. Data of week of 9/23/57
3. Data of 11/10/57
4. Data of week of 11/11/57

A preliminary analysis of the data indicated that the results at 20 Mw and 600 psig were in error. These were corrected. Both corrected and uncorrected points have been included on the graphs. The corrected points are indicated as circles on the 300 and 600 psig plots. As a further check, measurements of the decomposition were again made at 600 and 300 psig. During this run the purification system was not in operation. The conductivity of the water increased to approximately 0.55 micromho. There was no noticeable effect on the net decomposition.

Figures 22A-1, -3, -4, and -5 are plots of the calculated cc of oxygen produced per minute by the net decomposition of the reactor water at 600, 500, 400, and 300 psig respectively. Figure 22A-6 is a cross plot of Fig. 22A-1, -3, -4, and -5. Also included are data not previously reported. These points are circled and the date on which they were obtained has been included. It should be noted that during all of the testing periods the oxygen concentration in the reactor was held constant at 0.24 ppm. The conductivity has generally been in the range of 0.3 - 0.5 micromho.

The data indicate that the decomposition is a linear, or very nearly linear function of the reactor power up to 20 Mw (thermal). The decomposition appears to be curvilinear with pressure, with the higher decomposition occurring at lower pressures. The decomposition observed is the net decomposition. This is the difference between the forward reaction (decomposition) and the back reaction (recombination). The recombination rate is temperature dependent, increasing with increasing temperature, and hence, is a function of system pressure in a boiling water reactor while the decomposition is only a function of the energy absorbed or the power level of the reactor. Other variables here are the stripping action of the steam as it removes the gases from the reactor water and the delay time of the water in the fuel channel since this gives a chance for the decomposition products to recombine while exposed to the gamma flux from the fuel. N. A. Kershaw and A. Gibson in Test Report No. 34 have shown that the percent voids and the velocity through the core increase with power for any given pressure and also increase with decreasing pressure for any given power.

Because preliminary analysis of the data indicated that the dependence on reactor power was definitely curvilinear, Fig. 22A-2 was made from the 600 psig data by dividing the decomposition rate by the steam rate and plotting the quotient against the steam rate. This process will ordinarily yield a straight line if the original plot is quadratic, however it requires considerably more accurate data than is available here, since for curves or lines close to the 45 degree axis, an almost horizontal line results.

It should be stated that the writers have not tried to give any quantitative expressions for the decomposition of the water at EBWR due to the complexity of the problem. Rather, they are trying to give the reader the data which is peculiar to EBWR and an insight into some of the variables upon which they believe the net decomposition depends.

Calculations

Measured:

Volume of condensed steam plus water displaced
in collecting gas 3550 cc

Volume of water remaining in gas collector 171 cc

Dissolved oxygen in condensed steam. 11.7 ppm

Volume of gas collector. 326 cc

Steam flow 57,250 lb/hr

Derived:

Net condensate = $3550 - (326 - 117) = 3495$ cc

cc O₂ in condensate = $3.495 (11.7)(0.7) = 28.62$ cc
 $0.7 \text{ cc O}_2 / 1000 \text{ cc H}_2\text{O} = 1 \text{ ppm O}_2$

cc O₂ in gas collected = $(326 - 171)(0.3) = 46.5$ cc
 (gas assumed to be 30% O₂)

Total O₂ = $46.5 + 28.6 = 75.1$ cc

cc O₂/liter condensate = $75.1 / 3.495 = 21.49$ cc/liter

cc O₂ generated/min = $\frac{21.49(28.32)(0.016)(57,250)}{60} = 9338 \frac{\text{cc O}_2}{\text{min}}$

Steam power = $\frac{57,250 (1204.4 - 69.1)}{3.416 \times 10^6} = 19.4$ Mw

cc O₂/lb steam = $9338 / \frac{57,250}{60} = 9.8$ CO₂/lb steam

28.32 liters = 1 cubic foot

Specific volume of water = 0.016 ft³/lb

1204.4 = heat content of steam @ 400 psig

69.1 = heat content of water @ 2 in. Hg

Analysis of collected gas = 30-31% by volume O₂

DEPOSITS FROM A FUEL ELEMENT AND DUMMY FUEL ELEMENTS

(Test Report No. 29, by C. B. Zitek)

During the shutdown of EBWR after 1370 Mw days of operation, the reactor vessel was opened to check out the operation of the fuel element transfer coffin and install thermocouples on one of the dummy fuel elements. When one fuel element (position 129-370) and two dummy elements were transferred to the fuel storage pit for inspection samples of deposits were procured. These samples were taken primarily for analyses and to determine the type of deposit, i.e., adherent or nonadherent. The samples were obtained underwater by wiping the surfaces with new cellulose sponges and washing the deposit off the sponge. This procedure was used to determine how adherent the deposit was and to eliminate the possibility of scratching the predeposited film on the Zircaloy cladding. Three samples were taken:

1. External surface of end fuel plate of fuel element (upper portion).
2. External surface of dummy element.
3. Internal surface of dummy element.

After the dummy elements had been removed from the storage pit and stored at the burial ground additional deposit samples were taken to determine the amount of deposit on the surfaces and to get additional samples for further chemical, x-ray diffraction, and spectrographic analyses.

Type of Deposit. All three deposits were nonadherent and readily removed by a single wipe of the sponge. The deposit on the fuel plates was reddish-brown in color and the dark grey zirconium oxide film appeared intact when the deposit was wiped off. The deposit on the external surfaces of the dummy elements also was reddish-brown in color. The deposit on the internal surfaces of the dummy element was greenish-blue when observed in the reactor core; however, when the dummy was moved to the storage pit the color was much lighter, and finally when the deposit was removed it was a dirty white color. The appearance of both the external and internal surfaces of the dummy was not bright when the deposit was wiped off. This would be expected since the aluminum alloy would have formed an oxide layer before the crud or corrosion product layer had been deposited.

The deposits were very soft and flocculent in texture and when washed off the sponge appeared to be very finely divided. An attempt was made to determine the particle size of the fuel element deposit but by the time the sample was photographed using an electron microscope, the particles had agglomerated. Later a sample of deposit from the external

surface of the dummy element was prepared and an electron microscope photograph taken at 3100 X. Most of the particles appeared to be less than 1.5 microns in diameter. This deposit should be similar to that on the fuel element.

The physical characteristics of the deposit on the dummy elements changed after drying. When additional samples were taken from the dummy elements later the deposit was very adherent and had to be scraped off. Two external surfaces had a dark reddish-brown color and the other two sides had a mottled appearance with light colored dots which probably were the aluminum alloy corrosion products.

The amount of deposit on the fuel element could not be determined; however, that on the external and internal surfaces of the dummy element was determined. The amounts in milligrams per square inch are given in Table I. These values were determined by scraping off the deposit and are approximate figures. It is understood that the Metallurgy Division is determining the amount by a stripping technique and the values included in this memo should be checked against the stripped values when available.

Table I - Spectrographic Analyses of Deposits from
EBWR Fuel Element and Dummy Elements*

Constituents (as metal)	Deposit from fuel plate of fuel element, %	Deposit from ex- ternal surface of dummy fuel element, %	Deposit from in- ternal surface of dummy fuel element, %
Iron	17.4	35	3
Nickel	13.9	13	16
Chromium	0.1	0.3	0.05
Aluminum	5.8	10	20
Copper	0.3	0.4	-
Lead	0.2	0.3	0.1
Boron	0.2	0.5	0.4
Manganese	0.2	0.3	0.04
Sodium	**	1	0.5
Potassium	**	0.1	0.1
Calcium	**	0.08	0.05
Magnesium	**	0.1	0.02
Zirconium	None	None	None
Amount of deposit	***	1.1 mg/in. ² 1.4 mg/in. ²	2.2 mg/in. ²
Color of deposit	Reddish-brown	Reddish-brown	Dirty white (wet) Buff (dry) Greenish-blue (viewed through reactor water)

*This table gives major components only - trace constituents not included.

**Sample contaminated by water from the fuel storage pit, determined values not valid.

***Sampled underwater - could not get amount of deposit per square inch.

Note: The individual metal component values do not total 100 as these values are of the metals only. The anions are not included in this analysis. Practically all of the metals were present as oxides and some of the oxides may have water of crystallization present also.

Spectrographic Analyses of Deposits. The analyses of the three deposits are given in Table I. Both the fuel element deposit and the deposit from the external surface of the dummy fuel element consisted primarily of iron, nickel, and aluminum oxides. The iron content of the fuel element deposit appears to be low and probably should be about the same as the external deposit from the dummy fuel element. The aluminum content of the deposit from the external surface of the dummy is higher as would be expected. The nickel content is considerably higher than would be expected and is due to two factors:

1. In the corrosion of stainless steel the nickel content is always much higher than the chromium content due to a preferential leaching out of the nickel.
2. The nickel pickup in the steam piping and turbine by the condensate must be quite high.

The fact that the chromium contents are as low as they are indicates that most of the nickel in the deposit was brought in with the condensate.

