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EBWR TEST REPORTS
(100-MWT OPERATION)

EBWR Project Group

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TABLE OF CONTENTS

	<u>Page</u>
NUMERICAL INDEX OF TEST REPORTS	26
CONTRIBUTORS	31
ACKNOWLEDGMENTS - PARTICIPATING PERSONNEL	32
I. INTRODUCTION.	33
II. FUEL	
EBWR Core Loading No. 55 (Test Report No. 110, dated March 10, 1960, by V. M. Kolba)	41
EBWR Core Loading No. 56 (Test Report No. 110C, dated April 22, 1960, by V. M. Kolba).	46
Antimony Source Transfer (Test Report No. 110F, dated November 10, 1960, by V. M. Kolba)	50
EBWR Core Changes (Test Report No. 110G, dated November 30, 1960, by V. M. Kolba)	52
EBWR Core Loading No. 57 (Test Report No. 110H, dated January 18, 1961, by V. M. Kolba)	52
EBWR Core Loading No. 58 (Test Report No. 110I, dated January 18, 1961, by V. M. Kolba)	60
EBWR Fuel (Test Report No. 110K, dated April 17, 1961, by V. M. Kolba)	67
EBWR Core Loading No. 59 (Test Report No. 110L, dated June 12, 1961, by V. M. Kolba)	68
Replacement of EBWR Source (Test Report No. 110M, dated October 23, 1961, by V. M. Kolba)	71
Reloading of Storage Pit After Cleaning (Test Report No. 110M Addendum, dated November 14, 1961, by V. M. Kolba).	73
Rotation of Spike Fuel Elements (Test Report No. 110N, dated November 22, 1961, by E. L. Martinec)	74

TABLE OF CONTENTS (Cont'd.)

	<u>Page</u>
Examination of Irradiated Boron Strips on Spike Fuel Element Frame (Test Report No. 110N Addendum, dated November 22, 1961, by V. M. Kolba)	76
Transposition of Enriched Thin Fuel Elements (Test Report No. 110O, dated January 10, 1962, by V. M. Kolba)	78
EBWR Loading 1B (24 Spike): Core Loading No. 60 (Test Report No. 110P, dated February 2, 1962, by E. L. Martinec).	83
Core Loading for 100-Mw Operation, Core Loading No. 61 (Test Report No. 100J Addendum and Supplements 1, 2, and 3, dated March 2, 1962, by V. M. Kolba).	87
EBWR Core Loading No. 62 (Test Report No. 110Q, dated May 24, 1962, by V. M. Kolba)	96
Core Loading No. 63 (Operation "Strip-Tease"), (Test Report No. 110R(A), dated October 2, 1962, by V. M. Kolba).	100
Examination of 1.1% Boron-Stainless Steel Burnable Poison Strips (Test Report No. 110O Addendum, dated November 7, 1962, by V. M. Kolba)	108
Measurement of Water Channels on EBWR Core 1 Fuel Elements (Test Report No. 114, dated September 22, 1961, by C. F. Reinke and W. N. Beck)	114
Measurement of Water Channels on EBWR Core 1 Fuel Elements (Test Report No. 114A, dated January 25, 1963, by C. F. Reinke, W. N. Beck, and W. A. Ahrens)	118
Analysis of Scale from EBWR Fuel Plates (Test Report No. 112A, dated January 21, 1963, by W. G. Knapp).	123
Detection of Possible Growth in Thin Enriched (ET) Fuel Plates of EBWR with High Burnup Exposure (Test Report No. 107, dated January 31, 1963, by H. Iskenderian and J. Haugsnes).	127
 III. PHYSICS	
Analysis of EBWR Startup Transients (Test Report No. 120A, dated February 1, 1962, by K. Almenas, N. Suda, and N. Ito).	133

TABLE OF CONTENTS (Cont'd.)

	<u>Page</u>
Temperature and H_3BO_3 Effect on EBWR Subcritical Counting Rate (Test Report No. 108C, dated June 19, 1961, by K. Almenas and T. Nomura)	163
Experimental EBWR Temperature Coefficients (Test Report No. 108D, dated June 30, 1961, by K. Almenas and J. Haugsnes) . .	174
Void Coefficient Measurements in EBWR (Test Report No. 124, dated August 27, 1962, by H. P. Iskenderian)	187
Relative Reactivity Measurements of EBWR Fuel Elements (Test Report No. 100D, dated December 21, 1961, by K. Almenas and V. Kolba)	194
Reactivity Effect of Rotating the Boron-Stainless Steel Strip Containing Spike Elements (Test Report No. 100F, dated November 24, 1961, by K. Almenas).	208
Completion of Loading 1A, with the Addition of Four Peripheral Spikes and One ET Element - Destripping of Eight Spikes by One Strip Each - Verification of the Reactor Meeting the Cold Shutdown Requirements - A Partial Calibration of the Control Rod System (Test Report No. 100J, dated March 8, 1962, by H. P. Iskenderian). .	215
Physics Termination Tests on EBWR, Core Loading 1A (Test Report No. 100N, dated January 14, 1963, by H. P. Iskenderian) . .	220
Reactivity Aspects of EBWR Core 1B (Test Report No. 100H, dated February 1, 1962, by K. Almenas).	226
Destripping of 16 More Spikes for a Gain of $\sim 2\%$ Reactivity - Check of 9-Rod Cold Shutdown Requirement (Test Report No. 110R(B), dated October 8, 1962, by H. P. Iskenderian).	231
EBWR Neutron Flux Distribution during Cold Critical Operation (Test Report No. 100K, dated October 23, 1962, by W. G. Knapp). .	235
Flux Distribution Measurements in EBWR during Operation (Test Report No. 100P, dated January 22, 1963, by W. G. Knapp).	258
Study of Operational Data from EBWR at 0-60-Mw Levels - $\Delta k/\Delta P$ Increasing with Power (Test Report No. 100M, dated October 1, 1962, by H. P. Iskenderian)	263

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
EBWR Reactor Transfer Function Measurements (Test Report No. 123, 123A, 123B, and 123C, dated January 31, 1963, by W. C. Lipinski, A. Hirsch, C. Hsu, and G. Popper)	268
IV. CONTROL AND CONTROL RODS	
Investigation of Irradiated EBWR Control Rods (Test Report No. 121A, dated November 15, 1960, by T. L. Kettles)	289
Investigation of Irradiated EBWR Control Rods (Test Report No. 121B, dated February 28, 1961, by T. L. Kettles)	294
Investigation of Irradiated EBWR Control Rods (Test Report No. 121C, dated March 21, 1961, by T. L. Kettles)	307
Radiochemical Analysis of Original EBWR Control Rod and Follower Samples (Test Report No. 121D, dated September 8, 1961, by R. J. Armani)	322
Boron Burnup in Original EBWR Control Rod No. 4 (Test Report No. 121E, dated September 21, 1961, by R. J. Armani)	325
Radiochemical Analysis of Crud Removed from the Bottom of EBWR Control Rods No. 4 and No. 9 (Test Report No. 112, dated November 30, 1960, by R. J. Armani)	326
Disposition of Original EBWR Control Rods (Test Report No. 121F, dated October 27, 1961, by T. L. Kettles)	327
Partial Investigation of Irradiated EBWR Control Rods (Test Report No. 121G, dated January 28, 1963, by T. L. Kettles and N. Balai)	329
Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth (Test Report No. 100C, dated May 11, 1960, by K. Almenas)	339
Calibration of Control Rod No. 9 (Test Report No. 100E, dated June 14, 1961, by K. Almenas)	345
EBWR Control Rod Replacement (Test Report No. 119, dated January 8, 1962, by V. M. Kolba, E. A. Wimunc, E. L. Martinec, and J. F. Matousek)	353

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
A Preliminary Calibration of the New System of Control Rods in the Cold EBWR (Test Report No. 100G, dated January 22, 1962, by H. P. Iskenderian)	363
Inspection of the EBWR Control Rod (Test Report No. 119A, dated July 6, 1962, by T. L. Kettles)	367
Calibration of Control Rod System in EBWR - Empirical Determination of β_{eff} (Test Report No. 100I, dated January 31, 1963, by H. P. Iskenderian)	368
 V. HYDRAULICS AND HEAT TRANSFER	
Temperature Distribution in the EBWR Core (Test Report No. 108A, dated November 18, 1960, by K. Almenas and J. Haugsnes)	375
Preliminary Hydrodynamic Tests on EBWR with Modified Internals (Test Report No. 105, dated August 1, 1961, by M. Petrick and E. A. Spleha. Incorporates "Errata for Test Report No. 105," dated August 16, 1961.)	378
Additional Hydrodynamic Test Data from EBWR at Powers to 60 Mw (Test Report No. 105A, dated October 3, 1962, by M. Petrick)	395
An Explanation of the Behavior of EBWR at the Higher Power Levels (> 65 Mw) (Test Report No. 105B, dated November 29, 1962, by M. Petrick and E. A. Spleha. Incorporates Test Report No. 105B-Addendum, dated December 3, 1962).	398
Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly (Test Report No. 106A, dated December 11, 1961, by G. F. Popper)	404
Further Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly (Test Report No. 106B, dated January 12, 1962, by G. F. Popper)	414
A Terminal Report on the Hydrodynamic Data Obtained from the EBWR Instrumented Fuel Assemblies (Test Report No. 106C, dated January 31, 1963, by G. F. Popper)	416

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
Reactor Steam Quality by Radioactive Sodium Tracer (Test Report No. 118, dated March 29, 1962, by W. G. Knapp)	424
EBWR Steam Quality during High Power Operation (Test Report No. 118A, dated January 24, by W. G. Knapp).	430
 VI. COMPONENTS	
Volume of Water in the EBWR Reactor (Test Report No. 103, dated May 10, 1960, by C. E. Carson).	437
EBWR Turbine Seal Leak Tests, Preliminary Results (Test Report No. 101A, dated April 12, 1960, by R. J. Gariboldi, D. R. Jacobson, and H. M. Sroka)	439
Relative Effectiveness of Sb-Be Source and Core-Induced Neutrons (Test Report No. 102, dated April 14, 1960, by A. Hirsch)	442
Source Level at Startup Counters and Sensitivity of Counters (Test Report No. 102A, dated April 28, 1960, by A. Hirsch)	444
Instrumentation for Startup (Test Report No. 102B, dated May 5, 1960, by A. Hirsch).	445
Effect of Source Installed in EBWR on November 8, 1960 (Test Report No. 102C, dated November 10, 1960, by A. Hirsch)	447
Frequency Response of EBWR High Flux Trip Circuit (Test Report No. 102D, dated January 31, 1963, by A. Hirsch and W. C. Lipinski)	448
EBWR Building Shell Leak Rate Test (of October 1960) (Test Report No. 125, dated May 24, 1962, by E. A. Wimunc)	453
EBWR Building Shell Leak Rate Test (of August 1962) (Test Report No. 125A, dated August 23, 1962, by E. A. Wimunc and V. M. Kolba)	455
Radiochemical Analysis of Samples from the EBWR Shock Shield (Test Report No. 104, dated May 12, 1960, by R. J. Armani)	457

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
EBWR Feedwater Filters - Cleanliness Determination (Test Report No. 116, dated May 19, 1961, by E. Swanson)	459
EBWR Subcooler Cleaning (Test Report No. 117, dated May 24, 1961, by J. Matousek)	462
Cleaning of Vent Condenser (Test Report No. 117A, dated July 13, 1961, by J. Matousek)	464
Performance Investigation of EBWR Purification and Decay Heat Removal Systems (Test Report No. 117B, dated April 3, 1962, by T. L. Kettles)	465
Test of Control Boric Acid Injection System (Test Report No. 109, dated March 8, 1962, by E. A. Wimunc)	471
Test of Control Boric Acid Addition to the EBWR during Power Operation (Test Report No. 109A, dated June 19, 1962, by W. G. Knapp. Includes Memo, "Boric Acid Concentration in EBWR Condenser Hotwell Prior to and during Reactor Operation," dated April 3, 1962, by W. G. Knapp.)	474
Safety Aspects of Operating Borated EBWR (Test Report No. 120, dated January 15, 1962, by K. Almenas, N. Suda, and N. Ito).	478
Heat Balance Data on EBWR 100-Mw Plant (Test Report No. 122, dated January 28, 1963, by E. Martinec).	487
Radiation Levels in Reboiler Building and Sub-Reactor Room (Test Report No. 126, dated November 27, 1962, by E. L. Martinec)	504
Short Duration Test of Demister Material (Test Report No. 111, dated February 13, 1962, by V. M. Kolba).	509
Second Test of Demister Material (Test Report No. 111A, dated July 12, 1962, by V. M. Kolba)	513
 VIII. APPENDICES	
A. EBWR Control Rod - Fuel Element Location (Memo, dated March 7, 1962, by V. M. Kolba).	517

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
B. Investigation of SA-212 Grade B Material Removed from the EBWR Pressure Vessel Wall (R.E. No. 763) (Memo, dated April 24, 1961, by T. L. Kettles)	518
C. Summary Report - Investigations Into Causes for Catastrophic Pre-Operational Failures of the 80-Mw EBWR Primary Reboilers (Memo, dated June 29, 1962, by N. Balai, J. F. Matousek, C. R. Breden, and M. Odie)	536
D. Summary of EBWR Test Procedures and Reports during the Period 1960-1962	566
BIBLIOGRAPHY	570

LIST OF FIGURES

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
Section I. INTRODUCTION		
1	EBWR Operating Levels and Significant Events (1960). .	35
2	EBWR Operating Levels and Significant Events (1961). .	36
3	EBWR Operating Levels and Significant Events (1962). .	37
Section II. FUEL		
110C-1	Core - Fuel Status Board (Core Load No. 56, 4/12/60) .	48
110C-2	Storage Pit - Fuel Status Board (Core Load No. 56, 4/12/60)	49
110H-1	Thermal Flux in EBWR Elements (4/22/60)	57
110H-2	Thermal Flux in EBWR (4/22/60)	57
110H-3	Core - Fuel Status Board (Core Load No. 57, 4/28/60) .	58
110H-4	Storage Pit - Fuel Status Board (Core Load No. 57, 4/28/60)	59
110I-1	Core - Fuel Status Board (Core Load No. 58, 1/18/61) .	65
110I-2	Storage Pit - Fuel Status Board (Core Load No. 58, 1/18/61)	66
110K-1	Storage Pit - Fuel Status Board (4/14/61).	67
110L-1	Core - Fuel Status Board (Core Load No. 59, 6/6/61). .	69
110L-2	Storage Pit - Fuel Status Board (Core Load No. 59, 6/6/61).	70
110M-Add.-1	Storage Pit - Fuel Status Board (11/8/61).	73
110N-1	Core - Fuel Status Board (Core Load No. 59a, 11/14/61).	75

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
110O-1	Changed Fuel Elements	80
110O-2	Core - Fuel Status Board (Core Load No. 59b, 1/4/62) .	81
110O-3	Storage Pit - Fuel Status Board (Core Load No. 59b, 1/4/62).	83
110P-1	Core - Fuel Status Board (Core Load No. 60, 1/26/62). .	86
110P-2	Storage Pit - Fuel Status Board (Core Load No. 60, 1/26/62).	87
110J-Add.-1	Boric Acid Removal Tests (9-Rod Shutdown).	95
110J-Add.-2	Core - Fuel Status Board (Core Load No. 61, 2/23/62) .	95
110J-Add.-3	Storage Pit - Fuel Status Board (Core Load No. 61, 2/23/62).	96
110Q-1	Core - Fuel Status Board (Core Load No. 62, 5/21/62) .	98
110Q-2	Storage Pit - Fuel Status Board (Core Load No. 62, 5/21/62).	99
110R(A)-1	Effect of Boric Acid Removal.	104
110R(A)-2	Effect of Boron Strip Removal	105
110R(A)-3	Core - Fuel Status Board (Core Load No. 63 (Rev), 9/27/62).	106
110R(A)-4	Storage Pit - Fuel Status Board (Core Load No. 63 (Rev), 9/27/62).	107
110O-Add.-1	A Scan of Gamma (γ) Activity Level of a Boron Strip Removed from EBWR Irradiated Spike Fuel Element . .	110
110O-Add.-2	Metallographic Samples (As Received and Irradiated Section 2)	112
110O-Add.-3	Metallographic Samples (Irradiated Section 3).	113
112A-1	Surfaces of Fuel Assemblies from Which Samples of Scale Were Obtained.	124

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
107-1	Installation of Incore Flux Monitors.	128
107-2	"B" Detector Versus Power Characteristics, $P \simeq 1.02D/1 + 0.0026D$ (D = Detector Current in μA , P = Power in Mw).	129
Section III. PHYSICS		
120A-1	EBWR Startup Transient for 40.5-sec Startup Period (No H_3BO_3).	142
120-2	Analog Trace of Test 1-A Startup Transient	142
120A-3	EBWR Startup Transient for 24.5-sec Startup Period (No H_3BO_3).	143
120A-4	Analog Trace of k_{ex} , Power, and Energy Time Depend- ence for Test 2-A	143
120A-5	EBWR Startup Transient for 15-sec Period (No H_3BO_3).	144
120A-6	Analog Trace of k_{ex} , Power, and Energy Time Depend- ence for Test 3-A.	144
120A-7	EBWR Startup Transient for 40-sec Startup Period (3.66 gm/gal H_3BO_3).	145
120A-8	Analog Trace of k_{ex} , Power, and Energy Time Depend- ence for Test 1-B	145
120A-9	EBWR Startup Transient for 26.5-sec Startup Period (3.66 gm/gal H_3BO_3).	146
120A-10	Analog Trace of k_{ex} , Power, and Energy Time Depend- ence fot Test 2-B.	146
120A-11	EBWR Startup Transient for 15-sec Period (3.66 gm/gal H_3BO_3).	147
120A-12	Analog Trace of k_{ex} , Power, and Energy Time Depend- ence for Test 3-B	147

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
120A-13	Compensated Reactivity Versus Energy	149
120A-14	Analog Trace of Compensated Reactivity Versus Energy	149
120A-15	Compensated Reactivity Versus Energy	150
120A-16	Analog Trace of Compensated Reactivity Versus Energy	151
120A-17	Temperature Versus Excess Reactivity	151
120A-18	Sandborn Record of Test 3-B Startup Transient.	154
120A-19	Temperature Versus Energy	155
120A-20	$P_{(tm)}$ and $E_{(tm)}$ Versus Reciprocal Startup Period.	155
120A-21	Circuit Simulating the Coupling Equation [Eq. (B-3)]	161
108C-1	Counter Correction Factor (Normalized at 20°C. No H_3BO_3)	169
108C-2	Counter Correction Factor (Normalized to the Zero H_3BO_3 , 20°C Condition).	169
108C-3	Counting Rates Versus Temperature (No H_3BO_3)	171
108C-4	Counting Rates Versus Temperature (~ 3.6 gm/gal H_3BO_3).	171
108D-1	EBWR Core Temperatures on May 8, 1961	175
108D-2	EBWR Core Temperatures on May 9, 1961	175
108D-3	EBWR Core Temperatures on May 10, 1961	176
108D-4	dn/dt Versus $n_{(t)}$	179
108D-5	Boric Acid Concentration during Temperature Coefficient Tests	180
124-1a	Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements	187

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
124-1b	Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements	188
124-1c	Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements	188
124-1d	Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements	189
124-1e	Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements	189
124-2	Calibration of Rod No. 2. Differential Worth of Rod as Indicated.	190
124-3	Thermal Neutron Flux (Integrated for Each Element) Radial Distribution.	191
124-4	EBWR Void Coefficient Measurements for Regions Con- taining ET Elements Only ($\Delta\rho/\Delta v$ Versus H_3BO_3 Concentration).	193
100D-1	Location of Mapped Elements in Northwest Quadrant of EBWR Core.	197
100D-2	Integral Worth of No. 9 Rod with Remaining Eight Rods Banked at 21.1 in.	198
100D-3	Σ_{ET}^* Versus Reactor Mwd Burnup	198
100D-4	Experimentally Measured EBWR Burnup (Radial).	205
100F-1	$dn_{(t)} / dt$ Versus $n_{(t)}$ for Positive Periods	214
100J-1	EBWR Bank Criticalities Forming the Pattern of Core Loading 1A	218
100J-2	Bank Calibration of EBWR Control Rods (Zircaloy Followers)	219
100J-3	Worth of Central Control Rod No. 9 When Rods (Nos. 1-8) Are At 10.5 in. (Above the Bottom of the Core)	219

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
100N-1	EBWR Operation 0-40 Mw (8/13/62) Before Final Destripping of 16 Spikes	223
100N-2	Reactivity Versus Reactor Power	225
100N-3	Bank Calibration of Control Rods with Core Loading 1A	225
100H-1	Reactivity Changes During Loading of Core 1B	227
100H-2	Dilution Rate Versus $1 - k_{eff}$ (~250-gal Water Added/5 min.)	228
100H-3	$1 - k_{eff}$ Versus H_3BO_3 Concentration.	230
110R(B)-1	EBWR Destripping Operations of 16 Spikes By One Strip Each	234
110R(B)-2	EBWR Core 1A (10/1/62) Cold Criticality.	234
100K-1	Upper End of Flux Wire Shield Assembly	240
100K-2	Neutron Flux Isograms (S.W. Quadrant along Line A [Fig. 100K-6] . Boric Acid Concentration 125 ppm. Control Rods Banked at 10.25 in.)	241
100K-3	Neutron Flux Isograms (S.W. Quadrant along Line A [Fig. 100K-6] . Boric Acid Concentration 900 ppm. Control Rods at 16.65 to 17.0 in.)	242
100K-4	Neutron Flux Isograms (Same as Fig. 100K-3 Except along Line B [Fig. 100K-6].)	243
100K-5	Neutron Flux Isograms (S.W. Quadrant along Line A [Fig. 100K-6]. Boric Acid Concentration 1900 ppm. Control Rods at 32.5 in. Except Rod No. 9 at 32.3 in.) . .	244
100K-6	EBWR Core Loading No. 62, S.W. Quadrant and Adjacent Assemblies (Activations plotted in Figs. 100K-1 through 5 made along Lines A and B. Other wire-foils simultaneously irradiated in several or all positions marked with an "X").	245
100K-7	Axial Neutron Flux Distribution	246