The deposit from the internal surface of the dummy is quite different. The low iron content would be expected since the water in the dummies is stagnant and the deposit is primarily due to the corrosion products of the aluminum alloy M-388. The high nickel content of this deposit is of significance in that the ratio of aluminum to nickel in the deposit is not the same as in the alloy (1% Ni - 99% Al). This is similar to the corrosion products from stainless steel where the nickel content is higher than the chromium content.

Radiochemical Analyses of Deposits. Radiochemical analyses of the deposits are in the process of being made by the Chemistry Division and will be published when available. Only the long lived activities will be determined.

A gamma-ray analysis was made on the deposit from the internal surface of the dummy element by the Industrial Hygiene and Safety Division. The gamma activity present was due to Co^{58} and Co^{60} which are products of a (n-p) reaction on Ni^{58} and Ni^{60} respectively. The measured activity was 91×10^{-6} curies per gram of Co^{58} and 3.5×10^{-6} curies per gram of Co^{60} .

General Comments. The amount of deposit present on the fuel elements was more than had been expected since the quality of the reactor water has been maintained at less than one micromho conductivity during most of the reactor's operation. There is the possibility that most of the crud was deposited during a twelve day period of operation when the reactor water purification system was not being used. Boric acid was added to the water to simulate full poisoned conditions and observe neutron stability using the full core height. At the time it was planned to operate only one

or two days without the cleanup system. However, the boric acid was left in for twelve days and during this time the quality of the reactor water was poor, 26 micromhos conductivity. This may not be the cause of the heavy deposit because work has been done which showed that a deposit formed under poor water conditions was later redissolved when the water quality was improved.

When the reactor vessel is opened again the amount of deposit on one of the two dummy elements put in the reactor core should be determined and compared with that on one of the originally loaded dummies.

There has been much discussion about having an accumulation of crud in the bottom of the reactor vessel. During the shutdown when the reactor vessel was open an attempt was made to determine whether there was a layer of loose deposit at the bottom. A 28 ft long probe was put down through one of the outer dummy holes and water drawn up with a vacuum system. No deposit was drawn up. It is possible that the $\frac{1}{4}$ in. stainless steel probe, which was quite flexible, did not pass some of the plates located below the grid support plate and therefore was not at the bottom of the vessel.

SCALE ON FUEL ELEMENTS IN EBWR

(Test Report No. 29A, by R. H. Leyse)

This report discusses the scale presently on the fuel elements in EBWR. Our latest information on this scale is from fuel element ET-51, which has been dissected by the Metallurgy Division. This fuel element was in the reactor from startup until April, 1959. This was an enriched thin fuel element and had a maximum heat flux of 130,000 Btu/(hr)(ft²) for 20 megawatt operation.

Fuel Element Scale. The thickness of the scale from the general zone of high heat flux is 0.0046 in. This thickness has been established from photomicrographs of the scale in cross section. The thickness of the scale as measured with a micrometer is 0.005 in. (several measurements) and because of surface roughness these values are larger than established from photomicrographs.

The density of this scale is 2.5 grams/cm³. This was established from the weight and volume of a piece of scale.

The relative proportions of metal atoms in this scale have been approximately determined by spectrochemical analysis with a factor of two accuracy. Results are as follows:

Al	very strong
Fe	8 percent
Ni	14 percent
Others	1 percent

The percentages refer to amount present relative to total metal content and are not true weight percentages.

The chemical compound form of the metal atoms in the scale has been partially established by x-ray diffraction. Results show the presence of boehmite, and a second unidentified phase. The aluminum content of fuel element scale has never been determined with precision. We have provided the Chemistry Division with a specimen large enough for gravimetric aluminum determination.

Adherence of Fuel Element Scale. The scale is extremely adherent to the surface of the fuel element. Some of it tends to flake off of the elements during reactor operation. These statements appear inconsistent, however, each is true. All surfaces from fuel element ET-51 were photographed and these photographs reveal that some scale is flaked off from the surfaces in the course of reactor operation. The scale tends to flake off from the regions of higher heat flux, but all of the scale in these regions does not flake off. There is a tendency for the scale to flake off near the center of the fuel element and remain at the edges.

In general, it was found that the majority of scale which was on the as received fuel element was strongly adherent. Wire brushing removed only a small amount of the scale originally present. The exact amount of scale removed by wire brushing was not determined. This wire brushing was in preparation for ultrasonic testing of the fuel plates for bond integrity. The ultrasonic testing for fuel plate integrity showed no defects present. This testing effectively showed that the scale present was well bonded to the plate.

The fuel plates were cut into sections after the ultrasonic testing was completed. The cutting operation caused severe vibration of the plate. In most cases this did not loosen the scale, but on two occasions some flaked loose. This is the source of the scale which has been tested. Attempts were made to loosen scale from sections of these plates by severe hammering with no success. One section had a clearly defined area from which scale had previously flaked off, but severe hammering would not cause further release of scale. Photographs of plate no. 825 show that a large amount of flaking of scale occurred on one side of the plate, with very little flaking on the opposite side.

The adherence of fuel element scale may be summarized as follows: Scale with thickness of nearly 0.005 in. is strongly adherent to the fuel elements. This scale is present in the region of maximum heat flux and also in other areas. This scale may flake off from regions of higher heat flux, but flake off is never complete and seems to range from 20 to 60 percent of certain regions.

Distribution of Fuel Element Scale. The thickness measurements reported (0.005 in.) are from regions of high heat flux. No measurements have been made of this thickness as a function of fuel element length.

The scale deposits along the heat transfer surface and then tends to flake off from regions of high heat flux. Photographs of the plates after cutting from the subassembly, but prior to other work, indicate the following conditions:

1. Scale deposit is small on the picture frame. There is a distinct scale boundary along the edges at the picture frame.
2. For fuel plates 821 and 829 there is little flake-off on either side.
3. For plates 825 and 813 there is little flake-off from one side of the plate with flake-off on the other side. For plate 813 this flake-off appears to be almost complete from the region of high heat flux with evidence for the beginning of redeposit near the edges.
4. Plate 803 has flake-off from each side, however, the flaked-off regions for one face roughly correspond with scaled regions of the opposite face.
5. Plate 828 has flake-off from each side. Flake-off is quite complete for the high flux zone of one side and is about 50 percent for the corresponding area of the opposite side.

Fuel Element Environment. The presence of scale means that we must consider the resulting new environment of the fuel plate clad. When the reactor was first operated, the Zircaloy-2 clad was in boiling water. However, with scale present the clad is in contact with superheated steam because boiling occurs at the surface of the scale and steam permeates the scale. The scale is probably at least 25 percent porous. With the reactor at power these pores become filled with superheated steam.

Summary. Scale about 0.005 in. thick is present on the EBWR fuel elements. This scale tends to flake off in regions of high heat flux, but flake-off is not complete in these regions. Some of the plate faces examined showed little flake-off of scale, while others had varying degrees of flake-off.

TEST FOR SCALE REMOVAL FROM FUEL ASSEMBLY H-18

(Test Report No. 29B, by I. Charak and R. H. Leyse)

Introduction. The problem of scale deposition on the EBWR fuel elements has been previously reported.¹ Since a large percentage of the scale is boehmite, it has been proposed² that removal of the scale could be effected by heating the entire assembly to 850°F, thereby driving off the water of hydration and causing the scale to curl and flake off. This test report is concerned with the heating of a complete fuel assembly to 850°F in an oven located in the cave facilities of Building 301. The oven is described at the end of this report.

Test Procedure. Measurements of the overall dimensions of the assembly were made before and after heating to determine the amount of growth, if any. The fuel assembly was then slowly heated to 850°F, held at this temperature for 25 hours and then slowly cooled. Throughout the heating and cooling the assembly was blanketed with argon gas. Photographs of the fuel and side plates were taken before and after the test and are shown in Figs. 29B-1 through 29B-7.

Results. The most significant result of this test was that the $\frac{1}{16}$ in. thick Zircaloy-2 side plates buckled along their lengths between the spot welds on the fuel plates as shown in Figs. 29B-6 and 29B-7. The buckling amounted to at least $\frac{1}{8}$ in. on each side. This result, though undesirable, at least proves that this method of descaling, even if successful in removing the scale, could not be employed. The resulting side plate deformation would preclude the fuel assembly's use in the reactor due to insufficient clearances.

One theory concerning the cause of the plate buckling considers that the fuel plate has a thermal coefficient of expansion greater than that of Zircaloy-2. As the fuel plates heat up, they stress the side plates beyond the yield strength. Upon cooling, the greater fuel plate shrinkage causes the side plates to buckle outward from the spot welds. Since the cladding is in compression at room temperature and since the predicted radiation growth has not materialized, it is quite possible that the cladding never becomes stressed beyond the yield strength.

The thin scale coat which was present on fuel assembly H-18 did not flake off. This can be seen by examining Figs. 29B-1 through 29B-7. Marks on several of the plates are due to scratching.

¹Test Report no. 29A.

²R. H. Leyse, Private communication.

Fig. 29B-1. Fuel plate "A" before and after heating for 25 hr at 850°F

Fig. 29B-2. Fuel plate "C" before and after heating for 25 hr at 850°F

Lower half before heating

Lower half after heating

Upper half before heating

Upper half after heating

Fig. 29B-3. Side plate "B" before and after heating

Lower half before heating

Lower half after heating

Upper half before heating

Upper half after heating

Fig. 29B-4. Side plate "D" before and after heating

Fig. 29B-5. End view of fuel assembly before and after heating

Fig. 29B-6. Fuel assembly in oven after heating, showing side plate bowing

Fig. 29B-7. Closeup of side plate, showing bowing

Since the assembly was in a region of low neutron flux, there was less than 0.002 in. of scale present and it is possible that scale removal is a function of scale thickness. The reason for this belief is that coupons from fuel plates of assembly ET-51 with an appreciable thickness of scale have been successfully descaled using this method.¹

Finally, measurements of the assembly were made after cooling. Considering measurement errors, there was no indication of fuel plate deformation caused by this heating. The following table summarizes the measurement data:

Table I

Position*	Distance across fuel plates (inches)	
	Before heating	After heating
1	3.526	3.523
2	3.524	-
3	3.522	3.523
4	3.526	3.512
5	3.533	3.516
6	3.519	3.518

*Positions along the fuel assembly are arbitrary.

Oven Design. The oven used in this test was designed with two problems in mind. First, due to the space limitation in the cave facility the fuel assembly could not be inserted in the oven through an open end. It was necessary, therefore, to build the oven in two sections so that the fuel assembly could be lowered lengthwise into one section, the second section being lowered in turn onto the bottom section.

The second problem to be resolved concerned the need for an inert atmosphere around the fuel assembly to prevent the zirconium reaction with air. While a positive seal would have been desirable in this connection, it was felt that the importance of immediate results precluded spending a great deal of time on such a development. In any event, even with a poor seal, there should be no in-leakage of air. Here, the only disadvantage of a nonpositive seal is that the inert purge gas cannot be collected and analyzed for radioactivity and moisture content.

The seal finally decided upon consisted of a tongue and groove type arrangement where the groove was filled with sand. The groove ran around the periphery of the bottom section of the oven and the tongue on the top section. There was enough clearance all around the groove to allow for thermal expansion.

¹R. H. Leyse, op cit.

The lower section of the oven consisted of a carbon steel box lined with firebrick, with a long, rectangular channel in the middle. A stainless steel liner (Fig. 29B-8) was inserted in this channel after flat plate resistance heaters had been placed along the sides and bottom of the channel. This sheet metal liner was integral with the seal groove mentioned above. The liner had three argon inlets along one side and three thermocouples welded along the opposite side.

The upper section of the oven consisted of an inverted stainless steel pan, the edges of which formed the tongue of the seal. A row of heaters was placed along the pan parallel to those of the bottom section, and then covered with firebrick. Figure 29B-9 illustrates the entire oven after assembly in the cave.

Temperature Control. The twenty heaters used in this test were divided into three banks, each bank controlled manually by a variac. Each thermocouple was associated with temperature measurement at each of these banks. The temperatures were recorded on a Brown recorder and manual adjustments made with the variacs to keep the temperatures close to 850°F. By this means the heat generation was kept uniform.

Inert Purge. Argon gas was used to blanket the fuel assembly. The purge gas was metered at about one liter per minute, and then flowed to each of the three inlets via a stainless tube header.

Fig. 29B-8. Lower section of oven showing stainless steel liner

Fig. 29B-9. Assembled oven prior to heating

POST-IRRADIATION HEATING OF FUEL ELEMENT H-18

(Test Report No. 29C, by R. H. Leyse)

It has been reported that post-irradiation heating of sections cut from EBWR fuel element ET-51 caused swelling of the uranium alloy.¹ For this reason, an entire fuel element was heated at 600°C to establish the gross effects of this swelling on fuel element integrity. In addition, two individual plates were heated at 700°C.

Fuel Element Heating at 600°C. EBWR fuel element H-18 was heated to 600°C and held at this temperature for 100 hours. The procedure and equipment have been previously described.

Prior to heating at 600°C, this fuel element was heated at 450°C for one day. This treatment caused no changes in fuel plate dimensions, but caused the side plates to buckle outwardly about $\frac{1}{8}$ in. between spot welds.²

Heating at 600°C caused further buckling of the side plates (about $\frac{1}{2}$ in.) as indicated in Fig. 29C-1. This heating also caused formation of large amounts of white oxide of zirconium which tended to flake off the assembly.

Fig 29C-1 Buckling of side plates caused by heating to 600°C

¹Test Report no 29-B

²Private communication, C. Reinke.

The appearance of the fuel element prior to removal from the furnace is given in Fig. 29C-2. This photograph shows peeling of the zirconium oxide along the picture frame. The appearance of the furnace after removal of the fuel element is shown in Fig. 29C-3.

Fig. 29C-2. Scale formation at 600°C

In general, the white oxide of zirconium tended to form beneath the black oxide of zirconium originally present. This is illustrated in Fig. 29C-4. The upper photograph shows white oxide on a side plate. This area was then mechanically abraded causing removal of black scale and revealing white scale beneath. This is shown in the lower photograph.

Fig. 29C-3. Scale in furnace after removal of fuel element

The large amount of oxide scale was formed because air leaked in through the sand seal of the furnace. This scale was removed from the furnace and the fuel element with a vacuum cleaner and thus gross contamination of the hot cell was avoided.

The lengths of the two outside fuel plates were measured after this heating. The original length of each fuel plate was 54 in. The final length of one plate was $54\frac{3}{32}$ in.; the final length of the opposite plate was 54 in. This uneven growth caused slight bowing of the fuel element.

Fig. 29C-4. Side plate before and after mechanical abrading

The fuel plates were removed from the element using a simple, but effective, cutting device as illustrated in Fig. 29C-5. Visual examination of the plates gave no indication of clad failure or bulging. The thickness of fuel plate 558 was measured at about twenty locations after the 600°C heating. These measurements ranged in value from 0.264 to 0.288 in. Nominal plate thickness is 0.280 in. and departures from this thickness measured on plate 558 were caused by oxide scale formation and flake off.

Heating of Fuel Plates at 700°C. Fuel plate 558 from assembly H-18 was heated at 700°C for twenty hours. Final length after heating was $54\frac{3}{8}$ in. indicating an increase of $\frac{1}{4}$ in. A large amount of zirconium oxide scale was formed. After the loose scale was removed, the plate had an interesting frost-like appearance (Fig. 29C-6). Visual examination of this fuel plate revealed no clad failure or bulging.

Fuel plates 558 and 528 were then heated together at 700°C for about three hours. This caused no change in length for these plates (original and final length of plate 528 was $54\frac{3}{32}$ in.).

Fig. 29C-5. Method of removing plates from fuel element

Fig. 29C-6. Frosted appearance of fuel plate 558 after 20 hours at 700°C

The two fuel plates were then heated at 700°C for forty-six additional hours. This heating caused plate 528 to grow about $\frac{1}{8}$ in. in length to a total of $54\frac{1}{4}$ in. Thickness measurements of plate 528 ranged from 0.263 in. to 0.285 in. and these variations are probably a result of oxide scale formation. There was no indication of clad failure or swelling for this plate.

Fig. 29C-7. Swelling of fuel plate 558

This furnace run, however, caused spectacular failure of plate 558. There the clad failed and the fuel alloy swelled and deformed at two regions of this plate. Final length of plate 558 was about $55\frac{1}{4}$ in., an increase of one inch over the previous heating. The failure of plate 558 is shown in Figs. 29C-7, -8, and -9.

Plates 558 and 528 had adjacent positions in fuel element H-18 and therefore had about the same irradiation history. Fuel plate 558 had about eighteen hours more of heating at 700°C than fuel plate 528; aside from this factor each plate had the same furnace history. Figure 29C-10 shows both of these fuel plates after the final heating at 700°C . The greater length of plate 558 is obvious.

Significance of Results. There was very little growth of the fuel plates at 600°C . There was little growth and no clad failure of plate 528 at 700°C . With the clad failure of plate 558, there was substantial growth and distortion at 700°C . It is evident that swelling and distortion of fuel plates at high temperatures is limited by a sound clad. Lack of this sound clad allows extreme fuel plate growth and distortion.

On this basis, it is argued that the only heating tests that are applicable to fuel plate stability are those conducted on integral fuel plates.

Fig. 29C-8. Swelling and clad failure of fuel plate 558 (side view)

Data collected for swelling of coupons cut from fuel elements indicate greater swelling for given conditions than if an entire plate were heated. This is because the restraining influence of the clad is greatly reduced at the ends of the coupons where the uranium is unrestrained.

The maximum burnup of plates 528 and 558 was only about 0.1 atom percent. It would be of interest to conduct similar tests on plates with greater burnup. Such tests should be conducted in an inert atmosphere without possibility of oxygen contamination.