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
100K-8	Axial Neutron Flux Distribution	246
100K-9	Axial Neutron Flux Distribution	247
100K-10	Axial Neutron Flux Distribution	247
100K-11	Neutron Flux Isograms (S.W. Quadrant along Line A [Fig. 100K-13]. Boric Acid Concentration 180 ppm. Control Rods Banked at 7.55 in.)	248
100K-12	Neutron Flux Isograms (Same as Fig. 100K-11 Except along line B [Fig. 100K-13].)	249
100K-13	EBWR Core Loading No. 63, S.W. Quadrant and Adjacent Assemblies (Activations Plotted in Figs. 100K-11, 12, 16, and 17 Made along Lines A and B, Wire-foil also Irradiated in Position Marked "X".)	250
100K-14	Axial Neutron Flux Distribution	251
100K-15	Axial Neutron Flux Distribution	251
100K-16	Neutron Flux Isograms (S.W. Quadrant along Line A [Fig. 100K-13]. Boric Acid Concentration 1110 ppm. Control Rods Banked at 15.4 in.)	252
100K-17	Neutron Flux Isograms (Same as Fig. 100K-16 Except along Line B [Fig. 100K-13].)	253
100K-18	Axial Neutron Flux Distribution (Center of Core)	254
100P-1	Location of Inconel Tubing "Irradiation Loops" Used for Neutron Flux Distribution Measurements in the Southeast Quadrant of the Core and the Downcomer of EBWR during Power Operation.	260
100M-1	EBWR Operation at 20-Mw Power Level (3/30 - 4/5/62)	264
100M-2	EBWR Operation at 40-Mw Power Level (9/1 - 9/4/62).	265
100M-3	EBWR Operation at 40-60-Mw Power Levels (9/5 - 9/10/62)	265

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
100M-4	Period Measurement Data Converted Using $\beta = 6.55 \times 10^{-3}$	266
100M-5	Reactivity Increments Required for Indicated Power Increases	267
Section IV. CONTROL AND CONTROL RODS		
121A-1	Rig for Cutting Irradiated EBWR Control Rods Assembled in Fuel Storage Pit	289
121A-2	Radioactivity Measurement of EBWR Irradiated Control Rods	290
121A-3	Hafnium-Zircaloy-2 Transition, No. 9 EBWR Control Rod Showing Spot Weld Failures	292
121A-4	Hafnium-Zircaloy-2 Transition, No. 1 EBWR Control Rod	292
121A-5	2% Boron-Stainless Steel-Zircaloy-2 Transition, No. 4 EBWR Control Rod Showing Rivet Failures	293
121B-1	Tube and Angle Rack in EBWR Fuel Storage Pit Prepared for Control Rod Examination	295
121B-2	Riveted Transition Wedge Test on Original EBWR Control Rods (Transition No. 19; Rods [Nos. 2, 4, 6, and 8] Shown).	296
121B-3	EBWR Control Rod Rivet Head Fractures	298
121B-4	Zircaloy-2 Section (10 in.), No. 4 EBWR Control Rod . .	305
121B-5	Zircaloy-2 Section (10 in.), No. 9 EBWR Control Rod . .	305
121B-6	Relative Location of Control Rod and Fixed Reactor Components for 36-in. Control Rod Withdrawn (4-ft Core)	306
121B-7	Type 304 Stainless Steel Tail Section, No. 2 EBWR Control Rod	306

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
121C-1	Hafnium Section, No. 7 EBWR Control Rod	309
121C-2	Retrieving End, No. 4 EBWR Control Rod	311
121C-3	2% Boron-Stainless Steel Section, No. 2 EBWR Control Rod	312
121C-4	Schematic of Clamping Arrangement Used for Sawing Irradiated EBWR Control Rods	312
121C-5	Radioactivity of Original EBWR Control Rods (Nos. 4 and 9) after 18-Month Decay	315
121C-6	Retrieving End, No. 9 EBWR Control Rod	319
121C-7	Hafnium-Zircaloy-2 Transition, No. 1 EBWR Control Rod	319
121C-8	Positive Action Handling Tool, View at Clamping End . .	320
121C-9	Tube and Angle Rack Assembled in EBWR Fuel Storage Pit for Control Rod Examination	320
121C-10	Hafnium Section, No. 9 EBWR Control Rod	321
121C-11	2% Boron-Stainless Steel Section, No. 4 EBWR Control Rod	321
121C-12	Zircaloy-2 Sections (3 in.), Nos. 4 and 9 EBWR Control Rods	321
121C-13	Portion of 2% Boron-Stainless Steel Section of EBWR Type Control Rod Used to Test Control Rod Drives . . .	322
121E-1	Relative Control Rod Positions in the Core of EBWR and Typical Radial Locations of Samples	325
121G-1	Radioactivity of Original EBWR Control Rods (Nos. 4 and 9) after 18-Month Decay (Location of Test Specimens Shown).	330
121G-2	Bend Test Jig	331

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
121G-3	2 w/o Boron-Stainless Steel Specimen Undergoing Bend Test.	332
121G-4	Static Bend Traces (2 w/o Boron-Stainless Steel) Load Rate - 200-lb Minimum	333
100C-1	Differential Worth of Center Rod (Based on $\beta = 6.27 \times 10^{-3}$)	341
100C-2	Differential and Integral Rod Worth for 9-Rod Bank (Based on $\beta = 6.27 \times 10^{-3}$)	344
100E-1	Differential Worth of Central Control Rod (Rods [Nos. 1-8] Banked at 21.1 in.)	345
100E-2	Derivative of Negative Periods Versus $n_{(t)}$	348
119-1	Core - Fuel Status Board after Fuel Element Removal (Core Load No. 59a, 12/15/61)	357
119-2	Storage Pit - Fuel Status Board during Control Rod Change (Core Load No. 59a, 12/15/61)	358
119-3	Loading of Spikes into Reactor [1/(counts/min) Versus Number of Spikes]	362
100G-1	Worth of Central Rod When Rods (Nos. 1-8) Are at a 16.5-in. Position (above the Bottom of the Core)	366
100G-2	Bank Calibration of EBWR Control Rods	366
100I-1	Calibration of Control Rods (Zircaloy Followers).	369
100I-2	Calibration of Control Rod No. 9 (Rods, Nos. 1-8, at 23.01-25.98-in. Positions)	370
100I-3	For Criticality: $\frac{\delta \text{Position of Rods, Nos. 1-8}}{\delta \text{Position of Rod No. 9}} \simeq -\frac{1}{9}$. .	370

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
Section V. HYDRAULICS AND HEAT TRANSFER		
108A-1	Temperature Distribution in EBWR Core after Non-nuclear Heating	375
108A-2	Thermocouple Temperature Readings	376
108A-3	Temperature Sampling Points.	377
105-1	Location of Probes within Reactor Vessel	386
105-2	Total Recirculation Flow Rate as a Function of Reactor Power (600 psig).	386
105-3	Total Recirculation Flow Rate as a Function of Reactor Power (300 psig).	387
105-4	Comparison of EBWR Recirculation Flow Rate Data (600 psig)	387
105-5	Riser Steam Volume as a Function of Reactor Power (600 psig)	388
105-6	Downcomer Steam Volume Fraction as a Function of Reactor Power (600 psig).	388
105-7	Downcomer Steam Volume Fraction as a Function of Reactor Power (Run D, 600 psig).	389
105-8	Riser Steam Volume Fraction as a Function of Reactor Power (300 psig)	389
105-9	Downcomer Steam Volume Fraction as a Function of Reactor Power (300 psig).	389
105-10	Downcomer Steam Volume Fraction as a Function of Reactor Power (Run D, 300 psig).	390
105-11	Steam Volume Fraction Differential between Riser and Downcomer as a Function of Reactor Power (600 psig).	390
105-12	Actual Interface Level as a Function of Reactor Power [Level (Transducer)]	391
105-13	Differential Height between the Actual Interface Level and the Water Column Level as a Function of Reactor Power	391
105-14	Interface Height Difference between the Center and Periphery of Vessel as a Function of Reactor Power	392

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
105-15	Heat Balance Flow Rates as a Function of Reactor Power (600 psig)	392
105-16	Comparison of Laboratory Correlation with Computed Slip Ratios in Riser	392
105-17	Comparison of Laboratory Correlation with Computed Slip Ratios in Riser	393
105-18	Percentage of Steam Carryunder as a Function of Reactor Power (600 psig).	393
105-19	Percentage of Steam Carryunder as a Function of Reactor Power (300 psig).	393
105-20	Comparison of Reactor Data with Laboratory Correlation.	394
105A-1	Reactor Subcooling as a Function of Reactor Power . . .	395
105A-2	Total Recirculation Flow Rate as a Function of Reactor Power	396
105A-3	Steam Carryunder as a Function of Reactor Power . . .	396
105A-4	Heat Balance Flow Rates as a Function of Reactor Power	397
105A-5	Calculated Steam Volume Fraction in Core as a Function of Reactor Power.	397
105A-6	Height Differential between Actual Interface Level and Water Column Level as a Function of Reactor Power. .	398
105B-1	Total Recirculation Flow Rate as a Function of Reactor Power	400
105B-2	Reactor Subcooling as a Function of Reactor Power (Increase of Subcooling Due to Carryover)	400
105B-3	Percent Carryover as a Function of Reactor Power. Comparison of Three Techniques	401
105B-4	Riser Steam Volume Fraction as a Function of Reactor Power	403
106A-1	EBWR Instrumented Fuel Assembly (Unit II).	405
106A-2	Reactor Subcooling as a Function of Reactor Power . . .	410

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
106A-3	Recirculation Flow Rate as a Function of Reactor Power	411
106A-4	Inlet Velocity as a Function of Reactor Power.	412
106A-5	Exit Liquid Velocity as a Function of Reactor Power . .	413
106B-1	Recirculation Flow Rate as a Function of Reactor Power	415
106B-2	Inlet Velocity as a Function of Reactor Power.	415
106B-3	Exit Liquid Velocity as a Function of Reactor Power . .	416
106C-1	EBWR Instrumented Fuel Assembly (Unit III)	417
106C-2	Reactor Performance as a Function of Reactor Power for Various Steam-Water Interface Levels	421

Section VI. COMPONENTS

102-1	Counts Per Minute Versus Pulse Height Selector	443
102B-1	In-core Chamber Current and $1/\Sigma_{1+2}$ cr Versus Rod Position (Criticals 57-8-2 and 3, April 28, 1960, EBWR 8-Spike Core).	446
102D-1	Meter Reading and Helipot Setting for Trip Versus Current Input.	449
102D-2	Ratio of Peak Signal to d-c Signal to Produce Trip Versus Frequency	450
125-1	Results of EBWR Leakage Rate Test	454
125A-1	Results of EBWR Containment Shell Leak Rate Test. . .	456
116-1	Orifice Differential Versus Flow	459
116-2	Filter Operation	460
116-3	Number of Filters Versus Total Feedwater Flow.	461
109-1	H ₃ BO ₃ Injection - Linear Flux Record.	473
109A-1	Water Conductivity and Log Flux Record of Shutdown from 20 Mw with Soluble Poison	476
120-1	Unborated Cold Feedwater (50 gal) Addition Prior to Startup. Startup Period 15 sec	480

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
120-2	Unborated Cold Feedwater (150 gal) Addition Prior to Startup. Startup Period 28 sec.	481
120-3	Unborated Cold Feedwater (240 gal) Addition Prior to Startup. Startup Period 34 sec.	481
120-4	Unborated Cold Feedwater (250 gal) Addition during 7-Mw Operation	482
120-5	EBWR Startup Transients for Various Startup Periods (No H_3BO_3)	483
120-6	EBWR Startup Transients for Various Startup Periods (3.66 gm/gal H_3BO_3).	483
120-7	Peak Power Versus Reciprocal Startup Period	484
122-1	Reactor Water Level Versus Reactor Power. Effect of Level on Carryover	493
122-2	Effect of Reactor Water Level and Boric Acid on Control Rod Position for Various Power Levels	494
122-3	Primary Flow Versus Reactor Power	494
126-1	EBWR Reboiler System Building Layout, 100 Mw. Location of Radiation Measurement Points	506
126-2	Forced Circulation Nozzle Configuration	507
111-1	Demister Material Test Flow Loop	509
111-2	Demister Material (After Test).	511
111-3	Coarse Mesh Demister Material (After Test)	512

Section VII. APPENDICES

A-1	Relative Location of Fuel Zone and Control Rod Poison Section.	517
B-1	Location of Test Specimens from EBWR Vessel Feedwater Plug	519
B-2	Location of Test Specimens from EBWR Vessel Steam Plug	520
B-3	Location of Test Specimens from Original EBWR Vessel Head Material	521

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
B-4	EBWR Pressure Vessel Showing Locations Where Plugs Were Removed for Installation of a New Steam Outlet Nozzle and an Additional Feedwater Supply Nozzle	522
B-5	Transition Curves (Plots 1 and 2)	523
B-6	Appearance of Charpy V-Notch Impact Specimen Fractures (Above Transition Temperature) Correlated with Specimen and Notch Orientation	526
B-7	Longitudinal Specimens, Radial and Transverse Notches, Vessel and Base Material, Quarter and Mid-Thickness Locations	526
B-8	Transverse Specimens, Longitudinal Notch, Vessel and Base Material, Quarter and Mid-Thickness Locations. .	527
B-9	Radial Specimens, Longitudinal and Transverse Notches, Vessel and Base Material	527
B-10	Transition Curves for EBWR Vessel Wall (Plots 3 and 4)	528
B-11	Typical Load-Strain Trace for EBWR Specimens Showing Mechanics Employed to Obtain Data.	529
C-1	Flow Diagram of EBWR 100-Mw System.	537
C-2	Primary Reboiler.	538
C-3	EBWR Primary Reboiler Tubing (Nitric Acid Tests on 3/4-in.-Type 304 Stainless Tubing)	544
C-4	EBWR Primary Reboiler Tubing (Nitric Acid Tests on 3/4-in.-Type 304 Stainless Tubing)	545
C-5	Reboiler Tubing (About 1 ft from Tube Sheet)	546
C-6	Virgin Metal Tubing As Received	550
C-7	Virgin Metal Tubing (Heat Treatment 0.5 hr - 900°F) . .	551
C-8	Virgin Metal Tubing (Heat Treatment 17.5 hr - 900°F) .	552
C-9	Virgin Metal Tubing (From Middle of Tube Length) As Received.	553
C-10	Virgin Metal Tubing (From Middle of Tube Length) As Received.	554
C-11	Virgin Metal Tubing (From Middle of Tube Length) As Received.	555

LIST OF FIGURES (Cont'd.)

<u>Fig. No.</u>	<u>Title</u>	<u>Page</u>
C-12a	Reboiler No. 1 Tubing - Top of U-Tube - Rolled End - As Received Longitudinal Section	556
C-12b	Reboiler No. 1 Tubing - Top of U-Tube - Rolled End - As Received Transverse Section	557
C-13	Reboiler No. 1 Tubing - 37 in. from Rolled End - Top of U-Tube - As Received	558
C-14	Reboiler No. 1 Tubing - Bottom of U-Tube - Rolled End - As Received	559
C-15	Reboiler No. 1 Tubing - 36 in. from Rolled End - Bottom of U-Tube - As Received	560
C-16	Reboiler No. 1 Tubing - Top of U-Tube - after 48 hr Huey Test (Surfaces Previously Prepared by Manual Mechanical Polishing with Emery Cloth) Longitudinal Section	561
C-17	Reboiler No. 2 Tubing - As Received Typical Longi- tudinal and Transverse Structures	562
D-1	Summary of Test Procedures and Reports (1960)	567
D-2	Summary of Test Procedures and Reports (1961)	568
D-3	Summary of Test Procedures and Reports (1962)	569

NUMERICAL INDEX OF TEST REPORTS

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
100C	Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth	IV	339
100D	Relative Reactivity Measurements of EBWR Fuel Elements	III	194
100E	Calibration of Control Rod No. 9	IV	345
100F	Reactivity Effect of Rotating the Boron- Stainless Steel Strip Containing Spike Elements	III	208
100G	A Preliminary Calibration of the New System of Control Rods in the Cold EBWR	IV	363
100H	Reactivity Aspects of EBWR Core 1B	III	226
100I	Calibration of Control Rod System in EBWR - Empirical Determination of β_{eff}	IV	368
100J	Completion of Loading 1A, with the Addition of Four Peripheral Spikes and One ET Element - Destripping of Eight Spikes by One Strip Each - Verification of the Reactor Meeting the Cold Shutdown Requirements - A Partial Calibration of the Control Rod System	III	215
100K	EBWR Neutron Flux Distribution during Cold Critical Operation	III	235
100M	Study of Operational Data from EBWR at 0-60-Mw Levels - $\Delta k / \Delta P$ Increasing with Power	III	263
100N	Physics Termination Tests on EBWR, Core Loading 1A.	III	220
100P	Flux Distribution Measurements in EBWR During Operation	III	258

NUMERICAL INDEX OF TEST REPORTS (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
101A	EBWR Turbine Seal Leak Tests, Preliminary Results	VI	439
102	Relative Effectiveness of Sb-Be Source and Core-Induced Neutrons.	VI	442
102A	Source Level at Startup Counters and Sensitivity of Counters	VI	444
102B	Instrumentation for Startup	VI	445
102C	Effect of Source Installed in EBWR on November 8, 1960	VI	447
102D	Frequency Response of EBWR High Flux Trip Circuit	VI	448
103	Volume of Water in the EBWR Reactor	VI	437
104	Radiochemical Analysis of Samples from the EBWR Shock Shield	VI	457
105	Preliminary Hydrodynamic Tests on EBWR with Modified Internals.	V	378
105A	Additional Hydrodynamic Test Data from EBWR at Powers to 60 Mw	V	395
105B	An Explanation of the Behavior of EBWR at the Higher Power Levels (> 65 Mw).	V	398
106A	Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly	V	404
106B	Further Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly.	V	414
106C	A Terminal Report on the Hydrodynamic Data Obtained from the EBWR Instrumented Fuel Assemblies	V	416

NUMERICAL INDEX OF TEST REPORTS (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
107	Detection of Possible Growth in Thin Enriched (ET) Fuel Plates of EBWR with High Burnup Exposure	II	127
108A	Temperature Distribution in the EBWR Core .	V	375
108C	Temperature and H ₃ BO ₃ Effect on EBWR Sub- critical Counting Rate	III	163
108D	Experimental EBWR Temperature Coefficients	III	174
109	Test of Control Boric Acid Injection System . .	VI	471
109A	Test of Control Boric Acid Additon to the EBWR during Power Operation	VI	474
110	EBWR Core Loading No. 55.	II	41
110C	EBWR Core Loading No. 56.	II	46
110F	Antimony Source Transfer	II	50
110G	EBWR Core Changes	II	52
110H	EBWR Core Loading No. 57.	II	52
110I	EBWR Core Loading No. 58.	II	60
110J-Add. & Supps. 1,2,& 3	Core Loading for 100-Mw Operation, Core Loading No. 61	II	87
110K	EBWR Fuel	II	67
110L	EBWR Core Loading No. 59.	II	68
110M	Replacement of EBWR Source	II	71
110M-Add.	Reloading of Storage Pit after Cleaning	II	73
110N	Rotation of Spike Fuel Elements.	II	74

NUMERICAL INDEX OF TEST REPORTS (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
110N-Add.	Examination of Irradiated Boron Strips on Spike Fuel Element Frame	II	76
110O	Transposition of Enriched Thin Fuel Elements	II	78
110O-Add.	Examination of 1.1% Boron-Stainless Steel Burnable Poison Strips.	II	108
110P	EBWR Loading 1B (24 Spike): Core Loading No. 60	II	83
110Q	EBWR Core Loading No. 62.	II	96
110R(A)	Core Loading No. 63 (Operation "Strip Tease")	II	100
110R(B)	Destripping of 16 More Spikes for a Gain of ~2% Reactivity - Check of 9-Rod Cold Shut- down Requirement	III	231
111	Short Duration Test of Demister Material . . .	VI	509
111A	Second Test of Demister Material	VI	513
112	Radiochemical Analysis of Crud Removed from the Bottom of EBWR Control Rods No. 4 and No. 9	IV	326
112A	Analysis of Scale from EBWR Fuel Plates . . .	II	123
114	Measurement of Water Channels on EBWR Core 1 Fuel Elements	II	114
114A	Measurement of Water Channels on EBWR Core 1 Fuel Elements	II	118
116	EBWR Feedwater Filters - Cleanliness Determination.	VI	459
117	EBWR Subcooler Cleaning.	VI	462
117A	Cleaning of Vent Condenser.	VI	464
117B	Performance Investigation of EBWR Purifi- cation and Decay Heat Removal Systems	VI	465
118	Reactor Steam Quality by Radioactive Sodium Tracer	V	424
118A	EBWR Steam Quality during High Power Operation.	V	430

NUMERICAL INDEX OF TEST REPORTS (Cont'd.)

<u>Test Report No.</u>	<u>Title</u>	<u>Section</u>	<u>Page</u>
119	EBWR Control Rod Replacement	IV	353
119A	Inspection of the EBWR Control Rod	IV	367
120	Safety Aspects of Operating Borated EBWR . .	VI	478
120A	Analysis of EBWR Startup Transients	III	133
121A	Investigation of Irradiated EBWR Control Rods	IV	289
121B	Investigation of Irradiated EBWR Control Rods	IV	294
121C	Investigation of Irradiated EBWR Control Rods	IV	307
121D	Radiochemical Analysis of Original EBWR Control Rod and Follower Samples	IV	322
121E	Boron Burnup in Original EBWR Control Rod No. 4.	IV	325
121F	Disposition of Original EBWR Control Rods . .	IV	327
121G	Partial Investigation of Irradiated EBWR Control Rods	IV	329
122	Heat Balance Data on EBWR 100-Mw Plant . .	VI	487
123, 123A, 123B, 123C	EBWR Reactor Transfer Function Measure- ments	III	268
124	Void Coefficient Measurements in EBWR. . . .	III	187
125	EBWR Building Shell Leak Rate Test (of October 1960).	VI	453
125A	EBWR Building Shell Leak Rate Test (of August 1962).	VI	455
126	Radiation Levels in Reboiler Building and Sub-Reactor Room.	VI	504

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I. INTRODUCTION

E. A. Wimunc

This collection of test reports represents the results of various experiments and tests conducted on the Experimental Boiling Water Reactor (EBWR) at Argonne National Laboratory since July, 1959. The results of tests performed prior to the July, 1959 date were published in ANL-6229, November, 1960.