Fig. 29C-9. Swelling and clad failure of fuel plate 558 (lower view)

Fig. 29C-10. Plates 528 and 558 after heating at 700°C

VISUAL INSPECTION OF FUEL ASSEMBLIES DURING CORE LOADING

(Test Report No. 29D, by I. Charak)

During the reloading of the EBWR core from February 29 to March 3, 1960 each fuel subassembly was visually inspected as it was raised from the fuel storage pit into the transfer coffin. The purpose of this inspection was to determine the degree of flaking off of the scale coating on the heat transfer surfaces. While any immediate use of the data obtained was not anticipated, it was felt that this inspection would perhaps be useful at some later date, especially during high power operation when fuel growth at elevated temperatures is anticipated. The fuel loading diagram is shown in Fig 29D-1.

Procedure. The removal of the assemblies from the storage pit involved a series of operations which did not afford much time for the inspection of the fuel plates. As the assembly was raised in the pool, the appearance of the surface of each of the outside fuel plates was noted and an estimate was made of the scale thickness. However, as the element approached the coffin, the activity at the surface was too high to permit continued inspection. Furthermore, due to the refraction of light in the water, the assembly appeared foreshortened. These factors tend to make the accumulated data more qualitative than quantitative, but nevertheless of interest.

Results. The inspection showed that, in general, most of the descaling had occurred on the lower half of the fuel plates. Since the heat generation rate is higher in this area, descaling could have been accomplished by increased scale-Zircaloy interface temperatures. If this is indeed the mechanism of descaling which occurs in the reactor, then the formation of scale is a self-limiting phenomenon and one need not worry about achieving high fuel alloy temperatures with resultant grain growth under irradiation. However, the inspection of the EBWR fuel elements shows that virtually all of the scale deposit is still intact on the assemblies. The fact that there is at least some descaling on most of the assemblies is not sufficient cause to be optimistic about further descaling in the reactor.

It was noted that the scale thicknesses seem to become greater near the center of the core. This bears out the theory that scale formation is a direct function of burnup and consequently thermal neutron flux. Conversely, it can be seen that fuel assemblies H-11 and H-13 have little or no scale formation. Upon examining the history of these two assemblies, it was found that they had each had only about three months irradiation, hence further substantiation of the scale thickness - burnup relationship.

EBWR CORE FUEL STATUS BOARD

CIRCLED LETTERS & NUMBERS
INDICATE FUEL ASSEMBLY LOCATION
FOR INDEXING COFFIN AND CARRIAGE
(SEE DWG. RE-1-18120-D).

H	8
T	1
EH	50
ET	55
SD	114
	148

0.53	2
SD	
4	3

CORROSION
SAMPLE
INSERT
LOCATION

FIRST NUMBER INDICATES SETTING FOR
INDEXING SMALL ROTATING PLUG.
SECOND NUMBER INDICATES SETTING
FOR INDEXING LARGE ROTATING PLUG.
(TYP. ALL FUEL ASSEMBLIES FOR
INSERTION OR REMOVAL THRU 6" DIA.
PLUG HOLE (SEE DWG. RE-1-18137-E).

* INDEXING NUMBERS FOR (9) CONTROL
RODS INDICATE SAME PLUG
SETTINGS AS FUEL ASSEMBLIES
EXCEPT CONTROL RODS ARE INSERTED
OR REMOVED THRU 11" DIA. PLUG
HOLE (SEE DWG. RE-1-18137-E).

EXPERIMENT DEVICE CODE

* CORROSION SAMPLE INSERT

PREFIX LETTER CODE

H NATURAL "U" THICK PLATES
T NATURAL "U" THIN PLATES
EH ENRICHED "U" THICK PLATES
ET ENRICHED "U" THIN PLATES

SDS DUMMY W/SOURCE

SD DUMMY STAINLESS STEEL

DATE: MARCH 1, 1960

LOAD NO.: 55

Fig. 29D-1. Fuel loading diagram

DEPOSITS ON THE STEAM SIDE OF THE CONDENSER AND STEAM DRYER

(Test Report No. 30, by C. B. Zitek)

During a shutdown of EBWR the steam side of both the condenser and steam dryer were opened for inspection. The reactor had generated 1370 Mw days of heat power (about 68 days at 20 Mw), about half of which was during a practically continuous run.

Steam Side of Condenser. The hotwell was drained of condensate and both the lower and upper manholes were opened. There were three major points to be investigated:

1. Radioactivity level.
2. Erosion of condenser tubes.
3. Deposits, appearance, amount and analyses.

1. Radioactivity level:

Due to the fact that the condenser was not opened immediately after shutdown, the activity which was measured was due to the longer lived radioisotopes. The hotwell area, lower section, was one to two mr per hr while the upper portion, above the condenser tube banks, was only 0.1 to 0.2 mr per hr. This difference was expected because the activity which came over as droplet carryover would have time to be absorbed or deposited on the hotwell surfaces. The upper portion is subjected to steam flow and a washing action so that only that portion which came over in an insoluble form and was deposited on the condenser tubes would contribute to the activity level.

The measured low activity should not be misunderstood. Most of the activity (about 99%) in the reactor water has a half-life of less than a day so that all of the major activity, due to droplet carryover, had decayed before the condenser was opened. When the condenser is again opened for inspection it should be done as soon after shutdown as possible so that the short lived activity level can be obtained.

Samples of the deposits, from the condenser tube shields, were sent to the Chemical Division for radiochemical analyses. The results are not available at this time. An addendum to this memo will follow when the analyses are finished. The radiation survey with a Juno showed that the major portion of the activity was gamma activity. Two samples were sent to the Industrial Hygiene and Safety Division. Gamma ray analyses were made to identify the activity and determine the amount. The only measurable activity present about 56 days after shutdown was Co^{58} (72 day half-life). A quantitative figure of the amount of activity was 0.088 and 0.054×10^{-6} curies per gram of deposit ($\pm 20\%$). At the time of shutdown this would have been 0.15 and 0.09×10^{-6} curies per gram.

2. Erosion of condenser tubes:

The lower condenser tubes in the vicinity of the desuperheater outlet distributor were examined for signs of erosion due to steam impingement. There were no indications of erosion of the aluminum tubes. If the erosion was uniform an external viewing would not be a suitable means of determining whether erosion was taking place. However, there was a very thin film, dark brown to black in color, on the surface of the tubes. This deposit was very adherent and no sample could be obtained. It is thought that if erosion was taking place the surfaces would be clean, bright, and free of deposit. As an additional check on erosion in this area it may be advisable to have the wall thickness determined by a Probalog test.

The upper row of condenser tubes in the steam inlet area are protected from erosion by stainless steel snap on shields. These shields were covered with a thin layer of deposit which would indicate that there had been no erosion attack by the steam flow in this area.

The exhaust end of the turbine casing and the last row of turbine blades also were inspected at this time. The interior surface of the turbine casing, which is Kanigen plated, was free of deposits and the plating appeared to be in good condition. The last row of blading was clean and the trailing edges of the blades showed no sign of erosion.

Due to the relatively short period of operation one would not expect to see signs of blade erosion unless the corrosion-erosion effect of the high oxygen steam was very bad.

3. Deposits

As previously mentioned, the condenser tube deposits in the lowest section of the tube banks, just above the hotwell, varied in color from a dark reddish brown to black. The layer was very thin and adherent and appeared to be a discoloration film rather than a deposit. An attempt was made to obtain a sample for analysis by scraping with a knife blade but no sample was procured.

An appreciable amount of deposit was present on the top row of condenser tubes. The fact that the stainless steel shields were readily removable enabled determination of the amount of deposit. Four sections were removed at first for examination. Later eight more were removed after receiving a dozen new sections from Allis-Chalmers for replacement. Figure 30-1 shows the location of the new replacement shields. The deposit in sections A and D was light brown in color and was dull in appearance, a powdery type of deposit. It was not a scale type of deposit and could be removed quite readily by scraping or by rubbing with a cloth. The

deposit in sections B and C was different both in color and appearance. Most of the shield area in these sections had a burnished copper-brass colored appearance. The surface was not dull and the deposit was more of the scale type. It was very adherent but could be scraped off.

Fig. 30-1. Location and amount of deposit on EBWR condenser tubes (July, 1957).

The underneath surfaces of two shields had a number of white spots along the edges which appeared to be aluminum oxide. This was the area in contact with the aluminum alloy tube and indicated that an electrolytic cell type of corrosion was initiated at these points. The surfaces of the condenser tubes were examined but no pitting could be detected. This condition was called to the attention of the vendors but was disregarded as being of no importance. An increase in this type of corrosion should be watched for on subsequent shutdowns.

A tabulation of the data taken on the deposit from 10 of the shields is given in Table I. There was no good correlation of the amount of deposit (in milligrams per square inch) according to location or color. The weight of deposit on individual shields was quite close (except for one high and one low value) and averaged about 0.9 milligrams per sq in.

Table I - Deposits from Steam Side of EBWR Condenser
(Scraped off of Stainless Steel Shields)

Sample number	Location in condenser*			Color of Deposit	Amount of deposit		
	Section	Bank**	Row†		Area sampled, in. ²	Amount, grams	Concentration, ‡ mg/in. ²
1	A	1	2	Burnished brass appearance	21	.0154	0.74
2	A	2	5	Dull light red-brown	11.3	.0093	0.82
3	A	3	6	Dull light red-brown	22	.0199	0.91
4	A	4	4	Half dull brown Half burnished brass	10	.0044	0.44
5	B	1	5	Burnished brass	20	.0230	1.15
6	B	2	8	Burnished brass	11.3	.0083	0.73
7	B	4	4	Dull light red-brown	16.5	.0127	0.77
8	C	2	6	Dull light red-brown	22	.0166	0.76
9	D	1	2	Dull light red-brown	18	.0182	1.01
10	D	3	6	Burnished brass	12	.0192	1.6
							Average 0.89

*All shields from top row of tubes.

**Counted from north side of condenser.

†Counted from south side of each bank.

‡Deposited during 68 days of operation at 20 Mw.

The amount of deposit appears to be high for the length of service of the unit. This value could be due to two factors:

1. Initial flushing of steam piping and turbine with plant steam which was known to contain insoluble iron oxide.
2. Operation of EBWR for about 11 days in January, 1957, when the reactor water was of poor quality and contained a high concentration of soluble metallic oxides.

Spectrographic Analyses. A tabulation of the spectrographic analyses of four individual samples is given in Table II. The major constituent is iron with small amounts of aluminum and nickel. The fact that the ratio of iron to nickel or aluminum is apparently so much different than in the deposits on the fuel plate and external surface of the dummy element points toward the possibility that the greater portion of the condenser tube deposit did not come over in the steam from the reactor vessel.

Table II - Spectrographic Analyses of Deposits from Steam Side of
EBWR Condenser and Steam Dryer - Emergency Cooler

Constituent (Reported as metal. Present as oxide)	Condenser tube shield samples				Steam-dryer sample
	Dull light red-brown color		Burnished brass appearance		Dark red-brown
	Single shield, %	Composite of four shields, %	Single shield, %	Composite of four shields, %	Second row of tubes (from top), %
Iron	VS	VS	VS	VS	VS
Aluminum	1.5	3	2	2	3
Nickel	1.5	3	3	4	1
Chromium	0.02	0.15	0.02	0.2	1.5
Copper	0.1	0.3	0.1	0.2	0.1
Lead	1.5	1.5	2	3	0.1
Sodium	0.08	0.7	0.08	0.7	0.2
Potassium	0.05	0.15	0.03	0.2	0.04
Calcium	0.08	0.4	0.02	0.3	0.04
Boron	0.1L	0.2	0.1L	0.2	0.1L

VS = very strong - major constituent

L = less than

The presence of lead in these deposits is quite disconcerting since there is supposed to be no lead in the system. The fact that all four samples have about the same amounts and that the analyses were made two at a time at different times indicates that there is no error in the determination.

Steam Side of Steam Dryer. The radiation level in the steam-dryer was between 5 and 6 mr per hr and was primarily gamma activity. The walls of the vessel were in general free of deposit. Small areas did have a brownish appearance but no deposit sample could be obtained.

The emergency cooler coil in the lower half of the vessel did have some loosely adherent deposit, dark red-brown in color. This deposit in general was located on the upper surface of the tubes and a sample was obtained for spectrographic and radiochemical analyses.

The spectrographic analysis is given in Table II and is similar to the condenser tube deposits except for the higher chromium content, lower nickel content and a lower lead content.

REACTOR STEAM PURITY BY RADIOACTIVE SODIUM TRACER

(Test Report No. 37, by J. Barrett and H. Cadoff)

A sodium tracer was used to determine the decontamination factor, which is analogous to steam purity. This is an accurate, inexpensive, and comparatively safe method of indicating carryover.

Discussion. Problems of direct-cycle boiling reactors approximate those of conventional steam plants with respect to steam quality and carryover. The problems of boiling reactors however are magnified by the fact that the carryover could be extremely radioactive and cause plant maintenance problems. These tests were run to check the performance of the EBWR boiler and steam dryer.

The ability of a nuclear reactor to produce radioactive tracer materials from common elements was utilized in this experiment. The radioactive sodium-24 produced in the reactor from natural sodium-23 allowed work with minute amounts of contaminant without fear of analysis difficulties. Contamination of sodium samples by handling, which is significant in the usual sodium analysis, was insignificant because normal body sodium is sodium-23 and is not measured in the analysis. Handling steps were minimized to lower chances of gross sample contamination by other samples.

Two types of samples were counted. One sample group was plated out, by evaporation technique on standard aluminum dishes, and counted. The other sample group was counted directly from the liquid samples contained in polyethylene bottles using equal source volumes and sample geometry. Both methods proved reproducible and counting of the samples in the sample bottles was used for all further work.

The analysis of results was carried out on a 200 channel γ - spectrometer scanning energy levels from 0 to 2.0 Mev. The 1.37 sodium peak was utilized in the analysis and was not near enough to any other normally occurring peaks to affect the validity of the results. Another check of validity of analysis was a recheck of the apparent decontamination factor of samples at different decay times. The decontamination factors obtained proved reproducible showing that the peak used was truly a manifestation of one substance. This experiment includes data taken at 20, 30 and 40 megawatts of thermal power.

Experimental Procedure. Thirty-six g of sodium sulphate (Na_2SO_4) was added to the reactor. This is equivalent to one part per million of sodium in the contents of the reactor vessel. The reactor was operating at 40 megawatts. The reactor demineralizer was taken out of service and the plant was allowed to run three and one-half hours to activate the sodium

before samples were taken. The radioactivity levels of various points in the plant were recorded periodically to check the effect of sodium addition to the reactor (see Table I).

Table I - Radiation Data, Na²⁴ Tracer

Station	20 Mw w/o	20 Mw with Na ²⁴	30 Mw with Na ²⁴	40 Mw w/o	40 Mw with Na ²⁴
1 Turbine casing	.006	.007	.014	.013	.023
2 Plant, main floor	-	-	-	-	-
3 Steam dryer	.17	.2	.35	.5	.56
4 Air ejector	.4	.36	1.3	2	2.1
5 Condenser hotwell	.03	.004	.008	.013	.015
6 F.W. filters	.003	.010	.014	.039	.041
7 Control rod room	.018	.06	.007	.006	.007
8 I.X. Column #1	4.0	5.7	5.7	8.4	6.0
9 I.X. #1 after filter	2.2	2.5	2.5	2.5	2.5
10 I.X. column #2	.8	.5	.5	.6	.5
11 I.X. column #1 pre-filter	1.1	2.9	2.9	3.8	3.0
12 Plant air exhaust duct	.008	.011	.026	.028	.039
13 Startup heater	.47	.61	.64	.077	.67
14 Effluent, retention tank	.005	.018	.019	.014	.018
15. Cooling water, 2nd cooler	.006	.009	.010	.01	.011
16 Air ejector tank- fission monitor	-	.10	.11	.24	.13

Sodium added at 1025 and 40 Mw on 2/12/58.

40 Mw readings taken at 1525 - changed to 30 Mw at 1540.

30 Mw readings taken at 1620 - changed to 20 Mw at 1633.

20 Mw readings at 1650.

Samples of reactor water, condensed steam before the steam dryer, and condensed steam after the steam dryer were taken at three power levels of the reactor 20, 30, and 40 Mw thermal. The data were interpreted as de-contamination factors and apparent steam qualities (see Table II).

Table II - Data for Sodium Tracer

Date taken	Date counted	Reactor power	Reactor water	Activities, cpm		DF		
				Before steam dryer	After steam dryer	Reactor BSD	Reactor ASD	SD
2/12	2/12	*40	-	-	-		205	
2/12	2/13	*40	242/cc	52/cc	1.1/cc	4.3 [†]	212	
2/12	2/13	**40	16360	-	98.3		167 [†]	38.8 [†]
2/12	2/13	**30	26190	34.7	15.8	754	1.66 x 10 ³	2.2
2/13	2/13	**20	58900	36.7	18.9	1.6 x 10 ³	3.11 x 10 ³	1.94

Weight % Moisture

	Steam before dryer	Steam after dryer
40 Mw	23.2	0.6
30 Mw	0.133	0.06
20 Mw	0.062	0.032

*Evaporated on plates.

**Gross samples.

[†]Numbers used to determine DF across dryer for 40 Mw.

Curves of decontamination factors versus power are shown in Fig. 37-1. Steam quality at two locations is shown in Figs. 37-2 and 37-3.

Fig. 37-1

Decontamination factor vs power.

Fig. 37-2. Steam quality after dryer.

Fig. 37-3. Steam quality before dryer.

Boiling decontamination at the highest power (40 Mw) observed proved very poor. The steam dryer, in this case operating at 200 percent of design, proved surprisingly effective. At lower powers the dryer does not appear to be necessary since entrainment is very small.