This publication encompasses most of the results of those experiments associated with the facility since it was modified to dissipate 100 Mw of thermal energy. The reports included here were selected on the basis of general value to those interested in nuclear science and engineering.

Many of these reports were originally prepared for the sole purpose of distributing the data and results promptly on problems of immediate interest to Laboratory personnel and as such they do not constitute finished documents; however, they are valuable as a continuing record of experiments and tests of the reactor complex.

It should also 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 individual test report author(s) and do not necessarily reflect those of the Laboratory. Some discrepancies will be noted in results on the same item as reported by different authors. This can be attributed to the method used in making the determination. As an example, carryover was determined by several methods, i.e., carryover of sodium, steam sampling nozzles, and heat balance and mass transport of fluid. Nevertheless, there is agreement in the trend even though the results do not exactly agree.

The period covered by this report is from July, 1959 to December 6, 1962; the latter being the shutdown date and the termination of the water program at the Laboratory. Reactor and generator output for the two phases is tabulated below.

<u>Accumulated:</u>	<u>Total Up To:</u>	
	<u>7/3/59</u>	<u>12/6/62</u>
Generator Hours	8,745	10,603
Generator Mw Hours	35,006	39,696
Reactor Hours	11,225	15,665
Reactor Mw Hours	211,509	294,028

The final core loading used in reaching the 100-Mw(t) output was as follows:

- 52 Enriched (1.44%) Thick Plate Elements
- 54 Enriched (1.44%) Thin Plate Elements
- 9 Natural Thick and Thin Elements
- 32 Enriched Spike Elements

Final arrangement of the above elements, control rods, and the source in the reactor are shown on the fuel status board for core loading No. 63 (Rev). Various core changes made prior to the final loading are shown on the fuel status boards for core loadings Nos. 55 to 63 (Rev). Significant activities and operating levels for EBWR during the period 1960-1962 are shown in Figs. 1, 2, and 3.

Fig. 1. EBWR Operating Levels and Significant Events (1960)

Fig. 2. EBWR Operating Levels and Significant Events (1961)

Fig. 3. EBWR Operating Levels and Significant Events (1962)

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Section II. FUEL

<u>Title</u>	<u>Page</u>
EBWR Core Loading No. 55 (Test Report No. 110, dated March 10, 1960, by V. M. Kolba)	41
EBWR Core Loading No. 56 (Test Report No. 110C, dated April 22, 1960, by V. M. Kolba)	46
Antimony Source Transfer (Test Report No. 110F, dated November 10, 1960, by V. M. Kolba)	50
EBWR Core Changes (Test Report No. 110G, dated November 30, 1960, by V. M. Kolba)	52
EBWR Core Loading No. 57 (Test Report No. 110H, dated January 18, 1961, by V. M. Kolba)	52
EBWR Core Loading No. 58 (Test Report No. 110I, dated January 18, 1961, by V. M. Kolba)	60
EBWR Fuel (Test Report No. 110K, dated April 17, 1961, by V. M. Kolba)	67
EBWR Core Loading No. 59 (Test Report No. 110L, dated June 12, 1961, by V. M. Kolba)	68
Replacement of EBWR Source (Test Report No. 110M, dated October 23, 1961, by V. M. Kolba)	71
Reloading of Storage Pit after Cleaning (Test Report No. 110M Addendum, dated November 14, 1961, by V. M. Kolba).	73
Rotation of Spike Fuel Elements (Test Report No. 110N, dated November 22, 1961, by E. L. Martinec).	74
Examination of Irradiated Boron Strips on Spike Fuel Element Frame (Test Report No. 110N Addendum, dated November 22, 1961, by V. M. Kolba)	76
Transposition of Enriched Thin Fuel Elements (Test Report No. 110O, dated January 10, 1962, by V. M. Kolba).	78
EBWR Loading 1B (24 Spike): Core Loading No. 60 (Test Report No. 110P, dated February 2, 1962, by E. L. Martinec)	83

Section II. FUEL (Cont'd.)

<u>Title</u>	<u>Page</u>
Core Loading for 100-Mw Operation, Core Loading No. 61 (Test Report No. 110J Addendum and Supplements 1, 2, and 3, dated March 2, 1962, by V. M. Kolba).	87
EBWR Core Loading No. 62 (Test Report No. 110Q, dated May 24, 1962, by V. M. Kolba)	96
Core Loading No. 63 (Operation "Strip-Tease"), (Test Report No. 110R(A), dated October 2, 1962, by V. M. Kolba).	100
Examination of 1.1% Boron-Stainless Steel Burnable Poison Strips (Test Report No. 110O Addendum, dated November 7, 1962, by V. M. Kolba)	108
Measurement of Water Channels on EBWR Core 1 Fuel Elements (Test Report No. 114, dated September 22, 1961, by C. F. Reinke and W. N. Beck).	114
Measurement of Water Channels on EBWR Core 1 Fuel Elements (Test Report No. 114A, dated January 25, 1963, by C. F. Reinke, W. N. Beck, and W. A. Ahrens)	118
Analysis of Scale from EBWR Fuel Plates (Test Report No. 112A, dated January 21, 1963, by W. G. Knapp).	123
Detection of Possible Growth in Thin Enriched (ET) Fuel Plates of EBWR with High Burnup Exposure (Test Report No. 107, dated January 31, 1963, by H. Iskenderian and J. Haugsnes).	127

EBWR CORE LOADING NO. 55

(Test Report No. 110, dated March 10, 1960, by V. M. Kolba)

Loading of the EBWR core was started 2/29/60. Unloading of the storage pit and loading of the core was essentially as described in the "EBWR Core Loading Plan," (March, 1960). Several innovations were made in the loading operation to permit safe and sure latching of the fuel elements, and incidentally, a more rapid loading rate. Attempts to use the loading slot and the new fuel holder were unsuccessful. Thus, it was decided to move the fuel directly from the storage pit to the core using the coffin. This was done, with a minimum of hazard to personnel, by the following means:

1. The storage pit water level was raised.
2. The coffin was lowered into the water as far as possible.
3. The coffin was positioned properly in the storage pit, for the raising operation, to shield the operator.
4. All personnel, other than the operator and a Health Physics monitor, were directed to leave the storage pit area for shielded areas.

This procedure proved highly successful even with the "hottest" fuel element in the core at that time.

At the start of the loading operation, some trouble was experienced with the fuel handling tool in the coffin. Tool heads were changed and, during the balance of the loading, the problem did not recur. Some difficulty was also experienced with the cable rubbing the guide plate on the coffin head. This was alleviated when the slots were ground down to accommodate the cables.

Dummy elements were loaded before the fuel elements for expediency, however; because of this, one of the ears of a dummy element was bent while positioning a fuel element next to it. The dummy elements, being lighter than the fuel elements, do not seat or stand as stably as the fuel elements and thus are more prone to shifting and misalignment.

During core loading three shifts were used as follows:

a.	Shift No. 1	24:00 — 08:30
b.	Shift No. 2	08:00 — 16:30
c.	Shift No. 3	16:00 — 00:30

Shift personnel included:

- a. A Co-ordinator
- b. A Chief Physicist
- c. A Chief Loader
- d. A Fuel Data Checker
- e. Reactor Instrumentation Personnel

Loading moves were made as listed in Table 110-I. Some fuel elements would not seat entirely when lowered with the coffin tool. Therefore, after removing the coffin tool and coffin, these elements were seated using the rigid tool.

Table 110-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(2/29/60-3/3/60)

Date	Move	Fuel Element No.	From - Location	Time	To - Location	Time	Remarks
2/29/60	1	H-10	S.P.* M-5	22:00	C.I.† 0-301	22:35	Could not pick up from loading slot
		S.D.S.	North of		S.P. M-1	23:45	
		Be Target	G-11				
3/1/60	2	H-2	K-11	00:50	C.I. 0-323	01:05	
	3	S.D.S.					
		Be Target	↓ M-1	01:20		0-311	
	4	S.D.	New			0-357	
	5					0-371	
	6					0-384	
	7					0-395	
	8					0-407	
	9					0-417	
	10					0-6	
	11					0-16	
	12					0-28	
	13					0-39	
	14					0-53	
	15					0-67	

* S.P. - Storage Pit

† C.I. - Core Index

Table 110-I (Cont'd.)

Date	Move	Fuel Element No.	From - Location	Time	To - Location	Time	Remarks
3/1/60	58	EH-28	S.P. F-11	22:59	C.I. 44-220	23:11	
	59	EH-40	D-11	23:20	23-188	23:33	
3/2/60	60	EH-27	B-10	00:15	23-172	00:30	
	61	EH-22	C-2	00:40	44-181	00:50	
	62	EH-15	B-2	00:55	63-206	01:15	
	63	ET-10	E-9	01:30	72-202	01:40	
	64	EH-50	D-2	01:45	54-179	02:00	
	65	H-8	M-6	02:08	27-149	02:20	
	66	EH-1	M-2	02:30	34-145	02:40	
	67	EH-19	L-2	02:47	59-170	02:58	
	68	ET-49	F-9	03:05	77-191	03:17	
	69	ET-8	G-9	03:50	77-170	04:00	
	70	EH-47	L-10	04:05	59-153	04:15	
	71	EH-39	K-2	04:18	34-131	04:38	
	72	H-13	M-7	04:40	27-113	04:50	
	73	EH-32	K-10	05:00	54-135	05:13	
	74	ET-12	H-9	05:22	72-149	05:32	
	75	ET-37	I-9	05:40	63-126	05:50	
	76	EH-12	D-10	06:01	44-114	06:08	
	77	H-15	L-5	06:15	23-82	06:25	
	78	EH-13	C-10	06:30	47-97	06:40	
	79	EH-49	B-9	06:45	23-66	06:55	
	80	EH-36	B-7	07:05	44-75	07:15	
	81	EH-17	B-8	07:20	63-101	07:35	
	82	ET-28	D-4	07:40	72-96	07:55	
	83	EH-45	B-6	08:00	54-73	08:10	
	84	H-9	D-1	08:18	27-43	08:29	
	85	EH-29	B-4	08:36	34-39	08:48	
	86	EH-38	B-5	08:57	59-64	09:10	
	87	ET-34	C-4	09:17	77-85	09:29	
	88	ET-14	J-9	09:36	77-64	09:46	
	89	EH-4	E-10	09:53	59-48	10:05	
	90	EH-24	C-3	10:13	34-25	10:23	
	91	EH-26	B-3	10:31	27-7	11:05	Bent ears on dummy 0-407 replaced with dummy without locking mechanism
	92	EH-46	F-10	11:12	54-29	11:26	
	93	ET-6	D-3	11:34	72-43	11:45	
	94	ET-19	E-3	11:52	63-20	12:03	
	95	EH-33	G-10	12:09	44-9	12:20	
				12:25		13:10	Fixed tool wire guide on coffin head
↓	96	EH-3	↓ G-2	13:13	↓ 23-399	13:27	

Table 110-I (Cont'd.)

Date	Move	Fuel Element No.	From - Location	Time	To - Location	Time	Remarks
3/2/60	97	EH-9	S.P. F-2	13:34	C.I. 23-383	13:46	Inserted Be block at 0-122
	98	EH-43	E-2	13:52	47-414	14:03	
	99	EH-6	M-3	14:10	44-392	14:22	
	100	EH-18	L-3	14:28	63-418	14:41	
	101	ET-45	F-3	14:47	72-414	15:00	
	102	EH-42	K-3	15:06	54-390	15:18	
	103	H-3	E-1	15:24	27-361	15:34	
	104	EH-5	M-9	15:42	34-356	15:55	
	105	EH-25	L-9	16:02	59-381	16:12	
	106	ET-38	G-3	16:20	77-402	16:30	
	107	ET-22	E-5	16:48	95-400	17:00	
	108	ET-11	J-8	17:11	89-370	17:22	
	109	ET-44	I-8	17:35	80-342	17:47	
	110	ET-29	H-8	17:55	89-338	18:09	
	111	ET-23	F-7	18:22	95-324	18:33	
	112	ET-27	F-5	18:45	95-294	18:56	
	113	ET-39	G-4	19:04	89-264	19:27	
	114	ET-31	F-4	19:40	80-237	19:49	
	115	ET-47	E-4	19:59	89-233	20:10	
	116	ET-13	C-7	20:21	95-219	20:31	
	117	ET-26	C-5	21:22	95-188	21:31	
	118	ET-53	G-8	21:44	89-158	21:54	
	119	ET-4	F-8	22:05	80-131	22:15	
	120	ET-2	E-8	22:27	89-127	22:40	
	121	ET-3	D-7	22:51	95-113	23:02	
	122	ET-1	D-5	23:13	95-83	23:25	
3/2-3/60	123	ET-41	J-4	23:34	89-53	23:45	
	124	ET-7	I-4	23:54	80-25	00:03	
3/3/60	125	ET-24	H-4	00:15	89-21	00:25	Moved counters up in cans
	126	ET-16	E-7	00:30	95-7	00:45	
	127	ET-33	H-5	00:50	109-42	01:03	
	128	Sb Rod in S.D.	Y L-11	—	0-289	—	
	128a	Sb	C.I. 0-289	—	0-311	—	
	129	ET-50	S.P. H-6	01:25	109-417	01:55	
	130	ET-25	E-6	02:05	101-371	02:12	
	131	ET-52	I-7	02:20	109-359	02:30	
	132	ET-21	I-5	02:35	109-311	02:45	
	133	ET-48	F-6	02:53	101-266	03:02	
	134	ET-55	G-7	03:10	109-254	03:20	
	135	ET-42	G-5	03:28	109-205	03:40	
	136	ET-15	C-6	04:20	101-160	04:25	
	137	ET-17	G-6	04:35	109-148	04:40	

Table 110-I (Cont'd.)

Date	Move	Fuel Element No.	From - Location	Time	To - Location	Time	Remarks
3/3/60 ↓ ↓	138	ET-56	S.P. H-7	04:48	C.I. 109-99	05:05	
	139	ET-43	↓ D-6	05:10	101-54	05:20	
	140	ET-54	↓ J-5	05:28	129-159	05:40	
	141	ET-18	↓ J-6	05:45	129-53	05:55	
	142	ET-5	↓ I-6	06:00	129-370	06:10	
	143	ET-46	↓ J-7	06:15	129-264	06:30	
	144	EH-44	↓ C-11	06:50	47-203	07:00	
	145	S.D.	C.I. 0-53		0-265		
	146	↓	↓ 0-289		S.P. A-11		
	147	↓	New		C.I. 0-289		
	148	↓	S.P. A-10		0-278		
	149	↓	↓ I-1		0-53		
	150	↓	New		0-333		
	151	↓	New		↓ 0-346		
							Bent spring repaired
							Sb Transfer Dummy
							Corrosion samples

EBWR CORE LOADING NO. 56

(Test Report No. 110C, dated April 22, 1960, by V. M. Kolba)

In preparation for fuel movements and criticals, the stainless steel dummies from core index positions 0-90 and 0-112 were transferred to the storage pit. Counter tubes and counters were then inserted in these core index positions.

Transfer of the stainless steel dummies from the reactor to the storage pit was made, in air, using the rigid tool and the auxiliary winch on the crane. Fuel movement in the core was accomplished using the "A" frame mounted on the indexing plug and the rigid tool. The new spike fuel was inserted in the core using the new rigid tool and the auxiliary winch on the crane. All new fuel was handled with clean white gloves prior to placement in the reactor.

To accomplish the core changes, the handling moves were made as listed in Table 110C-I. The fuel status boards for the core and storage pit, for core loading No. 56, are shown in Figs. 110C-1 and 2.

Table 110C-1

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(4/12/60)

Time	Move*	Fuel Element No.	From -- Location		To -- Location	
09:10	1	S.D.	C.I.	0-159	S.P.	E-1
09:18	2			0-265		F-1
09:26	3			0-371		H-1
09:35	4			0-39		G-1
		↓			↓	
09:44	5	EH-15		63-206	C.I.	0-159
09:48	6	ET-5	↓	129-370		63-206
10:02	7	S-1		New		129-370
11:13	8	EH-21	C.I.	63-312		0-265
11:20	9	ET-18	↓	129-53		63-312
11:32	10	S-2		New		129-53
12:26	11	EH-17	C.I.	63-101		0-39
12:30	12	ET-46	↓	129-264		63-101
12:52	13	S-3		New		129-264
13:47	14	EH-18	C.I.	63-418		0-371
13:52	15	ET-54	↓	129-159		63-418
14:03	16	S-4		New	↓	129-159

* Following Moves 7, 10, 13, and 16 counting rates were obtained and Criticals 56-1, 56-2, 56-3, and 56-4 were run.

Fig. 110C-1. Core - Fuel Status Board (Core Load No. 56, 4/12/60)

Fig. 110C-2. Storage Pit - Fuel Status Board (Core Load No. 56, 4/12/60)

ANTIMONY SOURCE TRANSFER

(Test Report No. 110F, dated November 10, 1960, by V. M. Kolba)

The moves listed in Table 110F-I were made to exchange the old antimony source (1A) for a new irradiated antimony source (1B).

Table 110F-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(11/7 and 11/8/60)

Date	Move	Fuel Element No.	From - Location	To - Location	Time
11/7/60 ↓	1	S.D.	C.I. 0-371	S.P. 1-7	10:00
	2	S.T.S.D. *	S.P. A-1	S.P. E-9	
	3 } 4 }	Instrument Tubes	{ S.P. S.P.	C.I. 0-395 C.I. 0-371	
11/8/60 ↓	1	S.T.S.D.	S.P. E-9	C.I. 0-346	12:45
	2	Sb Source (1A)	C.I. 0-311	C.I. 0-346	13:00
	3	S.T.S.D.	C.I. 0-346	S.P. F-11	13:20
	4	Sb Source (1A)	S.P. F-11	S.P. G-12	13:30
	5	New Sb Source (1B)	CP-5† via Coffin; Loading Chute	S.P. F-11	13:45
	6	S.T.S.D.	S.P. F-11	C.I. 0-346	14:50
	7	New Sb Source (1B)	C.I. 0-346	C.I. 0-311	15:05
	8	S.T.S.D.	C.I. 0-346	S.P. A-1	15:40
	9 } 10 }	Instrument Tubes	{ C.I. 0-371 C.I. 0-395		

* Source transfer stainless dummy.

† The new Sb source was irradiated at CP-5 in location VT-22 from 07:00 on 4/26/60 to 08:30 on 11/7/60. The total irradiation was as follows:

	kw-hr	Power Level, kw
	6,816,032	2200
	3,625,066	4600
Total	10,441,098	

The flux at the 2200 kw power level was 2×10^{13} in the VT-22 position.

The newly irradiated Sb source (1B) was placed in a coffin and transferred from CP-5 to EBWR on 11/8/60. Using the overhead crane, the coffin was

placed above the loading slot in the storage pit. The source (1B) was then lowered from the coffin into the storage pit loading slot using the cable attached to the source. This cable had been attached to the source prior to its irradiation in CP-5. The cable was then secured and the transfer coffin removed from the loading slot area. Moves 1 to 3 on 11/8/60 were accomplished with no problem. During Move 4, just after the source was in the confines of the old Be target, stored in the G-12 location, the source dropped from the tool without intended disengagement.

After the new source (1B) was placed in the transfer dummy, Move 5, the cutting of the cable proceeded. This was accomplished remotely by a pincer-type cut-off tool. The cable was cut through the loop below the "stake." When the cable was removed only the "stake" portion came off, leaving the possibility that a portion of the loop remained in the hole in the knob of the source. Attempts were made to grasp the remaining cable; however, in 15-20 ft of water, 1/16 in. thick wire is difficult to see. In an attempt to obtain a better viewing position, the new source (1B) was grasped by the source handling tool and raised out of the transfer dummy. When the source cleared the dummy it dropped from the tool (~4 ft), fortuitously into the box, at position G-12, outside the storage rack. After successfully latching the source it was again placed in the transfer dummy at F-11. All subsequent moves, 6 to 8, were uneventful and proceeded in good order. It has not been determined if the wire is still on the source or if it fell out during the handling of the source. While handling the source, continual radiation monitoring was conducted.

When the old source (1A) was transferred from the core into the coffin the dose rate at a point ~1-1/2 ft from the source was 17 r/hr. The new source, (1B) while being loaded into the reactor from the coffin, had a dose rate of 500 r/hr at approximately the same location as the previous measurement. The above measurements were made with the probe lying on the core indexing plug at the base of the coffin.

With the new source (1B) in the fuel transfer coffin, the maximum dose rate at the external surface of the coffin was 14-16 mr/hr.

EBWR CORE CHANGES

(Test Report No. 100G, dated November 30, 1960, by V. M. Kolba)

To prepare the core for an experimental test run with 20 spike elements, core changes were made as listed in Table 110G-I.

Table 110 G-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(11/23/60)

Move	Fuel Element No.	From - Location		To - Location	
1	S.D.	S.P.	I-7	C.I.	0-371
2	↓	↓	H-11	↓	0-346
3			G-5		0-395
4			H-4		0-16
5	↓	↓	H-5	↓	0-417

The transfers above were made in air, using the fuel element handling tool and the auxiliary winch on the overhead crane. Appropriate radiation monitoring was maintained during these transfers.

EBWR CORE LOADING NO. 57

(Test Report No. 110H, dated January 18, 1961, by V. M. Kolba)

On opening the reactor, subsequent to performing cold and hot criticals on core loading No. 56, it was found that three of the four spikes, S-1, S-2, and S-4 were ~ 5-3/4 in. higher than the fourth spike S-3. This was attributed to either "hand up" on the core grid plate by the nose of the fuel elements. Measurements were taken from the grid plate to the top of the indexing plug. The new rigid tool was then inserted in a spare fuel frame and the measurement transposed and marked on the tool handle as a check point to assure the proper seating of spikes in the future.

To accomplish the core changes for the performance of power tests, with a few spikes, fuel element moves were made as shown in Table 110H-I.

Spikes were transferred from the reactor to the storage pit using the coffin. These new spike fuel element frames, while theoretically capable of fitting the storage rack, did not fit in all locations in the storage rack. This may have been due to gross fabrication tolerances in the storage rack, weld splatter, excessive weld bead, or a combination of the above.

New fuel frames, containing three 1.1% boron stripes tack welded to the lower end fitting, were placed in the storage rack using the rigid tool. All handling of the new frames was done with clean white gloves. Prior to being lowered into the storage pit, the boron plates were placed on the outside of the frame and held against the frame by a line.

Table 110H-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(4/19-21/60 and 4/25-28/60)

Move	Date	Time	Fuel Element No.	From - Location		To Location	Boron Location N↑
1	4/19/60	14:27	S-1	C.I.	129-370	S.P.	1-5
2		15:19	S-2	↓	129-53	↓	G-8
3		—	New Frame	—	—	↓	1-3
4		—	New Frame	—	—	↓	F-7
5		16:20	S-1	S.P.	1-5	↓	1-3
6		16:50	S-2	↓	G-8	↓	F-7
1	4/20/60	09:15	ET-5	C.I.	63-206	C.I.	129-370
2		09:25	EH-15	↓	0-159	↓	63-206
3		09:34	ET-18	↓	63-312	↓	129-53
4		09:41	EH-21	↓	0-265	↓	63-312
5		09:48	S.D.	↓	0-251	↓	0-265
6		09:52	EH-7	↓	23-278	↓	0-251
7		09:58	EH-10	↓	44-287	↓	23-278
8		10:04	EH-21	↓	63-312	↓	44-287
9		10:24	S.D.	↓	0-217	↓	0-159
10		10:31	T-13	↓	27-255	↓	0-217
11		10:45	EH-34	↓	54-284	↓	27-255
12		10:49	ET-40	↓	77-276	↓	54-284
13		11:36	S-1	S.P.	1-3	↓	63-312
14		11:56	S-3	C.I.	129-264	S.P.	1-3
15		13:20	S-2	S.P.	F-7	C.I.	77-276
16		13:46	S-4	C.I.	129-159	S.P.	F-7
17		14:45	ET-46	↓	63-101	C.I.	129-264
18		14:58	EH-17	↓	0-39	↓	63-101
19		15:05	ET-54	↓	63-418	↓	129-159
20		15:11	EH-18	↓	0-371	↓	63-418
21		16:05	S.D.	↓	0-206	↓	0-371
22		16:12	S.D.	↓	0-227	↓	0-39

Table 110H-I (Cont'd.)

Move	Date	Time	Fuel Element No.	From - Location		To - Location		Boron Location N↑
1	4/21/60	—	New Frame	—	—	S.P.	E-9	
2	↓	—	New Frame	—	—	↓	K-3	
3		09:35	S-4	S.P.	F-7	↓	E-9	
4		10:00	S-3	S.P.	I-3	↓	K-3	
5		11:03	S.D.	C.I.	0-240	C.I.	0-112	
6		11:08	EH-16	↓	34-250	↓	0-240	
7		11:14	EH-11	↓	59-276	↓	34-250	
8		11:18	ET-9	↓	77-297	↓	59-276	
9		11:22	S.D.	↓	0-196	↓	0-90	
10		11:27	H-11	↓	27-219	↓	0-196	
11		11:31	EH-2	↓	59-259	↓	27-219	
12		11:35	ET-40	↓	54-284	↓	59-259	
13		11:40	ET-32	↓	72-308	↓	54-284	
14		12:21	S-3	S.P.	K-3	↓	77-297	
15	↓	14:45	S-4	S.P.	E-9	↓	72-308	
1	4/25/60	11:12	S.D.	C.I.	0-16	S.P.	H-2	
2	↓	11:18	S.D.	↓	0-417	S.P.	H-3	
3		12:59	S.D.	↓	0-39	C.I.	0-206	
4		13:02	S.D.	↓	0-6	↓	0-227	
5		14:26	H-9	↓	27-43	↓	0-6	
6		14:33	EH-45	↓	54-73	↓	27-43	
7		14:35	ET-28	↓	72-96	↓	54-73	
8		14:58	S-11	↓	New	C.I.	72-96	
9		15:21	EH-49	C.I.	23-66	↓	0-39	
10		15:25	EH-36	↓	44-75	↓	23-66	
11		15:28	EH-17	↓	63-101	↓	44-75	
12	↓	15:38	S-12	↓	New	↓	63-101	
1	4/26/60	09:04	S.D.	C.I.	0-28	S.P.	H-4	
2	↓	09:13	S.D.	↓	0-407	↓	H-5	
3		09:20	S.D.	↓	0-384	↓	H-6	
4		09:27	S.D.	↓	0-395	↓	H-7	
5		10:18	EH-29	↓	34-39	C.I.	0-28	
6		10:22	EH-38	↓	59-64	C.I.	34-39	
7		10:24	ET-34	↓	59-64	C.I.	59-64	
8		10:45	S-13	↓	New	↓	77-85	
9		12:55	EH-26	C.I.	27-7	↓	0-407	
10		13:02	EH-4	↓	59-48	↓	27-7	
11		13:05	ET-14	↓	77-64	↓	59-48	
12	↓	14:20	S-14	↓	New	↓	77-64	
1	4/27/60	13:18	S-1	Rotated 180° and reinserted		63-312		
2	↓	14:42	S-3	Rotated 180° and reinserted		77-297		
3		14:56	S-12	C.I.	63-101	C.I.	0-384	
4		15:16	S-2	↓	77-276	↓	63-101	
5	↓	15:34	S-13	↓	77-85	↓	77-276	

Table 110H-I (Cont'd.)

Move	Date	Time	Fuel Element No.	From - Location		To - Location		Boron Location N
6	4/27/60	15:45	S-4	C.I.	72-308	C.I.	77-85	<div style="border: 1px solid black; width: 50px; height: 15px; margin: 0 auto;"></div>
7	↓	15:55	S-12	↓	0-384	↓	72-308	
1	4/28/60	15:15	S.D.	S.P.	H-4	C.I.	0-16	
2	↓	15:00	S.D.	↓	H-5	↓	0-417	
3	↓	14:40	S.D.	↓	H-6	↓	0-384	
4	↓	14:55	S.D.	↓	H-7	↓	0-395	

The fuel bundle was removed from the old frame, using a special tool attached to the auxiliary winch on the crane, and placed in the new frame containing the three boron-stainless steel strips. The loaded frame was then lifted out of the storage rack, using the new rigid tool and the auxiliary winch on the crane, raised to a position ~ 1 in. below the water level of the storage pit, and moved to the southeast corner of the storage pit. A boron strip was then bowed out, using a special tool, and snapped into place between the upper end fitting of the frame and the end fitting of the fuel bundle. When all three boron strips were in place, the spike fuel element was again lowered to the storage rack and the rigid tool disengaged. The spike fuel element was then returned to the core from the storage pit by means of the coffin.

Fuel movements in the reactor were made using the "A" frame on the indexing plug and the old rigid tool.

Fuel movements in the core and in the storage pit were monitored by counters located in the pit and in the reactor. Appropriate Health Physics monitoring was also conducted during all fuel element transfers. Shoe covers were worn by personnel while within the reactor-storage pit areas. All applicable fuel handling, operation, and safety procedures were used. All nine control rods were banked at 6 in. when any fuel movements were made in the core.

After Move 22, 4/20/60, the in-core instrument tubes were moved from locations 0-90 and 0-112 to 0-206 and 0-227. Critical 57-2-1 was then run. Spikes S-1, S-2, S-3, and S-4 had three boron-stainless steel strips per frame. With the addition of spike S-4 to the core, counts were taken at 3-in. intervals while lowering the element into the core.

After Move 13, 4/21/60, Critical 57-4-1 was run. In this instance, the open or non-boron face was located, using compass directions, as follows:

S-1, 270 deg; S-2, 90 deg; and S-4, 90 deg. On 4/22/60 the four spikes in the core were rotated so that the open, non-boron sides, were all at 180 deg, i.e., open to the center of the core. After rotation of these spikes, Critical 57-4-2 was run. During this Critical the neutron flux in the core was measured using gold foils. A plot of data secured is shown in Figs. 110H-1 and 2.

Subsequent loading of spike fuel elements was made in 3-in. increments while a careful check of the counters was maintained. A one-minute base count and a one-minute count after complete insertion were made. After loading S-11, 4/25/60, rods Nos. 7 and 8 were withdrawn to 47 in. to check if the reactor was critical. Spikes S-11 and S-12 had two boron-stainless steel strips per frame, 180 deg apart. These were located in the core such that the non-boron sides of the spike were at 90 deg and 270 deg. After Move 12, 4/25/60, Critical 57-6-1 was run.

After Move 8, 4/26/60, rod No. 7 was withdrawn to 47 in. Rod No. 8 was then withdrawn and, with rods Nos. 1, 2, 3, 4, 5, 6, and 9 banked at 4 in., the core became critical when rod No. 8 was at 41 in.

Two boron strips were attached to spikes S-13 and S-14. These spikes were oriented so that the non-boron sides of the spikes were at 90 deg and 270 deg. After Move 12, 4/26/60, Critical 57-8-1 was run. After running the Critical, rod No. 7 was pulled to 47 in., with the balance of the rods at a 6-in. bank. Rods Nos. 1, 2, 3, 4, 5, 6, and 9 were set, as a bank, at 4 in. and rod No. 8 was then withdrawn. The core was critical with rod No. 8 at 30 in.

Difficulty was experienced in seating S-14. A new frame with two boron strips was obtained and S-14 inserted in the new frame. The element was then re-inserted in the core and difficulty was again experienced in seating. The adjacent element ET-6 at 72-43 was raised ~6 in. and S-14 was seated. ET-6 was then lowered back into place.

On April 27, 1960, spikes in the core were interchanged and rotated to present the most effective use of the boron strips. After Move 7, 4/27/60, Critical 57-8-2 was run.

On April 28, 1960, the core was prepared for a power run. The in-core instrument tubes were removed and stainless steel dummies were placed in the periphery of the core to fill the vacant spots. The fuel status boards for the core and storage pit, for core load No. 57, are shown in Figs. 110H-3 and 4.

Fig. 110H-1. Thermal Flux in
EBWR Elements
(4/22/60)

Fig. 110H-2. Thermal Flux in
EBWR (4/22/60)

Fig. 110H-3. Core - Fuel Status Board (Core Load No. 57, 4/28/60)

NORTH

	A	B	C	D	E	F	G	H	I	J	K	L	
							S D S						
11		** H 7											11
10													10
9													9
8													8
7	4-4-60 S.D. 0-112										T 15		7
6	4-4-60 S.D. 0-90										T 7	T 21	6
5											T 9		5
4													4
3								4-25-60 S.D. 0-417					3
2								4-25-60 S.D. 0-16					2
1	S.D. SD TRANS- FER DUMMY				4-12-60 S.D. 0-159	4-12-60 S.D. 0-265	4-12-60 S.D. 0-39	4-12-60 S.D. 0-371		ACF			1
	A	B	C	D	E	F	G	H	I	J	K	L	M

T-23
5 PLATES

DATE: 4-28-60

LOAD NO.: STORAGE

PREFIX LETTER CODE

H-NATURAL "U" THICK PLATES
T-NATURAL "U" THIN PLATES
EH-ENRICHED "U" THICK PLATES
ET-ENRICHED "U" THIN PLATES
SD-DUMMY STAINLESS
SDS-DUMMY W/SOURCE
S-ENRICHED "U" RODS (SPIKES)

* DUMMY HAS AN INSTRUMENTATION
CHANNEL ATTACHED EXTERNALLY AND
WILL NOT FIT INTO STORAGE RACK.

** TOP OF FUEL ELEMENT BROKEN OFF,
ALSO BENT LIFTING TABS.

EBWR STORAGE RACK FUEL STATUS BOARD

Fig. 110H-4. Storage Pit - Fuel Status Board (Core Load No. 57, 4/28/60)

EBWR CORE LOADING NO. 58

(Test Report No. 110I, dated January 18, 1961, by V. M. Kolba)

From April 29 to May 8, 1960, time was devoted to preparation for a power run. The spikes, after the power run of 887.7 Mw-hr, were a light straw color.

Subsequent to the power run, in-core instrumentation was placed at locations 0-16 and 0-417 and loading of spikes was started again on May 9, 1960. The loading moves are listed in Table 110I-I. As before, after taking a one-minute base count, each spike was lowered in 3-in. increments and, when seated, another one-minute count was taken to check the increase in rate. After each spike was loaded, the control rods in the immediate area, and other sensitive rods, were checked, by withdrawal, to determine if the core was divergent with one rod removed to 47 in. while the balance of the rods were at a 6-in. bank.

Spikes loaded subsequent to May 9, 1960, were so oriented that the boron strips were in their most effective position. Spikes S-5, S-9, S-10, and S-27 contained two boron strips per spike. After Move 20, 5/9/60, Critical 57-12-1 was run.

On May 10, 1960, four spikes, S-6, S-7, S-8, and S-15, were loaded. These spikes had two boron strips per spike. After Move 18, it was determined that the core was divergent with rod No. 5 partially withdrawn and the balance of the rods at a 6-in. bank. Because of this, loading on the east side of the core was discontinued and spike S-15 was placed on the north side of the core. After Move 23, 5/10/60, Critical 57-16-1 was run.

Because one rod could not be fully withdrawn and have the core remain sub-critical, spike S-8 was removed from its position on the east side and moved to the south side of the core on May 11, 1960.

Spike S-8 was brought up out of the core and reactor to check the activity. At 11:45 on May 11, 1960, the reading was 600 mr/hr at ~ 2 in., after the critical run on May 10, 1960 at 16:30.

After Move 5, 5/10/60, Critical 57-16-2 was run. Four frames previously used for spikes S-1 to S-4 were removed from the storage pit, checked for activity (~15 mr/hr on the nose pieces), wrapped in polyethylene and sent to Building D-17.

Table 110I-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(5/9-13/60)

Move	Date	Time	Fuel Element No.	From - Location		To - Location		Boron Location N↑
1	5/9/60	09:35	S.D.	C.I.	0-16	S.P.	H-4	
2		09:45	↓	↓	0-417	↓	H-5	
3		10:02			0-227		H-6	
4		10:12			0-265		H-7	
5		10:17			0-278		H-8	
6		10:25	↓		0-289	↓	H-9	
7		11:37	EH-41		34-342	C.I.	0-265	
8		11:45	EH-20	↓	47-308		34-342	
9		12:00	S-5		New		47-308	
10		13:16	EH-37	C.I.	44-326		0-289	
11		13:18	ET-36	C.I.	63-337		44-326	
12		13:42	S-9		New		63-337	
13		14:04	EH-23	C.I.	23-293		0-278	
14		14:12	EH-30	↓	54-347		23-293	
15		14:20	ET-35	↓	72-360		54-347	
16		14:39	S-10		New		72-360	
17		15:24	EH-35	C.I.	27-325		0-227	
18		15:31	EH-14	↓	59-365		27-325	
19		15:37	ET-30	↓	77-382		59-365	
20		15:46	S-27		New	↓	77-382	
1	5/10/60	09:10	S.D.	C.I.	0-333	S.P.	H-10	
2		09:16	↓	↓	0-346	↓	H-11	
3		09:25			0-357		G-2	
4		09:35			0-371		G-3	
5		09:42			0-384		G-4	
6		09:53	↓		0-395	↓	G-5	
7		10:12	EH-5		34-356	C.I.	0-346	
8		10:14	EH-25		59-381		34-356	
9		10:19	ET-38	↓	77-402		59-381	
10		10:35	S-6		New		77-402	
11		11:01	H-3	C.I.	27-361		0-333	
12		11:05	EH-42	↓	54-390		27-361	
13		11:07	ET-45	↓	72-414		54-390	
14		11:19	S-7		New		72-414	
15		11:43	EH-9	C.I.	23-383		0-357	
16		11:49	EH-6	↓	44-392		23-383	
17		11:56	EH-18	↓	63-418		44-392	
18		13:41	S-8		New		63-418	
19		14:40	S.D.	C.I.	0-173	↓	0-371	
20		14:45	EH-40	↓	23-188		0-173	

Table 110I-I (Cont'd.)

Move	Date	Time	Fuel Element No.	From - Location		To - Location		Boron Location $N \uparrow$
21	5/10/60	14:52	EH-48	C.I.	54-241	C.I.	23-188	
22	↓	14:57	ET-20	↓	72-254	↓	54-241	_____
23	↓	15:16	S-15	↓	New	↓	72-254	_____
1	5/11/60	11:13	EH-3	C.I.	23-399	C.I.	0-384	
2	↓	11:25	EH-46	↓	54-29	↓	23-399	
3	↓	11:33	ET-6	↓	72-43	↓	54-29	_____
4	↓	11:58	S-8	↓	63-418	↓	72-43	_____
5	↓	14:21	EH-5	↓	0-346	↓	63-418	
1	5/12/60	09:50	S.D.	C.I.	0-100	S.P.	G-6	
2	↓	10:00	↓	↓	0-112	↓	G-7	
3	↓	10:11	↓	↓	0-67	↓	G-8	
4	↓	10:20	↓	↓	0-134	↓	G-9	
5	↓	10:38	H-8	↓	27-149	C.I.	0-112	
6	↓	10:43	EH-50	↓	54-179	↓	27-149	
7	↓	10:48	ET-10	↓	72-202	↓	54-179	
8	↓	11:02	S-28	↓	New	↓	72-202	_____
9	↓	11:30	EH-1	C.I.	34-145	↓	0-134	
10	↓	11:35	EH-19	↓	59-170	↓	34-145	
11	↓	11:40	ET-49	↓	77-191	↓	59-170	_____
12	↓	11:53	S-18	↓	New	↓	77-191	_____
13	↓	13:18	H-13	C.I.	27-113	↓	0-100	
14	↓	13:23	EH-47	↓	59-153	↓	27-113	
15	↓	13:28	ET-8	↓	77-170	↓	59-153	
16	↓	13:41	S-17	↓	New	↓	77-170	_____
17	↓	14:02	H-15	C.I.	23-82	↓	0-67	
18	↓	14:10	EH-32	C.I.	54-135	C.I.	23-82	
19	↓	14:15	ET-12	C.I.	72-149	↓	54-135	
20	↓	14:27	S-16	↓	New	↓	72-149	_____
1	5/13/60	09:43	S-16	Removed 1 Boron-Stainless Steel Strip and Reinserted		72-149		
2	↓	10:16	S-17	↓		77-170		
3	↓	10:41	S-18			77-191		
4	↓	11:03	S-28			72-202		

On May 12, 1960, four spikes, S-16, S-17, S-18, and S-28, containing three boron strips, were loaded into the core on the west side. These spikes contained three strips preparatory to obtaining the worth of the boron strips without changing the fuel loading. These spikes were loaded with the third strip positioned at 180 degrees. After Move 20, 5/12/60, Critical 57-20-1 was run.

On May 13, 1960, the spikes loaded on May 12, 1960, were removed, one at a time, from the core and reactor, and the third boron strip (at 180 degrees) was removed. The spikes were then reloaded into the core in the same place and orientation as they previously occupied. These spikes were not as "hot" as spike S-8 had indicated they might be. The activity level for each spike is listed below.

<u>Spike</u>	<u>Radiation Level</u>	<u>Radiation Level of Stainless Strip after Removal</u>
S-16	120 mr/hr at ~ 2 in.	2 mr/hr at ~ 1 in.
S-17	100 mr/hr at ~ 2 in.	2 mr/hr at ~ 1 in.
S-18	140 mr/hr at ~ 2 in.	2 mr/hr at ~ 1 in.
S-28	150 mr/hr at ~ 2 in.	2 mr/hr at ~ 1 in.

These spikes were handled with clean white gloves, and the boron strips were readily broken off at the tack welds.

After Move 4, 5/13/60, Critical 57-20-2 was run.

The control rod positions, after loading of the spikes, are shown in Table 110I-II.

The ACF (American Car & Foundry) element formerly in position J-1, in the fuel storage pit, was sent to building D-301, on May 23, 1960, for shipment to Oak Ridge.

The fuel status boards for the core and storage pit, for core load No. 58, are shown as Figs. 110I-1 and 2.

Table 110I-II

CONTROL ROD POSITIONS AFTER LOADING OF THE SPIKES

Date	Fuel Element No.	Balance Bank Pos. (in.)	Rod Tested	Removed (in.)	Divergent
4/25/60	S-11	6	7 & 8	47	No
4/26/60	S-13	4	7	47	Yes
			8	41	
	S-14	6	7	47	No
		4	7	47	Yes
			8	30	
5/9/60	S-5	6	4	47	No
		6	4	47	No
			3	47	
	S-9	6	3	47	No
		6	3	47	Yes
			4	37	
	S-10	6	3	47	No
		6	3	47	Yes
			4	29	
5/10/60	S-6	6	4	47	No
		6	4	47	Yes
			5	15	
	S-7	6	4	47	No
		6	5	47	Yes
	S-8	6	5	36	Yes
	S-8	5	5	38	Yes
		3½	5	40	Yes
		2	5	45	Yes
5/11/60	S-8	6	1	47	No
		6	2	47	No
		6	3	47	No
		6	4	47	No
		6	5	47	No
		6	6	47	No
		6	7	47	No
		6	8	47	No
		6	9	47	No
5/12/60	S-28	6	1	47	No
			1	47	No
			5	47	
	S-18	6	1-9 one at a time	47	No
	S-17	6	5	47	No
		6	4	47	No
		6	1	47	No
	S-16	6	1-9 one at a time	47	No

Fig. 110I-1. Core - Fuel Status Board (Core Load No. 58, 1/18/61)

Fig. 110I-2. Storage Pit - Fuel Status Board (Core Load No. 58, 1/18/61)

EBWR FUEL

(Test Report No. 110K, dated April 17, 1961, by V. M. Kolba)

The following irradiated core-I plate-type fuel elements were transferred from the fuel storage pit in building D-331 to storage at building D-330.