Conclusions. This method is adaptable to other boiling reactor systems. Na^{24} is easily traced and can be used as a tool for comparing boiling characteristics of such reactors. If reactor water level is an important decontamination factor criterion, an optimum reactor level may be chosen using this tracer or others more adaptable to other reactors. A high, optimum decontamination factor water level would allow use of a high chimney and increase natural circulation.

Carryover in EBWR in the range tested increases with power. Danger of serious contamination of the reactor system due to a fuel element rupture increases as carryover increases.

EFFECT OF H₂ ADDITION TO REACTOR FEEDWATER ON RADIOLYTIC DECOMPOSITION

(Test Report No. 66, by G. E. Goring and A. P. Gavin)

Introduction and Summary. Preliminary tests to determine the effects of hydrogen addition to EBWR feedwater have been completed. The effectiveness of this method for reducing radiolytic water decomposition was conclusively demonstrated, supporting the results of similar experiments on BORAX-III¹ and BORAX-IV². However, the quantity of hydrogen required to cause a given fractional inhibition is considerably lower for EBWR. At power levels of 10 and 20 Mw (thermal), 40 cc (STP) of added hydrogen per liter of feedwater reduced the oxygen content of EBWR steam from 30-35 ppm to about 1 ppm. The BORAX-IV data, taken at a power level of 4 Mw, show only a 50 to 65 percent oxygen reduction for this same hydrogen addition.

Although small amounts of hydrogen cause a sharp decrease in water decomposition, the effectiveness of additional amounts diminishes rapidly, and the data imply that it may be impractical to attempt reducing the oxygen content of the steam from below 1/2 to 1 ppm by this method.

The previous experiments on BORAX-IV included monitoring the primary loop for N¹⁶ activity changes effected by the hydrogen addition. A similar secondary set of data was also taken in these EBWR tests, N¹⁶ activity being counted at two locations within the reactor vessel. The observations concurred qualitatively with BORAX-IV in that N¹⁶ favored the vapor (steam) phase in the presence of added hydrogen while remaining largely in the liquid phase during routine operation.

An attempt was also made to identify other activities and their transients by making a gamma scan of the inert gases downstream of the condenser. These results are not included in this report and will be released later.

Experimental Procedure. Using a calibrated Brooks rotameter (Fig. 66-4) and a manifolded cluster of four standard high-pressure cylinders, hydrogen gas was metered into the feedwater side stream leaving the ion-exchange system. The feedwater pressure at this point is 600 psig and the fluid moves directly into the main feedwater line to the reactor. After setting a hydrogen flowrate, a 20 minute period was

¹W. H. Zinn et al, Nuclear Science and Engineering, I (1956), 420-37.

²G. K. Whitham and R. R. Smith, "Water Chemistry in a Direct Cycle Boiling Water Reactor," 1958 Geneva Conference Paper, A/Conf. 15/P/946.

allowed to establish steady-state conditions before the following samples were taken:

1. Non-condensable gas
 - a. Steam (upstream of steam dryer)
 - b. Vent (downstream of recombiner)
2. Condensate (upstream of steam dryer)
3. Feedwater (downstream of feedwater pump)

Separation of the steam sample into condensate and inert gas was accomplished using techniques developed in previous tests,¹ the volumetric ratio of the two phases being determined simultaneously with their sampling. The gas samples were analyzed by orsat for oxygen and hydrogen. Liquid samples were analyzed for dissolved oxygen, using the Winkler method for the condensate and a more sensitive, indigo-carmin colorimetric method for the lower oxygen levels in the feedwater.

A separate, subsidiary set of data were taken to determine, at least qualitatively, any changes in activity levels around the plant as a function of the hydrogen addition rate. These measurements included:

1. N¹⁶ gamma counts within the reactor vessel; at the boiling interface and in the vapor space seven feet above the interface.
2. Broad gamma scan of the inert gas stream leaving the first ejector stage.
3. General background at various points in the plant, principally the steam dryer, the condenser hotwell, the feedwater filters and the ion exchange system.

Items 1 and 2 were obtained from probes and the counts recorded on instruments located in the control room or on the top floor of the containment vessel. Item 3 was determined by using a portable flux meter.

Results and Discussion. The primary data and calculations are tabulated in this report with a sample calculation. Results are plotted in Figs. 66-1, -2, and -3.

Figure 66-1 shows the effect of hydrogen addition rate on radiolytic decomposition, curves for 20 Mw and 10 Mw on EBWR being compared with a set of 4 Mw BORAX-IV data. Data points are shown to represent the 20 Mw curve. The 10 Mw EBWR points are not included since the material balances on this run did not close, and it was felt that the position and shape

¹R. E. Bailey and E. W. O'Keefe, Test Report no. 22A.

of the 10 Mw curve could not be defined with an absolute error less than 20 percent. Thus the curve should be regarded as no more than semi-quantitative, although it is certain that it lies below the 20 Mw curve.

Fig. 66-1. Effect of H_2 addition on oxygen content of steam.

Fig. 66-2. Effect of H_2 addition on N^{16} distribution within reactor.

The difference in slopes between the EBWR and BORAX-IV curves is striking. It is apparent that the same amount of added hydrogen is considerably more effective in EBWR than in BORAX-IV. The reason for this is not clear, but probably the major cause is better circulation around the EBWR core.

Figures 66-2 and -3 show the distribution of N^{16} activity within the reactor vessel as a function of H_2 addition, for the 10 Mw runs, respectively. The activity at both the chosen locations, at the boiling interface and in the vapor space, increased exponentially with hydrogen addition. This implies that the hydrogen purges N^{16} from the liquid phase by some mechanism that is not deducible from present knowledge. One possible explanation is that the nitrogen exists in the liquid phase in combination with oxygen (i.e., as the nitrate ion), and that the reducing environment provided by the

hydrogen disrupts the N-O bond, thus releasing the N^{16} . Further experiments, on both a bench scale and in the reactor, will be required to clarify this.

Fig. 66-3. Effect of H_2 addition on N^{16} distribution within reactor.

Fig. 66-4. Calibration curve for Brooks rotameter.

In general, the activity levels at various points in the steam loop and auxiliary equipment outside the reactor vessel increased as the hydrogen addition rate was raised. This is consistent with the effects shown by Figs. 66-2 and -3, namely, that increased hydrogen tends to concentrate the N^{16} in the vapor phase. A set of data for the 20 Mw run, taken with a portable flux meter, is given in Table I.

Table I - Effect of H₂ Addition Rate on EBWR Radiation Levels
(all values in mr/hr at 2 in.)

Location	Before H ₂ addition	H ₂ at 9.5 cc (STP) per liter of feedwater	H ₂ at 26 cc (STP) per liter of feedwater
<u>Steam System</u>			
Main steam header 4th level	38	40	100
Main steam line 3rd level	70	80	200
Steam dryer	140	110	260
<u>Condensate</u>			
Condenser hotwell	29	30	60
Feedwater pump #1	10	6	5
#2	4	6	9
Feedwater filter #1	8	8	36
#2	4	5.5	7
Feedwater pump Filter #1	5	8	29
#2	6	7	26
<u>Air Ejector Gas System</u>			
After cooler	330	290	190
Inter cooler	150	110	70
Flowmeter	42	41	39
Stack exhaust	100	130	90
<u>Cleanup System</u>			
Prefilter	26	33	12
IX column	34	25	80
After filter	26	21	100
Secondary cooler	140	110	80
Return line	5	5	5.5

Tabulation of Primary Data and Calculations. The primary data for both 20 Mw and 10 Mw runs are given in Table II. Calculations based on these raw data are tabulated in Table III. Note that all quantities in Table III are given as cc (STP) per liter of feedwater (or condensed steam).