<u>Fuel Elements</u>	<u>Date Moved</u>
T-7	4/12/61
T-9	4/12/61
T-15	4/11/61
T-21	4/10/61
T-23 (5 plates)	4/14/61
H-7	4/13/61

Transfer was made using the EBWR fuel transfer coffin. One element was transferred at a time. The fuel status board for the storage pit, at the completion of the fuel transfer, is shown as Fig. 110K-1.

Fig. 110K-1. Storage Pit - Fuel Status Board (4/14/61)

EBWR CORE LOADING NO. 59

(Test Report No. 110L, dated June 12, 1961, by V. M. Kolba)

Fuel element moves were made (see Table 110L-I) to permit installation of an instrumented fuel element. The irradiated fuel element was transferred from the reactor to the storage pit using the fuel transfer coffin. The new instrumented fuel element was placed in the core using the rigid tool and the auxiliary winch on the crane.

Table 110L-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT (6/6/61)

Move	Time	Fuel Element No.	From - Location	To - Location
1	10:31	ET-48	C.I. 101-266	S.P. F-6
2	11:23	IFA-EH-II	New	C.I. 101-266

The revised core and storage pit fuel status charts, for core load No. 59, are shown as Figs. 110L-1 and 2.

CIRCLED LETTERS & NUMBERS
INDICATE FUEL ASSEMBLY LOCATION
FOR INDEXING COFFIN AND CARRIAGE
(SEE DWG. RE-1-18120-0).

S 20
H 8
T 1
EN 51
ET 54
SD 14
148

2
0.53
3.0
4 8

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
— BURNABLE POISON STRIPS

PREFIX LETTER CODE

H NATURAL "U" THICK PLATES
T NATURAL "U" THIN PLATES
EN ENRICHED "U" THICK PLATES
ET ENRICHED "U" THIN PLATES
S ENRICHED "U" RODS (SPIKES)
SDS DUMMY W/SOURCE
SD DUMMY STAINLESS STEEL

DATE: 6/6/61

LOAD NO.: 59

Fig. 110L-1. Core - Fuel Status Board (Core Load No. 59, 6/6/61)

NORTH

	A	B	C	D	E	F	S _D	H	I	J	K	L	M	
11														11
10								5-10-60 S.D. 0-333						10
9							5-12-60 S.D. 0-134	5-9-60 S.D. 0-289						9
8							5-12-60 S.D. 0-67	5-9-60 S.D. 0-278						8
7	4-4-60 S.D. 0-112						5-12-60 S.D. 0-112	5-9-60 S.D. 0-265						7
6	4-4-60 S.D. 0-90					6-6-61 ET-48 101-266	5-12-60 S.D. 0-100	5-9-60 S.D. 0-227						6
5														5
4							5-10-60 S.D. 0-384							4
3							5-10-60 S.D. 0-371	4-25-60 S.D. 0-417						3
2	5-25-61 T-19						5-10-60 S.D. 0-357	4-25-60 S.D. 0-16						2
1	S.D. Sb TRANS- FER DUMMY				4-12-60 S.D. 0-159	4-12-60 S.D. 0-265	4-12-60 S.D. 0-39	4-12-60 S.D. 0-371						1
	A	B	C	D	E	F	G	H	I	J	K	L	M	

SHELF

SD *

7-31-59

0-16

DATE: 6-6-61

LOAD NO.: STORAGE

PREFIX LETTER CODE

N-NATURAL "N" THICK PLATES
 T-NATURAL "N" THIN PLATES
 EN-ENRICHED "N" THICK PLATES
 ET-ENRICHED "N" THIN PLATES
 SD-DUMMY STAINLESS
 SDS-DUMMY W/SOURCE

* DUMMY HAS AN INSTRUMENTATION
 CHANNEL ATTACHED EXTERNALLY AND
 WILL NOT FIT INTO STORAGE RACK.

EBWR STORAGE RACK FUEL STATUS BOARD

Fig. 110L-2. Storage Pit - Fuel Status Board (Core Load No. 59, 6/6/61)

REPLACEMENT OF EBWR SOURCE

(Test Report No. 110M, dated October 23, 1961, by V. M. Kolba)

The moves noted in Table 110M-I were made to exchange the spent Sb source (I-B) for the reirradiated Sb source (I-A).

Table 110M-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(10/18/61)

Move	Time	Fuel Element No.	From - Location		To - Location	
1	09:18	Sb Source (1A)	S.P.	A-1	S.P.	Cask on Ledge
2	09:45	S.D.	C.I.	0-16	S.P.	B-1
3	09:50	S.T.S.D.	S.P.	A-1	C.I.	0-16
4	10:00	S.D.	C.I.	0-346		47-308
5	10:05	S.T.S.D.		0-16		0-346
6	10:08	S.D.		47-308		0-16
7	10:18	Sb Source (1B)		0-311		0-346
8	10:45	S.T.S.D. (1B)		0-346	S.P.	F-10
9	11:00	Sb (1B)	S.P.	F-10		G-12
10	11:02	S.T.S.D.		F-10		A-1
11	11:06	Sb (1A)		Cask on Ledge		A-1
12	11:12	S.T.S.D. (1A)		A-1		E-3
13	11:29	S.T.S.D. (1A)		E-3	C.I.	0-346
14	11:40	Sb (1A)	C.I.	0-346	C.I.	0-311
15	11:48	S.T.S.D.	C.I.	0-346	S.P.	G-11
16	12:50	Sb (1B)	S.P.	G-12	S.P.	G-11
17	13:00	S.D.	C.I.	0-371	C.I.	0-346
18	14:15	ET-48	S.P.	F-5	C.I.	0-371

The Sb (I-A) source was irradiated at CP-5 in location VT-22 from 15:00 on 2/20/61 to 14:30 on 10/11/61. The total irradiation was as follows:

kw-hr
14,410,519

Power Level
4600 kw

The flux level, at the 4600-kw power level, was $\sim 5.4 \times 10^{13}$ in the VT-22 position. The irradiated Sb source (I-A) was transferred to EBWR in a transfer cask, brought through the large access door, and, using the overhead crane, placed above the loading slot in the storage pit. The radiation level at the surface of the transfer cask varied from 200 mr/hr to ~ 1 r/hr. The source was then lowered from the cask into the storage pit loading slot, using the cable attached to the source. The cable was secured and the transfer cask removed from the loading slot area. It was noted that the

aluminum spacer from the CP-5 irradiation was still attached to the source. Efforts to remove this spacer with the source in the EBWR storage pit were finally rewarded. The Sb source (I-A) was then placed in the Source Transfer Stainless Dummy (S.T.S.D.) at location A-I and the cable was cut and removed from source. It should be noted here that in the future the cable should not be removed from the source until the source is loaded into the S.T.S.D. for loading into the reactor. To permit the use of the S.T.S.D., for the removal of the spent source (I-B) from the reactor, a small cask was lowered onto the ledge in the storage pit and the irradiated source (I-A) temporarily placed inside (Move 1).

Some difficulty was experienced in removing the Sb source (I-B) from the Be target dummy (Move 7). These units were separated, in the core, by raising the assembly ~ 1-2 in. and tapping on the Be target dummy until it came away from the Sb source (I-B). This same removal difficulty was experienced later between Sb source (I-B) and the old Be target dummy, at position 6-12 in the storage pit (Move 16). It was recommended that the Sb source (I-B) be carefully examined and this situation corrected before reirradiation of this source. The stainless dummy (S.D.) was transferred from the reactor to the storage pit, in air, using the overhead crane (Move 2). All other moves of the sources and fuel, to and from the reactor and storage pit, were made using the coffin. The source or fuel was transferred directly from the storage pit rack into the coffin by lowering the coffin to the water level of the storage pit. Using this method, the Sb source (I-B) gave a radiation level of ~ 6 r at ~ 2 ft and the Sb source (I-A), a level of ~ 60 r at ~ 2 ft. This may be compared to the level of the Sb source (I-B) during transfer from the reactor to the coffin where the level was ~ 200 r at ~ 2 ft through slots and loading holes.

Several additional fuel moves were made to prepare the storage pit for draining and cleanup. Fuel element ET-48 was removed from the storage pit and placed in the core (Move 18).

The Sb source (I-B) in S.T.S.D. and the plain stainless dummy (S.D.) were selected for storage in the large shipping cask. The old Be target dummy was designated for storage in the fuel transfer coffin. After completion of the storage pit cleanup and refill these items were to be returned to the storage pit.

Count rates were taken on the plant counters and the approximate average values listed as follows:

	Count Rate (counts/min)
Sb source (I-B) in Be target	~ 60
Sb source (I-B) removed from reactor	~ 2-3
Sb source (I-A) in Be target	~ 2200

RELOADING OF STORAGE PIT AFTER CLEANING

(Test Report No. 110M Addendum, dated November 14, 1961, by V. M. Kolba)

After cleaning and repainting the storage pit, the following equipment was installed: the spike fuel element storage rack, the fuel element inversion tool, the fuel element holder in the loading chute, the control rod storage rack, and the old fuel storage rack.

Storage items which had been removed previously were then replaced in the appropriate storage area. The stainless dummies from the external storage; the Sb source, in the source transfer dummy, from the shipping cask; and the old Be target from the fuel transfer coffin were all returned to the storage pit and placed in the storage rack as shown in Fig. 110M-Add.-1. On 11/8/61, the following fuel element moves were made using the coffin.

ET-48	from C.I. 0-371 to S.P. H-2
S.D.	from S.P. H-1 to C.I. 0-371

Fig. 110M-Add.-1. Storage Pit - Fuel
Status Board (11/8/61)

Because the lifting collar would not slide down, the tool could not be unlatched from the stainless dummy in the reactor. Upon return to the storage pit, the tool was unlatched from the dummy with no problem. The unit was then transferred to the reactor and again difficulty was experienced in unlocking the dummy; however, after slight shifting of the tool and dummy from the shroud, the collar released and the tool was unlatched. The trouble appears to be in a binding of the collar or lifting lines and lugs with the shroud at the external portion of the core.

ROTATION OF SPIKE FUEL ELEMENTS

(Test Report No. 110N, dated November 22, 1961, by E. L. Martinec)

The following spike fuel elements were rotated 90 or 180 deg in their respective core locations:

<u>90 deg</u>	<u>180 deg</u>
72-360	77-297
72-414	63-312
72-43	77-85
72-149	63-101
72-202	

The rotation was accomplished in three steps, with a cold critical conducted after each step, as follows: (1) four elements, 180 deg; (2) elements 72-414 and 72-202, 90 deg; and (3) three elements 72-360, 72-43, and 72-149, 90 deg.

Some significant observations were made during the rotation of the elements. Element 77-297 had a bow of 1 in. or more in one of the boron-stainless steel strips. This strip is now located on the east side of the element. There did not appear to be that amount of bowing which would cause the strip to fall out. Element 72-202 also had a bow of approximately 3/4 in. in a strip. The bowed strip is now located on the south side of that element. Element 72-414 was a little tight and some effort was required to remove it from the grid plate. Also, some difficulty was experienced in having the element inserted in its rotated position. Element 63-101 was found to have some severe discolorations on one boron-stainless steel strip which was also cocked in its position. This strip is presently located on the east side of the element.

The new spike arrangement is shown on Fig. 110N-1.

CIRCLED LETTERS & NUMBERS
INDICATE FUEL ASSEMBLY LOCATION
FOR INDEXING COFFIN AND CARRIAGE
(SEE DWG. RE-1-18120-D).

S	20
H	8
T	1
EN	51
ET	54
SD	134
	14
	148

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
— BURNABLE POISON STRIPS

PREFIX LETTER CODE

H NATURAL "U" THICK PLATES
T NATURAL "U" THIN PLATES
EN ENRICHED "U" THICK PLATES
ET ENRICHED "U" THIN PLATES
S ENRICHED "U" RODS (SPIKES)
SDS DUMMY W/SOURCE
SD DUMMY STAINLESS STEEL

DATE: 11/14/61

LOAD NO.: 59a

Fig. 110N-1. Core - Fuel Status Board (Core Load No. 59a, 11/14/61)

EXAMINATION OF IRRADIATED BORON STRIPS ON SPIKE FUEL ELEMENT FRAME

(Test Report No. 110N Addendum, dated November 22, 1961, by V. M. Kolba)

In conjunction with Reference 1 it was noted that there were patterns on the boron strips of spike S-2 frame and that one of the strips was cocked. To permit a closer evaluation of these strips and the frame the following was done (Table 110N-Add.-I). The spike was moved to the storage pit (Move 1). The spike bundle was removed from the old frame and placed in a new 3-strip frame (Moves 2 through 6). The spike with the new frame was replaced in the reactor (Move 7).

Table 110 N-Add. - I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT (11/15/61)

Move	Fuel Element No.	From - Location	To - Location	N ↑ Poison Strips
1	S-2	C.I. 63-101	S.P. C-14	
2	S-2	S.P. C-14	↓ Fuel Element Inversion Tool	
3	3 Strip Frame	New	L-5	
4	Bundle S-2	S.P. Fuel Inverter Tool	L-5	
5	S-2	↓ L-5	Shelf	
6	S-2	Shelf	↓ C-14	
7	S-2	↓ C-14	C.I. 63-101	<div style="border: 1px solid black; width: 100px; height: 20px; margin: 0 auto;"></div>

In the process of the above the following items were noted or checked out:

1. The new rack physically accommodates the spike elements satisfactorily.
2. The spikes can be lowered directly from the coffin into only the "C" row of the spike rack.
3. The spike frames fit and are held adequately in the fuel element inversion tool which permits the withdrawal of the rod bundle and inversion of the rod bundle.

4. The tool for removal of the rod bundle from the frame required some rework to properly locate on the lifting lug of the rod bundle. An old hook-type tool, used temporarily, previously, for the same purpose, was used again to perform Move 4.
5. After the rod bundle was placed in the new frame, the boron strips were slipped in place as follows:
 - a. The strips were held around the outside of the box by a line located below the mid point of the strips.
 - b. The frame containing the rod bundle, and attached to the rigid tool, was moved to the southwest corner of the storage pit; where the nose of the frame rested on the shelf and the frame, against the walls in the corner of the storage pit.
 - c. The strip was then bowed out and snapped into place, one side at a time; the element being rotated to present each strip in the most favorable orientation to perform the operation.
 - d. When all the strips were in place, the element was raised as close to the surface of the pit as practical to permit a visual check of the strip location. The line was then removed, and the spike element placed in the storage rack.

On November 16, 1961, the irradiated basket was examined in the storage pit. The frame was placed in a horizontal position, held with lines, and rotated to permit examination of all sides. The frame was under ~ 1 ft of water and the surface dose rate at the center of the frame was ~ 500 mr/hr.

Patterns were noted on the outside of the strips which, upon closer examination, appear to be flow patterns. The theory is that in these patterned areas the strip may have touched the control rod guide shroud and locally restricted flow on the outside of the strip. The remainder of the strip had a uniform brown oxide deposit. The inside of the strips, the side opposite to the externally noted flow patterns, showed a uniform oxide deposit and no indications of a heated area or other abnormality.

All of the tackwelds connecting the strips to the end-piece casting appeared sound with no separation noted. No corrosion product was noted on the spot welded areas where the Zircaloy-2 frame angles joined the Zircaloy-2 lifting plates. A uniform, dark-grey oxide film was on the Zircaloy-2 component. The Type 304 stainless steel nose-piece casting had a light-straw colored oxide film on it.

On 11/15/61, during Move 4, the fuel rod bundle was also visually examined, to the degree permitted by the facilities at hand. No bowing of the fuel rods was noted, the end fittings and side plates appeared sound, and no twist of the bundle was noted. The Zircaloy-2 was a uniform dark-grey in color and the stainless end fittings were a light-straw color.

REFERENCE

- (1) EBWR Test Report No. 110N.

TRANSPOSITION OF ENRICHED THIN FUEL ELEMENTS

(Test Report No. 110O, dated January 10, 1962, by V. M. Kolba)

The purpose of this fuel transposition was to place the most highly irradiated and scaled, enriched thin fuel elements in regions of lower importance in the core, and to replace these elements with the available enriched thin elements, within the reactor, which have less burnup and scale.

The stainless steel dummy element was removed from the core and transferred to the storage pit, in air, using the rigid tool and the overhead crane. All fuel element moves in the reactor were made using the "A" frame, mounted on the indexing plug, and the rigid fuel handling tool.

During fuel element moves, all nine rods were banked at 10 in. A base count was taken and after every odd-numbered move (i.e. a completed switch) two 1-minute counts were taken using the plant counters. The fuel elements were lowered in 6-in. increments, when in the active portion of the core, until fully seated.

The fuel moves are listed in Table 110O-I. The transposed fuel elements are shown in Fig. 110O-1. The final core loading and the storage pit fuel status are shown in Figs. 110O-2 and 3.

Several items of interest occurred during the switching. On lowering ET-3 it was found that it could not be completely lowered. It was raised again and a bent spring was observed on the adjacent spike S-4. ET-3 was then returned to its original location. The spike S-4 was then raised, rotated 90 deg counter-clockwise, and lowered, thereby compressing the bent spring against the control rod guide shroud. ET-3 was then placed in the core next to S-4.

Table 1100-1

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(1/2-4/62)

Date	Time	Move	Fuel Element No.	From - Location	To - Location
1/2/62		Critical with rods (Nos. 1-8) @ 14.5 in. and rod No. 9 @ 15 in.			
1/3/62	13:30	1	S.D.	C.I. 0-395	S.P. B-11
	13:50	2	ET-19	63-20	C.I. 0-395
	14:02	3	ET-6	54-29	63-20
	14:09	4	ET-56	109-99	54-29
	14:16	5	ET-14	59-48	109-99
	14:23	6	ET-1	95-83	59-48
	14:30	7	ET-28	54-73	95-83
	14:37	8	ET-17	109-148	54-73
	14:42	9	ET-34	59-64	109-148
	15:36	10	ET-3	95-113	59-64
	15:47	11	ET-8	59-153	95-113
	15:56	12	ET-26	95-188	59-153
	16:04	13	ET-12	54-135	95-188
	16:12	14	ET-42	109-205	54-135
	16:19	15	ET-37	63-126	109-205
1/4/62	08:46	16	ET-15	101-160	63-126
	08:53	17	ET-10	54-179	101-160
	09:02	18	ET-55	109-254	54-179
	09:09	19	ET-49	59-170	109-254
	09:18	20	ET-13	95-219	59-170
	09:30	21	ET-40	59-259	95-219
	09:38	22	ET-27	95-294	59-259
	09:44	23	ET-20	54-241	95-294
	10:00	24	ET-47	89-233	54-241
	10:07	25	ET-32	54-284	89-233
	10:14	26	ET-52	109-359	54-284
	10:18	27	ET-9	59-276	109-359
	10:28	28	ET-23	95-324	59-276
	10:33	29	ET-30	59-365	95-324
	10:41	30	ET-22	95-400	59-365
	10:46	31	ET-35	54-347	95-400
	10:52	32	ET-50	109-417	54-347
	10:57	33	ET-36	44-326	109-417
	11:05	34	ET-25	101-371	44-326
	11:09	35	ET-45	54-390	101-371
	11:17	36	ET-33	109-42	54-390
	11:21	37	ET-38	59-381	109-42
	11:28	38	ET-16	95-7	59-381
	11:34	39	ET-19	0-395	95-7
Critical with all rods banked @ 14.17 in.

Fig. 110O-1. Changed Fuel Elements

Fig. 1100-2. Core - Fuel Status Board (Core Load No. 59b, 1/4/62)

PREFIX LETTER CODE

WOLFF

[illegible]

Upon completion of the transposition, S-4 was rotated back to its original position and a critical was run. The change in control rod bank position was 14.5-14.7 in. = ~ 0.33 in. on the bank, equal to a reactivity change of $\sim 0.07\% \rho$. After the critical was run, the rods were banked at 10 in. and S-4 was transferred to the storage pit using the coffin. The fuel bundle was removed from the old 3-strip boron frame, placed in a new 3-strip boron frame, and returned to the reactor. No detectable change in critical rod position was noted after the switch of frames.

EBWR LOADING 1B (24 SPIKE): CORE LOAD NO. 60

(Test Report No. 110P, dated February 2, 1962, by E. L. Martinec)

The purpose of this loading was to determine the feasibility of centralizing the spike elements in a ring inside the 8 outer control rods. This configuration would enable the installation of a preferential riser around the fuel inside the 8 outer control rods. Available excess reactivity of this configuration was also determined.

During the fuel moves (Table 110P-I) the control rods were banked at 6 in. and two 1-minute counts were taken before the insertion of each element into the core. Each element was lowered in 6-in. increments until it was fully seated while the count rate was carefully monitored.

Table 110P-1

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT

Move	Fuel Element No.	From - Location		To - Location		Strip Orientation N ↑
	Critical					
1	S-1	C.I.	63-312	S.P.	C-14	
2	ET-44	C.I.	80-342	C.I.	63-312	
3	S-1	S.P.	C-14	C.I.	80-342	└───
4	S-2	C.I.	63-101	S.P.	C-14	
5	ET-4	C.I.	80-131	C.I.	63-101	
6	S-2	S.P.	C-14	C.I.	80-131	└───┐
7	S-3	C.I.	77-297	S.P.	C-14	
8	ET-31	C.I.	80-237	C.I.	77-297	
9	S-3	S.P.	C-14	C.I.	80-237	└───┐
10	S-4	C.I.	77-85	S.P.	C-14	
11	ET-7	C.I.	80-25	C.I.	77-85	
12	S-4	S.P.	C-14	C.I.	80-25	└───┐
13	S-5	C.I.	47-308	S.P.	C-14	
14	ET-20	C.I.	95-294	C.I.	47-308	
15	S-5	S.P.	C-14	C.I.	95-294	└───┐
16	S-9	C.I.	63-337	S.P.	C-14	
17	ET-35	C.I.	95-400	C.I.	63-337	
18	S-9	S.P.	C-14	C.I.	95-400	└───┐
19	S-17	C.I.	77-170	S.P.	C-14	
20	ET-12	C.I.	95-188	C.I.	77-170	
21	S-17	S.P.	C-14	C.I.	95-188	└───┐
22	S-27	C.I.	77-382	S.P.	C-14	
23	ET-28	C.I.	95-83	C.I.	77-382	
24	S-27	S.P.	C-14	C.I.	95-83	└───┐
25	ET-39	C.I.	89-264	C.I.	0-395	
26	S-15		72-254		89-264	└───┐

Table 110P-1 (Cont'd.)

Move	Fuel Element No.	From - Location		To - Location		Strip Orientation N ↑		
27	ET-30	↓	95-324	↓	72-254			
28	S-13		77-276		95-324			
29	ET-29		89-338		77-276			
30	S-12		72-308		89-338			
31	ET-11		89-370		72-308			
32	S-10		72-360		89-370			
33	ET-19		95-7		72-360			
34	S-6		77-402		95-7			
35	ET-24		89-21		77-402			
36	S-7		72-414		89-21			
37	ET-41		89-53		72-414			
38	S-8		72-43		89-53			
39	ET-8		95-113		72-43			
40	S-14		77-64		95-113			
41	ET-2		89-127		77-64			
42	S-11		72-96		89-127			
43	ET-53		89-158		72-96			
44	S-16		72-149		89-158			
45	ET-40		95-219		72-149			
46	S-18		77-191		95-219			
47	ET-32		89-233		77-191			
48	S-28		72-202		89-233			
49	ET-39		↘		0-395		↘	72-202
	Critical							
50	S.D.		C.I.		0-183		S.P.	C-2
51	↓		↓		0-78		↓	C-3
52					0-159			C-4
53					0-371			C-5
54	ET-20		↘		47-308		C.I.	0-183
55	S-21		NEW				C.I.	47-308
56	EH-13		C.I.		47-97			0-78
57	S-24		NEW					47-97
58	EH-44		C.I.		47-203			0-159
59	S-19		NEW					47-203
60	EH-43		C.I.		47-414			0-371
61	S-22		NEW				↓	47-414

Before any fuel was moved, the boric acid concentration of the reactor water was brought to 1.18 gm/gal; and the reactor was determined to be critical with control rods banked at 17.5 in. After Move 15, control rods, Nos. 2, 3, and 4 were raised, one at a time, and then dropped to check that the 8-rod shutdown criteria were being maintained. An additional 11 lb. of boric acid were added after Move 23 to bring the boric acid concentration to 1.9 gm/gal. Each rod was then raised to 48 in. and dropped to check the core criticality. After Move 49, the core was critical with the control rods banked at 15.5 in. The rods were dropped after the critical and then each rod raised to 48 in. and dropped again to check for 8-rod shutdown margin. Four new spikes were then added to the core and the reactor was then found to be critical with the control rods banked at 14.6 in. Each rod was again raised to 48 in. to check for 8-rod shutdown margin.

One boron-stainless steel strip was removed from each of eight spike elements. These elements were S-1, S-2, S-3, S-4, S-5, S-9, S-17, and S-27. The only difficulty encountered with strip removal occurred when one strip snapped off prematurely before it could be secured by the "noose" tool. The strip dropped between the storage rack and the storage pit wall. In trying to retrieve the strip with the "noose" tool, the tool became jammed and required latter reworking of the noose end of the tool before the remaining strips could be handled. The strip that was dropped was retrieved.

The spike fuel handling tool in the coffin was also severely damaged during this procedure. The coffin was moved laterally without having the tool completely raised into the coffin. The tool was wedged between the coffin and the floor, adjacent to the reactor vessel, causing the tool to be bent. Fortunately, none of the telescoping tubes were bent when this happened. As this occurred after all fuel moves, between the reactor and storage pit, were completed, there was no delay in the core loading. The tool has been repaired and is back in operation.

All stainless dummy elements were transferred, in air, with the use of the crane and the rigid handling tool. All fuel moves within the core were made using the "A" frame mounted on the indexing plug and the rigid handling tool.

The fuel status boards for the core and the storage pit are shown as Figs. 110P-1 and 2.

Fig. 110P-1. Core - Fuel Status Board (Core Load No. 60, 1/26/62)

Fig. 110P-2. Storage Pit Fuel Status Board
(Core Load No. 60, 1/26/62)

CORE LOADING FOR 100-Mw OPERATION, CORE LOADING NO. 61

(Test Report No. 110J Addendum and Supplements 1, 2, and 3,
dated March 2, 1962, by V. M. Kolba)

The purpose of this loading was to:

1. return the IB spike loading to the ring outside of the control rods
2. complete the loading of the 28 spikes
3. completely fill in the core
4. remove boron strips to increase reactivity

Prior to beginning fuel movements, H_3BO_3 was added to the reactor to a concentration of 2 gm/gal. During the fuel moves, the control rods were banked at 6 in. and two 1-minute counts were taken before and after the insertion of each element into the core. Each element was lowered into the core in 6-in. increments until fully seated. The count rate was carefully monitored during this procedure.

The stainless steel dummies were transferred to the storage pit, in air, using the rigid tool and the winch on the overhead crane. They were moved in sequence to save time, in terms of tool changes and personnel movements. Radiation levels from the air transfer of the stainless dummies were ~ 100 - 170 mr/hr at ~ 15 ft. Fuel moves in the reactor were made using the "A" frame mounted on the indexing plug and the rigid handling tool. Transfer of irradiated fuel elements, to and from the storage pit and core, was accomplished with the fuel transfer coffin. New spikes were added to the core using the winch on the overhead crane and the rigid tool. All spikes were checked for proper number and handled with clean white gloves prior to loading into the reactor.

Boric acid concentrations and bank critical positions are noted in Tables 110J-Add.-I, IIa, IIb, and III. Periodically the rods were checked for 8-rod shutdown with H_3BO_3 . These tests are also noted in Tables 110J-Add.-IIa, IIb, and III. Table 110J-Add.-IV is a summary of the critical tests conducted (spike, poison strip, and boric acid data) during the loading.

During the course of the last core changes, several H_3BO_3 removal tests to check for 9-rod shutdown were conducted. These tests are summarized in Table 110J-Add.-V and plotted in Fig. 110J-Add.-1.

The removal of the boron strips was conducted in the following manner. A spike fuel element was lowered from the coffin into the storage pit, at location M-5, so that just the nose of the element was in the storage rack. The special tool was then used to bow out the strip and loosen the strip from the tucked-in position at the upper end. A wire noose was then placed over the strip and tightened at the center notches in the strip. The strip was then bent downward until the tack welds holding the strips to the nosepiece of the fuel element frame were broken. Some strips were removed with great difficulty while others came free readily. This seems to be a function of the heaviness of the tack welds. The strips were then placed in an aluminum container attached to the storage rack in the M-12 position. The strip from spike S-8 was set aside in the box located at K-12. This strip was to be examined metallurgically in the caves in building D-301. This strip was in the reactor for ~ 5664 Mw-hr of reactor operation (from May 10, 1960 to February 16, 1962). The boron strips had an activity of ~ 50 r/hr at 1 in.

The fuel status boards for the core and storage pit, for core load No. 61, are shown as Figs. 110J-Add.-2 and 3.

Reactivity changes and additional physics data are reported in Reference 1.

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT (1/30 and 31/62)

Move	Date	Time	Fuel Element No.	From - Location	To - Location	Strip Orientation N ↑
2.0 gm/gal H ₃ BO ₃ , Critical with rods (Nos. 1-9) banked @ 14.87 in.						
1	1/30/62	09:59	ET-39	C.I. 72-202	C.I. 0-395	
2		10:11	S-28	89-233	72-202	
3		10:23	ET-32	77-191	89-233	
4		10:31	S-18	95-219	77-191	
5		10:39	ET-40	72-149	95-219	
6		10:46	S-16	89-158	72-149	
7		10:55	ET-53	72-96	89-158	
8		11:04	S-11	89-127	72-96	
9		11:12	ET-2	77-64	89-127	
10		11:20	S-14	95-113	77-64	
11		11:28	ET-8	72-43	95-113	
12		11:38	S-8	89-53	72-43	
13		13:03	ET-41	72-414	89-53	
14		13:11	S-7	89-21	72-414	
15		13:20	ET-24	77-402	89-21	
16		13:27	S-6	95-7	77-402	
17		13:35	ET-19	72-360	95-7	
18		13:44	S-10	89-370	72-360	
19		13:52	ET-11	72-308	89-370	
20		13:55	S-12	89-338	72-308	
21		14:09	ET-29	77-276	89-338	
22		14:17	S-13	95-324	77-276	
23		14:25	ET-30	72-254	95-324	
24		14:32	S-15	89-264	72-254	
25		14:40	ET-39	0-395	89-264	
26		14:48	ET-28	77-382	0-395	
27		14:56	S-27	95-83	77-382	N
28		15:05	ET-28	0-395	95-83	
29		15:13	ET-12	77-170	0-395	
30		15:22	S-17	95-188	77-170	S
31		15:34	ET-12	0-395	95-188	
32		15:43	ET-35	63-337	0-395	
33		15:51	S-9	95-400	63-337	S
34		15:59	ET-35	0-395	95-400	
35		16:08	S.D.	0-206	S.P. C-6	
36		16:14	S.D.	0-346	C-9	
37		16:22	S.D.	0-145	C-7	
38		16:28	S.D.	0-90	C-8	
2.0 gm/gal H ₃ BO ₃ , Critical with rods (Nos. 1-8) @ 15.9 in. and rod No. 9 @ 16.0 in.						

Table 100J-Add.-1 (Cont'd.)

[illegible]

Table 110J-Add.-IIa

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(2/14/62)

Move	Time	Fuel Element No.	From - Location	To - Location	Strip N ↑ Orientation
H_3BO_3 1.79 gm/gal, Critical with rods (Nos. 1-9) banked @ 15.5 in.					
1	11:03	EH-8	C.I. 44-220	C.I. 0-183	
2	11:16	S.D.	0-122	S.P. D-1	
3	11:22	↓	0-16	↓ D-2	
4	11:28		0-417	↓ D-3	
5	11:34	↓	0-53	↓ D-4	
6	13:35	ET-48	S.P. H-2	C.I. 44-220	
7	13:51	H-3	C.I. 0-333	↓ 0-417	
8	14:03	S-20	New	0-333	
9	14:14	EH-35	C.I. 0-227	↓ 0-53	
10	14:23	S-23	New	0-227	
H_3BO_3 1.78 gm/gal, Critical with rods (Nos. 1-8) @ 15 in. and rod No. 9 @ 15.5 in.					
11	15:14	S-32	New	C.I. 0-16	
12	15:20	S-31	New	C.I. 0-122	
H_3BO_3 1.78 gm/gal, Critical with rods (Nos. 1-8) @ 14.7 in. and rod No. 9 @ 15 in.					

Table 110J-Add.-IIb

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(2/14/62)

Rod	Position, in.	Bank Position of Other 8 Rods, in.	Remarks
6	48	5	Subcritical
4	↓	5	Subcritical
2		5	Divergent
2		3.5	Divergent
2		2	Subcritical
8		0	Subcritical
7		0	Subcritical
5		0	Subcritical
3		0	Subcritical
1		0	Subcritical
9	↓	0	Subcritical

Table 100J-Add.-III

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(2/15, 16 and 19/62)

Move	Date	Time	Fuel Element No.	From - Location	To - Location	Strip Orientation N ↑
1	2/15/62	H ₃ BO ₃ 2.52 gm/gal, Critical with rods (Nos. 1-8) @ 16.9 in., and rod No. 9 @ 17.1 in.				
2		13:22	S-28	C.I. 72-202	S.P. M-5	Remove 1 strip
3		14:02	S-28	S.P. M-5	C.I. 72-202	W
4		14:15	S-7	C.I. 72-414	S.P. M-5	Remove 1 strip
		14:48	S-7	S.P. M-5	C.I. 72-414	E
		H ₃ BO ₃ 2.52 gm/gal, Critical with rods (Nos. 1-9) @ 16.2 in.				
5	2/16/62	09:21	S-15	C.I. 72-254	S.P. M-5	Remove 1 strip
6		09:49	S-15	S.P. M-5	C.I. 72-254	N
7		10:32	S-8*	C.I. 72-43	S.P. M-5	Remove 1 strip
8		10:58	S-8	S.P. M-5	C.I. 72-43	S
		H ₃ BO ₃ 2.56 gm/gal, Critical with rods (Nos. 1-8) @ 15.5 in. and rod No. 9 @ 15.1 in.				
		No. 2 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.19 10 x 10 ⁻¹²				
		No. 6 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.77 10 x 10 ⁻¹²				
9		14:22	S-16	C.I. 72-149	S.P. M-5	Remove 1 strip
10		14:54	S-16	S.P. M-5	C.I. 72-149	W
		Base level all rods @ 0 in. - linear flux 0.12 10 x 10 ⁻¹²				
		No. 2 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.18 10 x 10 ⁻¹²				
		No. 6 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.75 10 x 10 ⁻¹²				
		No. 1 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.13 10 x 10 ⁻¹²				
		No. 3 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.13 10 x 10 ⁻¹²				
		No. 4 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.15 10 x 10 ⁻¹²				
		No. 8 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.16 10 x 10 ⁻¹²				
		No. 5 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.21 10 x 10 ⁻¹²				
		No. 7 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.2 10 x 10 ⁻¹²				
		No. 9 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.13 10 x 10 ⁻¹²				
11		15:44	S-10	C.I. 72-360	S.P. M-5	Remove 1 strip
12		16:18	S-10	S.P. M-5	C.I. 72-360	E
		H ₃ BO ₃ 2.56 gm/gal, Critical with rods (Nos. 1-9) banked @ 14.85 in.				
	2/19/62	H ₃ BO ₃ 3.48 gm/gal, Critical with rods (Nos. 1-8) @ 17.2 in. and rod No. 9 @ 17.55 in.				
13		10:52	S-12	C.I. 72-308	S.P. M-5	Remove 1 strip
14		11:24	S-12	S.P. M-5	C.I. 72-308	N
15		11:33	S-11	C.I. 72-96	S.P. M-5	Remove 1 strip
16		12:01	S-11	S.P. M-5	C.I. 72-96	S
		H ₃ BO ₃ 3.48 gm/gal, Critical with rods (Nos. 1-8) @ 16.5 in. and rod No. 9 @ 17.1 in.				
		Base level all rods @ 0 in. - linear flux 0.11 10 x 10 ⁻¹²				
		No. 1 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.11 10 x 10 ⁻¹²				
		No. 2 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.11 10 x 10 ⁻¹²				
		No. 3 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.12 10 x 10 ⁻¹²				
		No. 4 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.13 10 x 10 ⁻¹²				
		No. 5 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.15 10 x 10 ⁻¹²				
		No. 6 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.25 10 x 10 ⁻¹²				
		No. 7 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.16 10 x 10 ⁻¹²				
		No. 8 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.13 10 x 10 ⁻¹²				
		No. 9 rod @ 48 in., balance of rods @ 0 in. - linear flux 0.11 10 x 10 ⁻¹²				

* This strip placed in box with beryllium target - to be examined.

Table 110J-Add.-IV

SUMMARY OF CRITICAL TESTS DURING LOADING

Date	Core Load	H ₃ BO ₃ , gm/gal	Rod Bank Position in.			Spike Location*	Spike Data			
			Nos. 1-8	No. 9	Nos. 1-9		No. of Spikes	1 strip	2 strip	3 strip
1/30/62	1B (No. 60)	2.0			14.87	4 outside 20 inside	24	4	20	--
1/30/62	Table 110J-Add.-I	2.0	15.9	16		19 outside 5 inside	24	4	20	--
1/31/62	Table 110J-Add.-I	2.19			16.6	28 outside	28	4	24	--
2/14/62	Table 110J-Add.-I	1.79			15.5	28 outside	28	4	24	--
2/14/62	Table 110J-Add.-IIa & b	1.78	15.0	15.5		30 outside	30	4	25	1
2/14/62	Table 110J-Add.-IIa & b	1.78	14.7	15.0		32 outside	32	4	26	2
2/15/62	Table 110J-Add.-IIa & b	2.52	16.9	17.1		32 outside	32	4	26	2
2/15/62	Table 110J-Add.-III	2.52			16.2	32 outside	32	6	24	2
2/16/62	Table 110J-Add.-III	2.56	15.5	15.1		32 outside	32	8	22	2
2/16/62	Table 110J-Add.-III	2.56			14.85	32 outside	32	10	20	2
2/19/62	Table 110J-Add.-III	3.48	17.2	17.55		32 outside	32	10	20	2
2/19/62	Table 110J-Add.-III	3.48	16.5	17.1		32 outside	32	12	18	2

* Reference control rod ring.

Table 110J-Add.-V
BORIC ACID REMOVAL TESTS
(9 Rod Shutdown Check)

Date	H ₃ BO ₃ , gm/gal	Rod position when critical, in.			Bank
		Nos. 1-8	No. 9	Nos. 1-9	
Loading 1B					
1/25/62	1.9	—	—	14.6	14.6
1/25/62	1.3	13.2	14.2	—	~13.25
1/26/62	0.91	—	—	12.35	12.35
1/26/62	0.6	11.6	11.58	—	~11.6
Loading 1A					
1/31/62	2.2	—	—	16.6	16.6
2/1/62	1.55	—	—	14.8	14.8
2/2/62	1.1	13.68	13.8	—	~13.7
2/5/62	0.89	—	—	13.2	13.2
2/5/62	0.66	—	—	12.68	12.68
2/5/62	~0.42	—	—	12.05	12.05
2/6/62	0.35/0.39	—	—	12.1	12.1
Loading 1A					
2/19/62	3.48	16.5	17.1	—	~16.55
2/21/62	2.43/2.28	13.7	13.6	—	~13.7
2/21/62	1.61/1.69	12.1	12.3	Bubbling Air	~12.1
2/21/62	1.61/1.69	12.1	12.4	Bubbling Air	~12.1
2/22/62	1.67/1.69	—	—	12.3	12.3
2/22/62	1.14	—	—	11.15	11.15
2/22/62	0.798	—	—	10.45	10.45
2/23/62	0.66	10.0	10.35	—	~10.0
2/23/62	0.43	9.65	9.82	—	~ 9.66
2/23/62	0.43	9.65	10.1	—	~ 9.66

Fig. 110J-Add.-1.

Boric Acid Removal Tests (9-Rod Shutdown)

Fig. 110J-Add.-2. Core - Fuel Status Board
(Core Load No. 61, 2/23/62)

EBWR STORAGE RACK FUEL STATUS BOARD

DATE 2/29/62

LOAD NO. 61

Fig. 110J-Add.-3. Storage Pit - Fuel Status Board
(Core Load No. 61, 2/23/62)

REFERENCE

- (1) EBWR Test Report No. 100J.

EBWR CORE LOADING NO. 62

(Test Report No. 110Q, dated May 24, 1962, by V. M. Kolba)

Fuel element moves for core loading No. 62 are listed in Table 110Q-I.

Table 110Q-I
HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(5/21/62)

Time	Fuel Element No.	From — Location	To — Location
11:54	ET-43	C.I. 101-54	S.P. L-1
14:07	EH-IFÄ-III	New	C.I. 101-54

Prior to performing the fuel moves a critical was run. The reactor was critical with rods, Nos. 1 through 8, at 16.9 in. and rod No. 9 at 17.45 in. During the fuel moves the rods were banked at 6 in. and the plant counters were noted as the fuel elements were removed from or added to the core in 6-in. increments. The irradiated element removed from the core was transferred to the storage pit by means of the coffin. The new element was loaded into the core using the rigid tool and the auxiliary hoist on the overhead crane. After the loading was completed a critical was run. With the rods set as follows: Nos. 1 through 8 at 16.9 in. and rod No. 9 at 17.45 in., the reactor was now on an ~ 55 -sec period, equal to $\sim 0.11\% \rho$. This is on the low end of the reactivity addition estimated prior to the test.

Prior to loading element IFA-III, it was necessary to elongate the holes in the handling end fitting to permit the handling tool pins to engage the fuel element. The coffin tool was also checked in the unit after the holes were elongated by filing. The new fuel element was then cleaned with acetone prior to loading into the reactor.

During the fuel movements, the reactor water level was ~ 16 ft, which was lower than usual. The usual water level during a fuel element transfer was about 27 ft. The radiation level, while transferring the fuel element from the core into the coffin, was > 500 r/hr (off scale) at the open hole in the core indexing plug, ~ 2 ft from the fuel element. The radiation level at the operator's location on the coffin bridge, was ~ 5 mr/hr.

The fuel status boards for the core and storage pit are shown as Figs. 110Q-1 and 2.

Fig. 110Q-1. Core - Fuel Status Board
(Core Load No. 62, 5/21/62)

DATE 5/21/62
LOAD NO. 62

H - NATURAL "U" THICK PLATES
T - NATURAL "U" THIN PLATES
EH - ENRICHED "U" THICK PLATES
ET - ENRICHED "U" THIN PLATES
SD - DUMMY STAINLESS
SDS - DUMMY WITH SOURCE
STSD - SOURCE TRANSFER STAINLESS DUMMY
S - ENRICHED "U" SPIKE
SF - SPIKE FRAME

CR 1	CR 3	CR 5	CR 7	
CR 2	CR 4	CR 6	CR 8	

[illegible]

27 26 25 24 23 22 21 20 19 18 17 16 15 14
SHELF

[illegible]

Fig. 100Q-2. Storage Pit - Fuel Status Board (Core Load No. 62, 5/21/62)

CORE LOADING NO. 63 (OPERATION "STRIP-TEASE")

(Test Report No. 110R(A), dated October 2, 1962, by V. M. Kolba)

The purpose of removing the boron strips during this core loading was to increase the reactivity of the core to permit operation at higher power levels.