Table II - Primary Data - EBWR Hydrogen Addition Runs

Point no.	Feedwater O ₂ , ppm	H ₂ addition rate, liters/min STP	Steam condensate*		Noncond. gas analysis			Recombiner effluent gas analysis		
			O ₂ content, ppm	Liters per liter of non-cond. gas	O ₂	H ₂	N ₂	O ₂	H ₂	N ₂
20 Mw Operation: Steam Flow 60,000 lb/hr										
1	0.1	0	6.0	16.8	31.2	66.9	1.9	-	-	-
4	0.1	4.3	4.5	20.9	27.1	68.6	4.3	15.1	-	-
6	0.075	8.9	2.8	33.0	16.4	78.6	5.0	4.6	0.5	94.9
7	0.05	5.2	3.4	25.2	24.4	72.3	3.3	12.8	0.4	86.8
9	0.05	6.3	3.0	27.7	21.4	75.9	2.7	10.6	-	-
10	0.05	14.6	0.7	35.8	4.3	93.0	2.7	0	12.8	87.2
11	0.05	17.7	0.3	29.1	1.7	95.4	2.9	0	22.1	77.9
10 Mw Operation: Steam Flow 30,000 lb/hr										
12	0.01	0	5.4	19.2	31.5	67.2	1.3	20.0	1.5	78.5
13	0.01	4.2	2.2	35.3	17.0	82.0	1.0	12.7	0.4	86.9
14	0.01	6.3	0.8	44.2	6.1	-	-	7.5	1.3	91.2
15	0.005	8.9	0.3	41.5	1.4	98.4	0.2	1.6	-	-
16	0.005	11.7	0.1	29.5	0.9	96.0	3.1	0.4	0.8	98.8

*Steam condensate temperature at sampling point:

for 20 Mw run, 40°C

for 10 Mw run, 24°C

Table III - Calculated Data - EBWR Hydrogen Addition Runs
(all quantities in cc (STP) per liter feedwater)

Point no.	Steam condensate										Metered H ₂	Cal. condenser air leak based on	
	Dissolved gas			Undissolved gas			Total gas					Excess H ₂	Metered H ₂
	O ₂	H ₂	N ₂	O ₂	H ₂	N ₂	O ₂	H ₂	N ₂				
20 Mw Operation													
1	4.6	7.1	0.1	18.6	39.8	1.1	23.2	46.9	1.2	0.5	0	-	-
4	3.5	6.4	0.3	12.9	32.8	2.1	16.4	39.2	2.4	6.4	9.5	45.9	68.1
6	2.2	7.4	0.3	5.0	23.8	1.5	7.2	31.2	1.8	16.8	19.5	48.3	56.0
7	2.6	5.5	0.2	9.7	28.7	1.3	12.3	34.2	1.5	9.6	11.4	50.2	59.5
9	2.3	5.9	0.1	7.7	27.4	1.0	10.0	33.3	1.1	13.3	13.8	57.0	59.0
10	0.5	8.4	0.2	1.2	26.0	0.8	1.7	34.4	1.0	31.0	32.2	57.9	60.1
11	0.2	9.3	0.2	0.5	32.8	1.0	0.7	42.1	1.2	40.7	39.0	63.2	60.6
10 Mw Operation													
12	4.1	5.3	0.1	16.4	35.0	0.7	20.1	40.3	0.8	0.1	0	-	-
13	1.7	5.0	-	4.8	23.2	0.3	6.5	28.2	0.3	15.2	18.5	78.7	95.8
14	0.61	5.6	-	1.4	21.2	-	2.0	26.8	-	22.8	27.7	75.5	91.5
15	0.23	9.8	-	0.34	23.7	-	0.57	33.5	-	32.4	38.7	82.0	97.8
16	0.08	5.1	-	0.31	32.5	-	0.39	37.6	-	36.8	51.5	89.0	124.0

Sample calculations:

a. Resolution of noncondensable gas analyses

Data point no. 1;

Basis: 1 liter of condensed steamO₂ content measured at 6 ppm (Table II)

$$\text{or } \left(\frac{6}{10^6}\right)\left(\frac{1000}{32}\right)(22400)\left(\frac{298}{273}\right) = \underline{4.6} \text{ cc at } 25^\circ\text{C \& 1 atm total press.}$$

Calculate H₂ content of condensed steam sample at 40°C = 104°F

Solubility at 104°F and 1 atm partial pressure,

$$\left. \begin{array}{ll} \text{O}_2 & 0.0230 \text{ ml/g H}_2\text{O} \\ \text{H}_2 & 0.0165 \end{array} \right\} \text{BMI-T-25*}$$

Assume Henry's law applies and that individual gas solubilities are independent of other dissolved gases.

Gas analysis (Table II)

O ₂	31.2 percent
H ₂	66.9
N ₂	1.9

Thus, equilibrium dissolved oxygen content would have been

$$(0.0230)(1000)(0.312) = \underline{7.2} \text{ cc}$$

so that sample was not at equilibrium.

Calculated H₂ content

$$\left(\frac{66.9}{31.2}\right)(4.6)\left(\frac{0.0165}{0.0230}\right) = 7.1 \text{ cc at } 25^\circ\text{C \& 1 atm total press.}$$

$$\text{volume of noncondensable gas} = \frac{1000}{16.8} = 59.5 \text{ cc at } 25^\circ\text{C and 1 atm (Table II)}$$

$\frac{\text{H}_2}{\text{O}_2}$ ratio in overall steam sample

$$= \frac{7.1 + (0.669)(59.5)}{4.6 + (0.312)(59.5)} = \underline{2.02}$$

or close to stoichiometric, which is an excellent check on the data and all the assumptions made above.

*H. A. Pray et al., "The Solubility of Hydrogen, Oxygen, Nitrogen, and Helium in Water at Elevated Temperatures." BMI-T-25, May 15, 1950.

O₂ content of overall steam sample

$$= [4.6 + (0.312)(59.5)] = \underline{23.2} \text{ cc (STP) per liter condensate}$$

(Listed in Table III)

b. Calculation of air leak into condenser

Take point no. 4 - Recombiner effluent gas analysis

O ₂	15.1
N ₂	84.9 (by difference)

$$\text{O}_2 \text{ disappearance} = \frac{84.9}{3.76} - 15.1 = 7.5$$

$$\therefore \text{equivalent H}_2 = (2)(7.5) = 15.0$$

Thus gas upstream of recombiner contains (per 100 cc of downstream gas)

N ₂	84.9
O ₂	22.6
H ₂	<u>15.0</u>
	122.5

(total gas leak based on 1 liter of condensed steam using figures from Table III.)

$$\left. \begin{array}{l} \text{based on excess H}_2 = (6.4) \left(\frac{84.9 + 22.6}{15.0} \right) = 45.9 \text{ cc} \\ \text{based on metered H}_2 = (9.5) \left(\frac{84.9 + 22.6}{15.0} \right) = 68.1 \text{ cc} \end{array} \right\} \begin{array}{l} \text{listed in} \\ \text{Table III} \end{array}$$

MEASUREMENT OF SUSPENDED AND DISSOLVED SOLIDS IN EBWR WATER

(Test Report No. 73A, by N. R. Grant)

To measure the suspended and dissolved solids in the water in EBWR, a pressurized evaporator has been built and tested. The basic design allows water from any place in the primary system to be pumped into a weighed crucible in an autoclave which is operating at 100°F above the saturation temperature. The steam is condensed, expanded to atmospheric pressure, and the total water flow measured. By using various safety features in the system, the experiment may be operated continuously with only occasional maintenance (Fig. 73A-1).

Fig. 73A-1. Suspended and dissolved solids measurement

Two tests were completed on EBWR water. The first was run on the feedwater, but a malfunction in the autoclave would not allow removal of the platinum crucible for a weight determination. After repair of the equipment had been completed, only four days remained until shutdown. Although a longer test period was desirable, the second run was made, this time using reactor water. A total of 107 liters of water was passed through the apparatus. The autoclave was operated at an average temperature of 650°F with a pressure of 700 psi.

After the platinum crucible was removed from the autoclave, a determination of the gross activity was made on a 200 channel analyzer. The activities present were Co^{58} and Co^{60} at a ratio of about 10 to 1. The weight gain of the crucible was 7.7 mg. The solids concentration was 0.07 ppm for the 107 liters of reactor water evaporated.

The crucible was sent to Chemistry for analysis of the solids. By using fuming sulfuric acid it was possible to remove 5.1 mg of solids, and determine the values shown in Table I.

Table I - Solids Content of EBWR Water

Analyzed	Equivalent in pure oxide	Amount, mg		Percent	
Al	$\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$	1.59		59%	
			3.53		70%
Fe	Fe_2O_3	0.68		25%	
			0.97		19%
Ni	NiO	0.43		16%	
			0.54		11%
		2.70	5.04	100%	100%

The assumption that the oxide formed by the metal is the pure form as indicated is in error, however, it shows the relative amounts of oxides present. In comparison to the oxide found on the fuel elements, the aluminum content and the nickel content are about the same, and the iron content found in the water is higher.

It is unfortunate that only one run was successfully completed at 20 Mw, since a comparison of solids in the water at different locations and at higher power levels would have been of interest. Testing will be continued when the reactor is started again. The system will be modified to use a smaller autoclave, and tests will be run for longer durations. This will increase the amount of solids collected and thereby increase the accuracy of the experiment.

CORROSION OF EBWR STEAM LINE SAMPLES

(Test Report No. 79, by N. R. Grant)

Six types of construction materials used in the EBWR turbine were received from the vendor. Coupons were machined from these pieces for insertion into the EBWR six-inch steam line. A sample holder was fabricated and welded to a flange which was then located in a tee just before the steam dryer. The first sample loading was made in November, 1957, and samples were withdrawn three times during the next nineteen months. Corrosion of all six materials was good until the last month of operation prior to shutdown for conversion to higher power operation. During that period pitting and accelerated corrosion occurred. This may have been due to a small addition of oxygen to the reactor water for a short time, or to the samples having been submerged in water in the steam line for five weeks before reactor startup.

The materials supplied by Allis-Chalmers are described in Table I.