Prior to beginning the fuel movements, the water level in the reactor was raised and boric acid was added to the reactor water to a concentration of 4.1 gm/gal. During the fuel moves the control rods were banked at 6 in. Each fuel element was removed and replaced in 6-in. increments until fully seated; during which time the count rate was carefully monitored. Transfer of irradiated fuel elements, to and from the storage pit and core, was accomplished using the fuel transfer coffin.

The removal of born strips was conducted in the following manner. For example, the spike fuel element was lowered from the coffin into the storage pit at location M-5 so that just the nose of the element was in the storage rack grid. The boron strip to be removed was facing west. The strip was then bowed out and loosened from the tucked-in position at the upper end using a special tool. A wire noose was then placed over the strip and tightened at the center notches in the strip. The strip was then bent downward until the tack welds holding the strip to the nosepiece of the fuel element frame were broken.

Some strips were removed with some difficulty (angle bend ~ 90 deg), a few came free at an angle bend ~ 30 deg, but the majority of the strips were removed at an angle bend between these extremes, i.e., an angle bend of ~ 60 deg. The strips were then placed in one of two aluminum containers attached to the storage rack at the J-12 and M-12 positions. An exception, the strip from S-3 was placed in the square box located at K-12 for future examination. This strip was in the reactor for $\sim 33,831$ Mw-hr of reactor operation from April 7, 1960 to September 19, 1962.

Fuel moves, boric acid concentrations, and bank critical positions are listed in Tables 110R(A)-I, II, and III. Periodically, the rods were checked for 8-rod shutdown with H_3BO_3 . These tests are also noted in the tables. Table 110R(A)-II includes a summary of the critical tests, spike, poison strip, and boric acid data during the loading. At the end of strip removal, and H_3BO_3 removal test was conducted to check for 9-rod shutdown. The results are summarized in Table 110R(A)-II and plotted in Figs. 110R(A)-1 and 2.

Table 110R(A)-1

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(9/19-21, 26, and 27/62)

Move	Date	Time	Fuel Element No.	From - Location	To - Location	Strip Orientation N ↑
	9/19/62	Critical with rods (Nos. 1-9) banked @ 19.5 in. $H_3BO_3 = \sim 4.1$ gm/gal				
1		09:40	S-3	C.I. 63-231	S.P. M-5	Remove 1 Strip*
2		10:15	S-3	S.P. M-5	C.I. 63-231	E
3		10:32	S-4	C.I. 63-20	S.P. M-5	Remove 1 Strip
4		11:00	S-4	S.P. M-5	C.I. 63-20	W
		Critical with rods (Nos. 1-9) banked @ 18.85 in. $H_3BO_3 = \sim 4.1$ gm/gal				
5		13:35	S-1	C.I. 63-312	S.P. M-5	Remove 1 Strip
6		14:05	S-1	S.P. M-5	C.I. 63-312	W
7		14:20	S-2	C.I. 63-101	S.P. M-5	Remove 1 Strip
8		14:55	S-2	S.P. M-5	C.I. 63-101	E
		Critical with rods (Nos. 1-9) banked @ 18.3 in. $H_3BO_3 = \sim 4.1$ gm/gal				
9		16:05	S-30	C.I. 63-418	S.P. M-5	Remove 1 Strip
10		16:45	S-30	S.P. M-5	C.I. 63-418	<u>N</u>
	9/20/62	09:18	S-25	C.I. 63-206	S.P. M-5	Remove 1 Strip
12		09:56	S-25	S.P. M-5	C.I. 63-206	<u>S</u>
		Critical with rods (Nos. 1-9) banked @ 17.8 in. $H_3BO_3 = \sim 4.1$ gm/gal				
13		10:50	S-26	C.I. 63-126	S.P. M-5	Remove 1 Strip
14		11:20	S-26	S.P. M-5	C.I. 63-126	<u>N</u>
		Critical with rods (Nos. 1-9) banked @ 17.51 in. $H_3BO_3 = \sim 4.10$ gm/gal Added H_2O				
		Critical with rods (Nos. 1-9) banked @ 16.71 in. $H_3BO_3 = \sim 3.78$ gm/gal				
15		15:10	S-6	C.I. 77-402	S.P. M-5	Remove 1 Strip
16		15:53	S-6	S.P. M-5	C.I. 77-402	<u>S</u>
17		16:12	S-18	C.I. 77-191	S.P. M-5	Remove 1 Strip
18		16:48	S-18	S.P. M-5	C.I. 77-191	<u>N</u>
	9/21/62	Critical with rods (Nos. 1-9) banked @ 16.08 in. $H_3BO_3 = \sim 3.78$ gm/gal				
19		09:20	S-10	C.I. 72-360	S.P. C-14	
20		09:35	S-29	63-337	C.I. 72-360	
21		09:55	S-29	72-360	S.P. M-5	Remove 1 Strip
22		10:27	S-29	S.P. M-5	C.I. 72-360	<u>S</u>
23		10:38	S-29	C.I. 72-360	63-337	<u>S</u>
24		11:00	S-10	S.P. C-14	72-360	E
25		11:23	S-15	C.I. 72-254	S.P. C-14	
26		13:32	S-13	77-276	C.I. 72-254	

* Strip from S-3 in square box in storage pit for possible examination.

Table 110R(A)-II
SUMMARY OF CRITICALS

Date	No. Strips Removed	H_3BO_3 (gm/gal)	Control Rod Bank (Nos. 1-9) Position, in.	Number of Spikes Having			
				3 Strips per Spike	2 Strips per Spike	1 Strip per Spike	0 Strip per Spike
9/19/62	0	~4.1	19.5	2	18	12	0
	2	~4.1	18.85	2	16	14	0
	2	~4.1	18.3	2	14	16	0
9/20/62	2	~4.1	17.8	2	12	18	0
	1	~4.1	17.51	2	11	19	0
	0	~3.78	16.71	2	11	19	0
	2	~3.78	16.08	2	9	21	0
9/21/62	2	~3.78	15.61	2	7	23	0
	1	~3.78	15.37	2	6	24	0
9/26/62	0	~3.73	15.31	2	6	24	0
	2	~3.73	14.8	2	6	22	2
	1	~3.73	14.5	2	6	21	3
9/27/62	0	~4.26	15.62	2	6	21	3
	1	~4.26	15.19	2	6	20	4
Dilution of H_3BO_3 Check for 9-rod shutdown							
9/28/62		~4.26	15.19	2	6	20	4
		~3.1	12.95	2	6	20	4
		~2.32	11.16	2	6	20	4
		~1.27	9.05	2	6	20	4
		~0.70	7.49	2	6	20	4
		~0.67	~ 7.55	2	6	20	4
Total Core				in Spike Ring			
Beginning average strips/spike = 1.687				1.57			
Present average strips/spike = 1.187				1.0			

Table 110R(A)-III
8-ROD SHUTDOWN CHECK

Date	Control Rod Setting	Results
9/26/62 (after Move 7)	$H_3BO_3 = \sim 3.73$ gm/gal	
	No. 6 rod @ 47.8 in., balance of rods @ 8.2 in.	Critical
	No. 8 rod @ 47.8 in., balance of rods @ 2 in.	
	No. 4 rod @ 47.8 in., balance of rods @ 5+ in.	
	No. 2 rod @ 47.9 in., balance of rods @ 8 in.	Subcritical

Table 110R(A)-III (Cont'd.)

Date	Control Rod Setting	Results
9/26/62 (after Move 7)	No. 9 rod @ 48 in., balance of rods @ 8 in. No. 1 rod @ 48.25 in., balance of rods @ 8 in. No. 3 rod @ 47.9 in., balance of rods @ 8 in. No. 5 rod @ 48.1 in., balance of rods @ 8 in. No. 7 rod @ 47.9 in., balance of rods @ 8 in.	
9/26/62 (after Move 9)	$H_3BO_3 = \sim 3.63$ gm/gal No. 6 rod @ 39.5 in., balance of rods @ 0 in. No. 7 rod @ 48 in., balance of rods @ 0 in. No. 8 rod @ 45 in., balance of rods @ 0 in.	Critical Subcritical Critical
9/27/62 (after Move 11)	$H_3BO_3 = \sim 4.26$ gm/gal No. 6 rod @ 47.8 in., balance of rods @ 0 in.	Subcritical
9/28/62	$H_3BO_3 = \sim 0.675$ gm/gal No. 1 rod @ 18.5 in., balance of rods @ 0 in. No. 2 rod @ 16.5 in., balance of rods @ 0 in. No. 3 rod @ 17.0 in., balance of rods @ 0 in. No. 4 rod @ 18.2 in., balance of rods @ 0 in. No. 5 rod @ 18.5 in., balance of rods @ 0 in. No. 6 rod @ 17.0 in., balance of rods @ 0 in. No. 7 rod @ 18.5 in., balance of rods @ 0 in. No. 8 rod @ 18.6 in., balance of rods @ 0 in. No. 9 rod @ 48 in., balance of rods @ 0 in.	Critical Slight rise

FIG. 1

Fig. 110R(A)-1. Effect of Boric Acid Removal

FIG. 2

Fig. 110R(A)-2. Effect of Boron Strip Removal

Water was added, after Move 14 on 9/20/62, to raise the level in the reactor to permit moving fuel elements in the reactor with reduced personnel exposure. Boric acid was added, on 9/26/62 after Move 9, to meet the 8-rod shutdown requirement. Because of the instrumentation spider and straps for the in-core flux leads, considerable difficulty was experienced with Moves 19 through 36; and, in fact, the fuel element sequence originally selected could not be utilized because S-28 could not be reached. While removal of other fuel elements was relatively simple the reinsertion of the fuel elements at locations 63-337, 72-360, 77-276, 72-254, 77-64, and 72-43 was difficult. Because a fuel element is designed to be inserted from the vertical and not from an angle, considerable lateral force was exerted to insert the fuel units under the aforementioned instrumentation supports. Once inserted, force was also required to position the fuel unit in a vertical position to permit insertion of the adjacent fuel unit. It is not recommended that this procedure be followed in the future; and if fuel element moves are required in areas where interference with instrument lines and guides is encountered, the offending items should be removed from the reactor.

Additional observations to be noted were:

1. Those elements which were hard to seat and those requiring maneuvering to seat were: S-30, S-6, S-29, S-10, S-13, S-14, S-8, and S-15.
2. The ff units which were finally seated with the rigid tool after release from the coffin tool were: S-30, S-6, S-29, S-10, S-13, S-15, S-14, S-8, S-7, S-16, and S-3.
3. Strips broke off very easily on S-13, S-14, and S-15. Strips were difficult to break on S-8 and S-16. The balance of the strips took normal force to break off.
4. In general, the boron strips appeared in good condition, showed no signs of cracking, and were ductile, i.e., bowed to arcs readily. The fuel rods also appeared in good condition, they were straight, showed no indication of bowing, and had a uniform appearing oxide coating.

The final core loading and the storage pit fuel status are shown in Figs. 110R(A)-3 and 4.

Fig. 110R(A)-3. Core - Fuel Status Board (Core Load No. 63(Rev), 9/27/62)

EBWR STORAGE RACK FUEL STATUS BOARD

DATE 9/27/62

LOAD NO. 63

PREFIX LETTER CODE

N - NATURAL "U" THICK PLATES
 T - NATURAL "U" THIN PLATES
 EN - ENRICHED "U" THICK PLATES
 ET - ENRICHED "U" THIN PLATES
 SD - DUMMY STAINLESS
 SDS - DUMMY WITH SOURCE
 STSD - SOURCE TRANSFER STAINLESS DUMMY
 S - ENRICHED "U" SPIKE
 SF - SPIKE FRAME

CR 1	CR 3	CR 5	CR 7	
CR 2	CR 4	CR 6	CR 8	

																		ENS-60	T-10	M
																			ET 43	L
																				K
																				J
																				I
																				H
																				G
																				F
																				E
																				D
																				C
																				B
																				A

SF (9-4)																				
SF (9-2)																				

27 26 25 24 23 22 21 20 19 18 17 16 15 14 12 11 10 9 8 7 6 5 4 3 2 1
 SHELF

Fig. 110R(A)-4. Storage Pit - Fuel Status Board (Core Load No. 63(Rev), 9/27/62)

EXAMINATION OF 1.1% BORON-STAINLESS STEEL BURNABLE POISON STRIPS

(Test Report No. 1100 Addendum, dated November 7, 1962, by V. M. Kolba)

As part of the continuing surveillance of the EBWR reactor core, periodic examinations of various components were performed. During the preparation of the core for 100-Mw operation some boron strips were removed. This provided an opportunity to inspect representative strips as irradiation progressed. The inspection of the first such strip removed is discussed here. In summary:

1. The tack welds were physically sound and securely holding the strip to the frame.
2. No defects, such as tears or cracks, were noted on visual examination of the strip.
3. Some oxide deposit was noted on the strips.
4. The strip was ductile to a 100-deg bend.

BACKGROUND

For ~1.1% natural boron in a Type 304 stainless steel, material was ordered to standard mill tolerances for dimensions and to the following limits on chemistry:

Mn	1.5% minimum
Si	0.5% minimum

The material received from Heat No. 15314 had the following certified chemical analysis (supplied by Superior Steel Company):

C	0.070%
P	0.032%
S	0.010%
Si	0.79%
Mn	1.54%
Ni	9.52%
Cr	18.70%
B	1.10%
Fe	Balance

ANL analysis indicated a higher boron content, i.e., $B = \sim 1.31\%$. The material was received as strip stock

$$2.25^{+0.0625}_{-0.000} \text{ in. wide} \times 71 \text{ in. long} \times 0.0625 \pm 0.0025 \text{ in. thick.}$$

OPERATIONAL HISTORY

Prior to assembly the strips were cleaned with acetone; then they were tack welded to the frame and locally cleaned. The strips were not passivated. The fuel elements were placed in polyethylene bags and subsequently handled with clean white gloves until placed in the reactor core. The spike S-8, from which the strip chosen for inspection was taken, was placed in the core on May 10, 1960. The strip was removed from the spike element on February 16, 1962, after 6325 Mw-hr of reactor operation and $\sim 1\text{-}3/4$ years of exposure to stagnant reactor water at ambient room temperature conditions.

EXAMINATION

The examination of the strip was performed in two phases: first, a gross visual check; and second, a more detailed examination. The initial visual examination was conducted in the storage pit after removal of the strip from the fuel element. The strip had an activity of ~ 50 r/hr (hard and soft) at ~ 1 ft and, therefore, was examined under water. No gross defects, cracks, etc., were noted; the oxide coating appeared to be uniform, the breaks at the tack weld areas were bright indicating new metal, and the strip had a slight bend at the bottom from yielding when bent to break the tack welds, indicating ductility. The results of the phase one examinations were subsequently confirmed in phase two.

After visual examination in the storage pit, the strip was cut into identified sections of 6 in., 10 in., and 5-3/8 in. (Fig. 110O-Add.-1), placed in a coffin, and sent to the hot cells for further examination.

In the hot cells the individual pieces were again visually examined, with no defects noted. The pieces were then subjected to a gamma scan. These readings are shown in Table 100O-Add.-I and plotted in Fig. 110O-Add.-1.

Table 1100-Add.-I

SCAN RESULTS

<u>Section</u>	<u>Activity, r/hr</u>
1	0.150; 0.200; 0.300
A	1.5; 1.5; 2; 2.5; 3
4	4; 6.5; 6.5
B	8; 9; 9; 10; 10
2	10.5; 10.5; 10
C	8; 7; 5
3	2; 1; 1

Fig. 1100-Add.-1

A Scan of Gamma (γ) Activity Level of a Boron Strip Removed from EBWR Irradiated Spike Fuel Element

Dimensional measurements were taken next. However, because of the wide tolerance variation on the incoming material, the ability to detect meaningful variation by this method appears questionable. A comparison of thickness of unirradiated strips and the irradiated strips showed no appreciable change as noted:

<u>Condition</u>	<u>Plate Thickness, in.</u>
as-received strip	0.060-0.0645
irradiated strip	0.059-0.064

Considerable variation occurred in the sheared width, from strip to strip, the stock generally being on the plus side of the required dimension; therefore, these measurements were of little value.

The pieces of the strip were next cut into samples for bend, metallographic and chemical tests as shown in Fig. 1100-Add.-1.

Chemical analysis of chips from the longitudinal saw cut of the three areas of the irradiated strip (Fig. 1100-Add.-1) are shown in Table 1100-Add.-II.

Table 1100-Add.-II

ANALYSIS OF BORON-STAINLESS STEEL STRIP

<u>Sample</u>	<u>Boron, %</u>	<u>Lithium, %</u>
As received (Superior)	1.1	--
As received (ANL)	1.31/1.31	--
Irradiated 1	1.16/1.18	0.031
Irradiated 2	1.22/1.23	0.047
Irradiated 3	1.29/1.29	0.018

Bend tests were conducted on the irradiated samples using an apparatus which was previously used for evaluation of irradiated control rod materials. The results are shown in Table 1100-Add.-III. From these bend tests it can be seen that the material in location C has lost some of its ductility while that in locations A and B have retained the ductility of the as-received material. Metallographic samples have been taken (Fig. 1100-Add.-1) of several areas of interest in section 2 (Fig. 1100-Add.-2) and section 3 (Fig. 1100-Add.-3), and of as-received material (Fig. 1100-Add.-2). All samples were prepared in the same manner and etched with a 75% HCl + 25% HNO₃ solution. Irradiated samples were prepared and examined remotely in the ANL hot cell facilities.

Table 1100-Add.-III

BEND TEST DATA

<u>Specimen (Location)</u>	<u>Angle,* degrees</u>	<u>Force, lb</u>	<u>Remarks</u>
As received	100	Not recorded	No cracks
As received	180	Not recorded	Began to crack at edges
A ₁	100		No cracks
A ₁	180	134	Began to crack at edges
A ₂	100		No cracks
A ₂	180	137	Began to crack at edges
B	100		No cracks
B	190	159	Small cracks at edges
C ₁	100		No cracks
C ₁	160	137	Sample broke in two
C ₂	100		No cracks
C ₂	110	147	Sample broke in two

* Pin diameter for all bends is 1/2 in.

AS RECEIVED
TRANSVERSE
500 x

IRRADIATED SECT. 2
TRANSVERSE
500 x

IRRADIATED SECT. 2
LONGITUDINAL
500 x

Fig. 110O-Add.-2. Metallographic Samples (As Received and Irradiated Section 2)

TRANSVERSE
500 x

PARENT METAL - WELD
TRANSITION ZONE
500 x

WELD ZONE TRANSITION
500 x

WELD ZONE
500 x

Fig. 110O-Add.-3. Metallographic Samples (Irradiated Section 3)

MEASUREMENT OF WATER CHANNELS ON EBWR CORE 1 FUEL ELEMENTS

(Test Report No. 114, dated September 22, 1961,
by C. F. Reinke and W. N. Beck)

SUMMARY

The water channel gaps on EBWR fuel elements ET-15, ET-25, and ET-43 were measured with a modified commercial eddy current detection device. The channels ranged from 0.330 in. to greater than the reference value of 0.438 in. The values are tabulated in Tables 114-I to III.

PROCEDURE AND DISCUSSION

The response of the measuring probe is affected by temperature; consequently, the temperature within a channel was measured with a chromel-alumel thermocouple. As there was no detectable gradient along a channel, the probe was calibrated under simulated core conditions using:

1. A standard, consisting of two sections of EBWR fuel plate whose spacing was adjustable to micrometer settings equivalent to the known separation between plates.
2. The measured temperature of the water channel.
3. Go-no-go gauges.

After calibration, the equipment was mounted on the indexing plug at a point above a positioning rack holding the EBWR fuel element to be examined. The probe holder was indexed over and driven into the channel to be measured. The signal from the probe, as it traversed a channel, was continuously recorded on suitable chart paper. Duplicate runs, on some channels, indicated that reproduction of the measurements was good. Relative variations along each channel were immediately apparent, and actual water channel spacings were obtained by comparison of the chart values with the values obtained from the known standard. The values reported in Tables 114-I to III were determined at 2-in. increments along the channel length.

The estimated accuracy of the reported values is ± 0.010 in. The variation in the values was more pronounced in the irradiated fuel elements as compared to the variation in the values of an unirradiated fuel element. The scale deposits on the irradiated elements are considered to have resulted in the observed fluctuations in the measured values.

Table 114-I

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-15

Distance from Top, in.	Channel*				
	1	2	3	4	5
4	> 0.438	0.438	> 0.438	0.422	> 0.438
6	0.426	0.413	0.438	0.407	
8	0.415	0.413	0.435	0.402	
10	0.438	0.420	0.426	0.404	
12	0.432	0.413	0.416	0.404	
14	0.413	0.407	0.426	0.404	
16	0.410	0.404	0.420	0.410	
18	0.430	0.413	0.432	0.410	
20	0.407	0.399	0.426	0.413	
22	0.399	0.396	0.426	0.406	
24	0.385	0.384	0.422	0.413	
26	0.385	0.381	0.416	0.416	
28	0.379	0.377	0.404	0.416	
30	0.367	0.384	0.394	0.396	
32	0.367	0.367	0.394	0.402	
34	0.365	0.371	0.390	0.396	
36	0.363	0.365	0.390	0.396	
38	0.363	0.362	0.399	0.399	
40	0.363	0.369	0.399	0.399	
42	0.360	0.367	0.396	0.394	
44	0.369	0.384	0.399	0.394	
46	0.363	0.367	0.399	0.388	
48	0.381	0.377	0.404	0.407	
50	0.377	0.371	0.407	0.399	↓

* Channel No. 5 was designated as the one below the identification tab in the top end fitting. The remainder of the channels are referenced from channel No. 5.