Table I - Materials Tested

No.	Designation	Material	Turbine part	A-C code no.
1	ASTM A-113 SSC	structural steel	diaphragm web	3043-489 A-133
2	AISI 4140	steel	wheels	3043-489 SAE 4140
3	403 SS	13% Cr stainless steel	buckets	3043-489 403 SS
4	-	1% Cr, 1/2% Mo steel	shaft	3043-489 ACM 153
5	-	low carbon steel	diaphragm band	3043-489 SSC
6	-	13% Cr stainless steel non-hardening	nozzle element	ACM 1413

Eight samples of each material were machined to 2.375 x 1.000 x 0.062 in., with a 0.265 hole near each end for mounting in the holder. All samples were cleaned in acetone and weighed. Four samples of each material were loaded into the holder. They were separated with spacers and supported by bolts made of 304 stainless steel.

The test was operated for three periods as indicated in Table II. One sample of each type was removed at the end of each period. At the end of the first period, one sample of each type was also sent to Chemical Engineering Division for a decontamination test. The remainder of the new samples were added to the holder at the end of the first and second periods. In order to determine the metal loss, the oxide was removed with Clark's Solution (100 gm conc. HCl, 2 gm Sb₂O₃, 5 gm SnCl₂). The

antimonious oxide inhibits the attack of the metal, and the stannous chloride reduces the ferric chloride formed by the acid reaction with the oxide.¹ Temperature of the solution was kept at $68 \pm 4^\circ\text{F}$ and the solution was continuously agitated. The normal immersion time for most of the samples was approximately ten minutes. A blank sample of each material was immersed for that length of time to determine metal attack due to stripping. Although the attack was small in each case, each corrosion coupon was corrected for the metal loss.

Table II - Test Results

Period	Inclusive dates	Reactor operation time				Alloy and sample no.	Metal loss, mg/cm ²
		For the period		Total			
		Hours	Mw-hr	Hours	Mw-hr		
1	11/2/57 to 4/6/58	2,249	41,861	2,249	41,861	1C	2.05
						2A	1.14
						3A	0.22
						4D	1.24
						5A	2.76
						6A	0.31
2	4/11/58 to 11/8/58	3,522	67,272	5,771	109,133	1A	2.98
						2C	1.47
						3C	0.19
						4A	1.33
						5C	3.70
						6C	0.18
3	4/29/59 to 7/3/59	798	16,178	6,569	125,311	1D	4.19
						2D	3.58
						3D	0.46
						4C	16.90
						5D	3.23
						6D	0.25
		798	16,178	798	16,178	1E*	11.68
						2E	11.97
						3E	0.23
						4H	9.70
						5G	15.58

*These five samples placed in holder at start of third period.

¹R. Fowler, Jr., D. L. Douglas, F. C. Zydes, "Corrosion of Reactor Structural Materials in High-Temperature Water: Descaling Methods," KAPL-1198, August 27, 1954.

The appearance of the samples after the first period was satisfactory (Fig. 79-1); even the corrosion rates over the first two periods were good (Fig. 79-2).

Fig. 79-1. Corrosion samples from EBWR steam line - first set

Fig. 79-2
Results of EBWR steam
line corrosion test

However, during the last month, the low alloy samples exhibited accelerated corrosion and pitting attack (Figs. 79-3, -4, -5, -6, -7). The new samples placed into the holder at the start of the last month were even worse than those which had been in the system since the start of the test, indicating that the oxide on the old samples offered some protection against attack.

Fig. 79-3. Corrosion samples from EBWR steam line - third set

IA - 5771 HOURS

ID - 6569 HOURS

IE - 798 HOURS

Fig. 79-4. Micrographs of structural steel samples corrosion tested in EBWR steam. (10X)

2C - 5771 HOURS

2D - 6569 HOURS

2E - 798 HOURS

Fig. 79-5. Micrographs of AISI 4140 steel samples corrosion tested in EBWR steam. (10X)

4A - 5771 HOURS

4C - 6569 HOURS

4H - 798 HOURS

Fig. 79-6. Micrographs of 1% Cr, 1/2% Mo steel samples corrosion tested in EBWR steam. (10X)

5C - 5771 HOURS

5D - 6569 HOURS

5G - 798 HOURS

Fig. 79-7. Micrographs of low carbon steel samples corrosion tested in EBWR steam. (10X)

There were two unusual occurrences during the course of the last month that might explain the unusual results. The first was that a quantity of oxygen was added to the reactor water for about two hours. The concentration of oxygen in the steam was about seven times as high as that found during normal operation. However, the addition was made only one week before the end of the test, and it is questionable whether this amount of damage could have been caused by a few hours of high oxygen followed by a week of normal operation. The second occurrence was that the reactor was not started until five weeks after the samples were placed into the steam line tee, and during that time the steam line was filled with reactor water. Previously the samples had been loaded just prior to reactor startup, and although that was also the intention this time, various delays caused the samples to be submerged in water for this extended period. An additional complication is that due to modifications, the reactor had not been operated during the five weeks previous to sample loading, thus the water quality before the reactor was finally started is questionable. Once again the exact value of this factor is unknown, because the steam line was flooded on three other occasions during shutdowns in the first and second corrosion periods.¹

It is difficult to determine which of the two possible explanations is the cause of the increased corrosion. However, it may be worthwhile to examine the turbine, at least through the manhole located near the low pressure end.

The activities of the last set of samples were measured on a 200 channel analyzer. Results indicate the major activities to be Co^{58} and Co^{60} at a ratio of approximately 6 to 1. This is a slightly lower ratio than that found for the reactor water (10 to 1).²

Plans have been made to allow testing of samples in two or three steam line positions during elevated power operation. Samples of other alloys of interest can be inserted into one of the new facilities which will be available. In an attempt to avoid repetition of the excessive pitting of this test, loading of samples will be made just prior to reactor startup, and samples will be removed if the steam line is to be flooded for more than a few days.

¹From 12/27/57 to 1/25/58 during the first period, and from 5/14/58 to 6/3/58 and from 7/12/58 to 7/19/58 of the second period.

²Test Report no. 73A.

RADIOCHEMICAL ANALYSIS OF SAMPLES FROM THE EBWR SHOCK SHIELD

(Test Report No. 104, by R. J. Armani)

Two samples were submitted from the EBWR shock shield for radiochemical analysis. Approximately .007 in. was ground from the top surface and .007 in. from the bottom surface of a specimen from the shock shield, and the resultant grindings constituted one sample. Circular slugs were cut from the shock shield, and the resultant chips constituted the second sample.

The samples were dissolved in hydrochloric acid, and an analysis was made for various radioisotopes, assuming that these isotopes were present. They were separated, decontaminated and counted. This method of analysis indicated not only the presence of certain radioisotopes, but the absence of others as well.

Since EBWR has been shut down, none of the short-lived nuclides were detected. Corrections for decay were made on the detected nuclides to July 15, 1959. On April 20, 1960, the specific activity of the grindings sample was 827.4 d/s/gram of iron, and for the chips sample, 1306 d/s/gram of iron. The specific activity on July 15, 1959 for the grindings sample was 4165 d/s/gram of iron and 5627 d/s/gram of iron for the chips samples. The results of the radiochemical separation are given in Table I. The isotopes listed consist of those for which the analysis was specifically performed. The results are given in percent of total activity on July 15, 1959.

Table I

Isotope	Grindings sample	Chips sample
Sb ¹²⁴	A*	A*
Mn ⁵⁴	A	A
Ni ⁶⁵	A	A
Cr ⁵¹	A	A
Co ⁵⁷	A	A
Co ⁵⁸	84.3%	80.3%
Co ⁶⁰	15.7%	19.7%
Fe ⁵⁹	A	A
Zn ⁶⁵	A	A
Cd ¹¹³	A	A
Cd ¹¹⁵	A	A

*Isotopes marked A are absent.

There is no reason to believe that the percentage of active elements should be different in the two samples, because the samples were taken from the same source. Table I indicates that there is a small difference in the two samples. This is explained in the following way. The specific activity of the samples was very low. Long counting times still resulted in statistics which were ± 5 percent at best. So the indicated values in Table I could easily vary ± 5 percent.

A smear from the walls of the EBWR shock shield was taken and analyzed by the utilization of a 2 x 2 in. Na-I crystal and a single channel pulse height analyzer. The resultant peaks were attributed to Co^{58} and Co^{60} . The activity ratio of Co^{58} to Co^{60} was found to be 1.79 to 1.0 on December 30, 1959 and 8.42 to 1.0 on July 15, 1959. This is equivalent to 89.4 percent Co^{58} and 10.6 percent Co^{60} . The smear sample was taken and analyzed by personnel of the Industrial Hygiene and Safety Division.

Previous investigations indicate that 90 percent of the activity on the shock shield was on the surface, so that the specific activity of the grindings sample should be greater than the specific activity of the chips sample. This investigation shows the reverse to be true. It is suspected that the samples do not truly represent the surface and volume conditions. The grindings and chips were not prepared specifically for the radiochemical analysis, but rather were the results of other operations. Therefore, there is a question as to the interpretation of the specific activity results.

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