Table 114-II

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-43

Distance from Top, in.	Channel*				
	1	2	3	4	5
4	0.363	0.378	0.363	0.388	0.375
6	0.362	0.372	0.365	0.370	0.368
8	0.359	0.372	0.365	0.368	0.363
10	0.363	0.376	0.365	0.371	0.369
12	0.365	0.378	0.382	0.374	0.365
14	0.365	0.400	0.369	0.380	0.369
16	0.369	0.407	0.367	0.374	0.371
18	0.369	> 0.438	0.366	0.380	0.372
20	0.383	> 0.438	0.365	0.382	0.383
22	0.400	> 0.438	0.378	0.388	0.405
24	> 0.438	0.433	0.376	0.388	0.369
26	0.369	0.383	0.381	0.391	0.359
28	0.362	0.388	0.357	0.394	0.354
30	0.355	> 0.438	0.378	0.400	0.374
32	0.363	> 0.438	0.383	0.402	0.354
34	0.350	> 0.438	0.380	0.404	0.351
36	0.352	0.373	0.383	0.405	0.351
38	0.369	0.397	0.380	0.413	0.369
40	0.365	0.400	0.382	0.405	0.374
42	0.363	> 0.438	0.378	0.402	0.359
44	0.362	> 0.438	0.374	0.400	0.353
46	0.388	> 0.438	0.373	0.388	0.357
48	0.400	> 0.438	0.371	0.383	0.388
50	0.388	> 0.438	0.369	0.388	0.402

* Channel No. 5 was designated as the one below the identification tab in the top end fitting. The remainder of the channels are referenced from channel No. 5.

Table 114-III

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-25

Distance from Top, in.	Channel *				
	1	2	3	4	5
4	>0.438	0.344	0.340	> 0.438	0.363
6		0.343	0.335		0.362
8		0.341	0.335		0.363
10		0.349	0.338		0.378
12		0.343	0.340		0.423
14		0.359	0.347		>0.438
16		0.351	0.346		
18		0.357	0.346		
20		0.362	0.346		
22		0.374	0.355		
24		0.365	0.331		
26		0.355	0.343		
28		0.355	0.349		
30		0.338	0.365		↓
32		0.343	0.361		0.419
34		0.341	0.365		0.403
36		0.333	0.335		0.396
38		0.349	0.336		0.406
40		0.335	0.331		0.399
42		0.338	0.340		0.423
44		0.378	0.329		0.395
46		0.365	0.349		0.386
48		0.348	0.347		0.413
50	↓	0.344	0.346	↓	0.399

*Channel No. 5 was designated as the one below the identification tab in the top end fitting.
The remainder of the channels are referenced from channel No. 5

The charts are to be retained and compared with others made periodically throughout the life of the Core 1 elements after additional burnup and elevated temperature operation is obtained.

CONCLUSIONS

A comparison of channel measurements taken periodically, throughout the Core 1 lifetime, will yield data on the relative and absolute change in water channels as a function of reactor operation; and, as such they can be used to determine any elevated temperature swelling and detection of possible trouble spots.

MEASUREMENT OF WATER CHANNELS ON EBWR CORE 1 FUEL ELEMENTS

(Test Report No. 114A, dated January 25, 1963,
by C. F. Reinke, W. N. Beck, and W. A. Ahrens)

SUMMARY

The water channel gaps on EBWR fuel elements ET-2, ET-5, ET-11, and ET-31 were measured, with a modified commercial eddy current detection device, prior to and following 100-Mw operation. Prior to high power operation the channels ranged from about 0.370 in. to greater than the reference value of 0.438 in. After 100-Mw operation the minimum water gap was found to be about 0.400 in. The data indicate that swelling, if indeed any occurred, was negligible and masked by a loss of oxide from the fuel plate surfaces. The measured water channel values are tabulated in Tables 114A-I to IV.

PROCEDURE AND DISCUSSION

The response of the measuring probe is temperature dependent; consequently, the temperature within a channel was measured with a chromel-alumel thermocouple.

As there was no detectable gradient along a channel the probe was calibrated under simulated core conditions using:

1. A standard, consisting of two section of EBWR fuel plate whose spacing was adjustable to micrometer settings equivalent to the known separation between plates.
2. The measured temperature of the water channel.
3. Go-no-go gauges.

After calibration, the equipment was mounted on the indexing plug at a point above a positioning rack holding the EBWR fuel element to be examined. The probe holder was indexed over and driven into the channel to be measured. The signal from the probe, as it traversed a channel, was continuously recorded on suitable chart paper. Duplicate runs, on some channels, indicated that reproduction of the measurements was good. Relative variations along each channel were immediately apparent, and actual water channel spacings were obtained by comparison of the chart values with those values obtained from the known standard. The values reported in Tables 114A-I to IV were determined at 2-in. increments along the channel length.

Table 114A-1

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-2
BEFORE AND AFTER 100-Mw OPERATION

Dist. From Top, in.	Channel 1		Channel 2		Channel 3		Channel 4		Channel 5	
	Before	After	Before	After	Before	After	Before	After	Before	After
4	≥ 0.432	≥ 0.432	≥ 0.432	0.430	0.430	≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432
6			0.416	0.424	0.424	0.424				
8			≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432				
10					≥ 0.432	≥ 0.432				
12					0.430	0.424				
14					≥ 0.432	0.428				
16			▼		0.430	≥ 0.432				
18			0.416		≥ 0.432					
20			≥ 0.432							
22			0.428							
24			≥ 0.432			▼				
26						0.420				
28						0.430				
30				▼	▼	0.408				
32				0.414	0.412	≥ 0.432				
34				0.428	0.414	0.424				
36				≥ 0.432	≥ 0.432	≥ 0.432				
38					≥ 0.432					
40	▼				0.414					
42	0.428				0.404					
44	≥ 0.432				≥ 0.432					
46					≥ 0.432					
48			▼		0.428					
50	▼	▼	0.428	▼	0.428	▼	▼	▼	▼	▼

Table 114A-II

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-5
BEFORE AND AFTER 100-Mw OPERATION

Dist. From Top, in.	Channel 1		Channel 2		Channel 3		Channel 4		Channel 5	
	Before	After	Before	After	Before	After	Before	After	Before	After
4	0.399	≥ 0.432	0.394	0.428	0.428	≥ 0.432	0.414	≥ 0.432	0.384	0.414
6	0.399		0.378	0.428	≥ 0.432		0.399		0.378	≥ 0.432
8	0.414		0.384	≥ 0.432			≥ 0.432		0.396	
10	≥ 0.432		0.394	≥ 0.432					0.420	
12	≥ 0.432		0.373	≥ 0.432					0.399	
14	≥ 0.432		0.368	0.396					0.399	
16	0.428		0.373	0.420			▼		0.404	
18	0.428			0.430			0.428		0.396	
20	0.412			0.414			≥ 0.432		0.399	
22	0.430		▼	≥ 0.432			≥ 0.432		0.386	
24	0.428		0.416				0.414		0.404	
26	≥ 0.432		0.408				≥ 0.432		0.391	
28			0.389						≥ 0.432	
30			0.384						≥ 0.432	
32			0.405	▼					0.391	
34			0.402	0.428					0.389	
36			0.386	0.420					0.414	
38			0.378	0.428					0.380	
40			0.382	≥ 0.432					0.384	
42			0.424				▼		0.378	
44	▼		0.386				0.408		≥ 0.432	
46	0.414		0.384				≥ 0.432		0.416	
48	0.416		≥ 0.432				≥ 0.432		≥ 0.432	
50	0.404	▼	≥ 0.432	▼	▼	▼	≥ 0.432	▼	0.414	▼

Table 114A-III

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-11
BEFORE AND AFTER 100-Mw OPERATION

Dist. From Top, in.	Channel 1		Channel 2		Channel 3		Channel 4		Channel 5	
	Before	After	Before	After	Before	After	Before	After	Before	After
4	0.408	≥ 0.432	0.414	0.416	≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432	≥ 0.432
6	≥ 0.432		0.408	≥ 0.432					0.412	
8			0.428						≥ 0.432	
10			≥ 0.432							
12			≥ 0.432							
14			≥ 0.432							
16			0.428							
18	▼		≥ 0.432						▼	
20	0.416								0.420	
22	≥ 0.432								≥ 0.432	
24										
26	0.428									
28	≥ 0.432									
30	0.408									
32	≥ 0.432								▼	
34									0.420	
36									0.406	
38									0.428	
40									≥ 0.432	
42									0.398	
44			▼						0.394	
46			0.430						0.416	
48			≥ 0.432						0.402	
50	▼	▼	≥ 0.432	▼	▼	▼	▼	▼	0.396	▼

Table 114A-IV

WATER CHANNEL DIMENSIONS ON EBWR FUEL ELEMENT ET-31
BEFORE AND AFTER 100-Mw OPERATION

Dist. From Top, in.	Channel 1		Channel 2		Channel 3		Channel 4		Channel 5	
	Before	After	Before	After	Before	After	Before	After	Before	After
4	0.391	≥ 0.432	0.402	≥ 0.432	0.414	≥ 0.432	0.420	≥ 0.432	≥ 0.432	≥ 0.432
6	0.384		0.406		≥ 0.432		0.430	≥ 0.432		
8	0.399		≥ 0.432				≥ 0.432	0.428		
10	0.402									
12	0.414									
14	0.414									
16	0.428	▼								
18	0.416	0.430								
20	0.399	0.414								
22	0.378	0.399								
24	0.378	0.420								
26	0.378	0.402								
28	0.380	≥ 0.432								
30	0.399	≥ 0.432								
32	0.420	≥ 0.432								
34	0.396	0.430								
36	0.402	≥ 0.432								
38	0.428	0.424								
40	0.428	≥ 0.432								
42	0.420									
44	0.416	▼								
46	0.416	0.402								
48	0.402	0.412								
50	0.380	0.408	▼	▼	▼	▼	▼	▼	▼	▼

Duplicate runs, made on at least one channel in each fuel element, indicated a reproducibility of ± 0.010 in. in the measurement of channels having gaps of about 0.400 in. The probe was designed to provide maximum accuracy with water channels of approximately 0.350 in.; thus, the exponential shape of the calibration curve introduces more uncertainty in the accuracy of the readings, when the water channels exceed 0.400 in.

A series of ten calibration runs, with water temperatures of between 18°C to 25°C indicated that within that temperature range the response of the probe was constant.

CONCLUSIONS

The data indicate that swelling, if indeed any occurred, was negligible and masked by a loss of oxide from the fuel plate surfaces.

ANALYSIS OF SCALE FROM EBWR FUEL PLATES

(Test Report No. 112A, dated January 21, 1963, by W. G. Knapp)

INTRODUCTION

Subsequent to the first operation of the EBWR at powers approaching 100 Mw, the head was removed from the reactor vessel for further destripping of spike fuel assemblies and other work. During this time selected plate-type assemblies were lifted from the core for visual examination and for the obtaining of samples of scale for chemical analysis. Fuel assemblies ET-12 and ET-53 were examined. Samples were obtained in satisfactory quantities from one outer surface of each of these assemblies as shown in Fig. 112A-1.

In addition, a sample of scale which had flaked off the fuel assemblies and had collected in the blowdown tank was obtained.

The samples were submitted for chemical analysis. The results and a discussion of that analysis are given below.

COLLECTION OF SAMPLES

Scale samples were collected from the fuel assemblies by means of a long-handled tool specifically designed for this purpose. The tool contained a plastic scraping edge mounted on a stainless container in such a way that scale, scrapped from the fuel assemblies, would drop into the container and thus, be removed from the reactor. Removal of scale, from the examined assemblies, was attempted only on the outer plate surfaces from the areas which had exhibited evidence of the flaking during high power operations. Though some sample material was obtained from all four of the surfaces indicated in the figure, samples of sufficient size for chemical examination were obtained only from surfaces A and D in the figure.

As a result of high power operation, considerable quantities of scale had flaked off the assemblies and had, in part, collected in the blowdown tank. Samples, with the obvious flake or scale characteristic, were collected from this tank. Due to the high activity levels of this material, the sample size was kept to a few milligrams. All samples were identified, placed in containers, and kept in the reactor building until dry.

A few days after collection of the various samples, including that from the blowdown tank, the thickness of the samples was measured with a

hand micrometer. Those samples collected from the fuel assemblies measured 0.10 to 0.11 mm (0.004-0.0045 in.), and those from the blowdown tank measured about 0.15 mm (0.006 in.).

Since the samples were taken from water containing boric acid, this was removed, prior to submitting the samples for chemical analysis, by careful repeated washings with demineralized water at room temperature.

A preliminary spectrographic analysis was requested with particular interest in aluminum-iron-nickel ratios and in boron content. The results were as follows (factor of two):

	% <hr/>						
	<u>Al</u>	<u>Fe</u>	<u>Ni</u>	<u>Cr</u>	<u>Mg</u>	<u>Mn</u>	<u>B</u>
ET-12 (north side)	12	12	30	0.08	0.07	0.2	<0.1
ET-12 (south side)	10	10	18	0.06	0.03	0.2	<0.1
ET-53 (south side)	10	10	10	0.07	0.05	0.2	<0.1
Blowdown	very strong	8	10	0.05	0.05	0.2	<0.1

Due to the small sample size, chemical analyses were performed by the Chemistry Division on a semi-micro scale for the following elements: B, Fe, Ni, and Al, with results as follows:

<u>Sample</u>	B (as H_3BO_3), ppm	% <hr/>		
		<u>Fe</u>	<u>Ni</u>	<u>Al</u>
From ET-12 (north side)	1980	12.65	28.51	13.9
From ET-53 (south side)	1500	12.0	27.5	6.5
From Blowdown	462	8.51	13.45	31.01

Since boric acid is fairly soluble at room temperature, it is safe to assume that the free boric acid would be removed from the porous and readily wettable scale during the thorough wash with demineralized water. The boron (or borate) reported here must be assumed to be permanently fixed in the scale. It is certainly safe to assume that this is true of the samples scrapped from the fuel assemblies.

DISCUSSION

From the chemical analyses several conclusions may be drawn, as follows:

1. The sample taken from the blowdown tank was accumulated on one or more fuel plates primarily during reactor operation prior to a 100-Mw conversion as evidenced by the high aluminum content.

This does not eliminate the possibility of leaching of aluminum from scale in the high flux zone. However, the solubility of dehydrated aluminum oxide is very low or negligible, and the tendency of this oxide to become soluble through hydration is also very low particularly with the presence of the high temperature as would be experienced during reactor operations.

2. Though the aluminum content on assembly No. ET-53 was only approximately 50% of that on the ET-12 assembly, the iron/nickel ratio for both samples was 0.44, while the same ratio on the blowdown scale sample was 0.63, which would indicate nearly identical "scale histories" for the two fuel assembly locations, and a different history for the blowdown sample.
3. There is probably a limited degree of permanent deposition of boric acid in the fuel plate scale, though not enough to significantly affect the reactivity of the core. It would likewise appear that the boric acid is less readily deposited in the old scale (i.e., the scale with the high aluminum content) than it is, coincidentally, with the deposition of fresh scale (i.e., that with high iron and nickel content).
4. It would appear that the high iron and nickel content of the freshly deposited scale such as scrapped from the two fuel assemblies, is quite possibly due to the larger amount of stainless steel and Inconel area exposed to the reactor water at high powers as compared to that during the relatively low power operation of the reactor prior to 100-Mw modification. At the present high power operation, the water level is much higher than it was during previous operations. The great amount of agitation of the water at this higher power causes the water to make frequent contact with the inside surface of the entire reactor vessel and, due to the high carryover rates, with much of the external steam system. The large amount of experimental equipment that is now present, which was not present during previous operation, would likewise contribute corrosion products to the reactor water.

All indications point to a very dirty reactor during the following high power operations when, due to the high water velocities and high temperature of the fuel plates, crud or scale flaked off the fuel plates. During visual examination, it was noticed that the side plates of the plate-type fuel assemblies showed some buckling at intervals between the spot welds. This would indicate that the plates expanded beyond the elastic limit of the side plates, during high power operation, and then contracted again to a dimension near their previous size. This expansion-contraction phenomenon could likewise have been the cause of the descaling effect. Since the observed descaling on the examined fuel assemblies occurred in the same region as that in which

the buckling of the side plates occurred, this would lead to the logical conclusion that this phenomenon of expansion-contraction contributed to the descaling that occurred.

Not only was the blowdown tank loaded with crud and scale, but the full flow filters were highly radioactive, the reactor water was murky with reddish-colored finely divided crud, and the assemblies showed areas where the scale had flaked off. Under these conditions it is highly possible, even probable, that the scale samples taken from the high flux zone of these two assemblies were deposited very recently - following flaking off or descaling of a previous deposit. In fact, this phenomenon could have been repeated more than once and could account for the sharply defined bands or striations of color that made up the affected zone of the assemblies visually inspected.

Previous studies of scale deposition on EBWR fuel assemblies assumed that scale would only continue to build up. We have now observed what appears to be a self-cleansing action that should be a valuable asset in economical reactor design and should warrant further study. Additives to promote flakable scale formations could possibly result in controlled self-cleansing, even at lower power operations.

DETECTION OF POSSIBLE GROWTH IN THIN ENRICHED (ET) FUEL PLATES OF EBWR WITH HIGH BURNUP EXPOSURE

(Test Report No. 107, dated January 31, 1963,
by H. Iskenderian and J. Haugsnes)

The purpose of this test report is to indicate the results obtained by a 1/4-in. diameter, G.E. "B" detector which was placed in the water channel of an ET element for detection of possible fuel growth. This fuel element (ET-2) had a high burnup of $\sim 0.4\%$, and was located in a high flux region (see Fig. 107-1).

It was shown by 2-group DSN calculations that the proposed "B" detector would indicate if growth of fuel plates surrounding the detector occurred. The possibility of excessive fuel growth had been estimated to occur at power levels of about 100 Mw due to possible fuel plate temperatures of $\sim 600^{\circ}\text{C}$ as a result of $\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$ scale on the fuel plates. The "B" detector would, in addition, indicate variations in "local" power, as the reactor power was varied.

Two "B" detectors were installed in the locations indicated on Fig. 107-1; one of these detectors became inoperative due to loss of electrical insulation. The remaining "B" detector was used as a direct power measuring instrument.

THE FLUX MONITORS WERE PLACED IN THIN ENRICHED FUEL ELEMENT ET-2 IN CORE LOCATION 89-127.

THE CENTERS OF THE FLUX MONITORS WERE 61.0 cm ABOVE THE BOTTOM OF THE CORE.

Fig. 107-1. Installation of Incore Flux Monitors.

Fig. 107-2. "B" Detector Versus Power Characteristics.

$$P \approx \frac{1.02D}{1 + 0.0026D}$$

(D = Detector Current in μ A
P = Power in Mw)

RESULTS OF MEASUREMENTS

At no time during the high power runs did the "B" detector indicate the effects of possible growth of fuel, or loss of water in the channel where the detector was located. (It was subsequently noted that there was a loss or flaking off of scales from the fuel plates, after the high power runs. This was possibly due to a loss of the water of crystallization from $\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$. This fortunate circumstance would enable these fuel elements to operate at relatively low temperatures.)

The readings of the "B" detector were nearly proportional to the reactor power at low power levels, but they deviated at higher powers. During the 100-Mw power runs (see Fig. 107-2), it was noted,

$$P = \frac{1.02D}{1 + 0.0026D} \quad (1)$$

where:

P denotes the reactor power in Mw,
D is the detector current in μ amp.

Section III. PHYSICS

<u>Title</u>	<u>Page</u>
Analysis of EBWR Startup Transients (Test Report No. 120A, dated February 1, 1962, by K. Almenas, N. Suda, and N. Ito)	133
Temperature and H_3BO_3 Effect on EBWR Subcritical Counting Rate (Test Report No. 108C, dated June 19, 1961, by K. Almenas and T. Nomura)	163
Experimental EBWR Temperature Coefficients (Test Report No. 108D, dated June 30, 1961, by K. Almenas and J. Haugsnes)	174
Void Coefficient Measurements in EBWR (Test Report No. 124, dated August 27, 1962, by H. P. Iskenderian)	187
Relative Reactivity Measurements of EBWR Fuel Elements (Test Report No. 100D, dated December 21, 1961, by K. Almenas and V. Kolba)	194
Reactivity Effect of Rotating the Boron-Stainless Steel Strip Containing Spike Elements (Test Report No. 100F, dated November 24, 1961, by K. Almenas)	208
Completion of Loading 1A, with the Addition of Four Peripheral Spikes and One ET Element - Destripping of Eight Spikes by One Strip Each - Verification of the Reactor Meeting the Cold Shutdown Requirements - A Partial Calibration of the Control Rod System (Test Report No. 100J, dated March 8, 1962, by H. P. Iskenderian)	215
Physics Termination Tests on EBWR, Core Loading 1A (Test Report No. 100N, dated January 14, 1963, by H. P. Iskenderian)	220
Reactivity Aspects of EBWR Core 1B (Test Report No. 100H, dated February 1, 1962, by K. Almenas)	226
Destripping of 16 More Spikes for a Gain of $\sim 2\%$ Reactivity - Check of 9-Rod Cold Shutdown Requirement (Test Report No. 110R(B), dated October 8, 1962, by H. P. Iskenderian)	231
EBWR Neutron Flux Distribution during Cold Critical Operation (Test Report No. 100K, dated October 23, 1962, by W. G. Knapp)	235
Flux Distribution Measurements in EBWR during Operation (Test Report No. 100P, dated January 22, 1963, by W. G. Knapp)	258

Section III. PHYSICS (Cont'd.)

<u>Title</u>	<u>Page</u>
Study of Operational Data from EBWR at 0-60-Mw Levels - $\Delta k / \Delta P$ Increasing with Power (Test Report No. 100M, dated October 1, 1962, by H. P. Iskenderian)	263
EBWR Reactor Transfer Function Measurements (Test Report No. 123, 123A, 123B, and 123C, dated January 31, 1963, by W. C. Lipinski, A. Hirsch, C. Hsu, and G. Popper).	268

ANALYSIS OF EBWR STARTUP TRANSIENTS

(Test Report No. 120A, dated February 1, 1962,
by K. Almenas, N. Suda, and N. Ito)

SUMMARY

EBWR startup transients for two core conditions (poison free and with 3.66 gm of H_3BO_3 / gal of reactor water) are analyzed in detail and reproduced with an analog model. A general energy-reactivity coupling equation involving natural circulation is proposed and analyzed. The specific version of the model used to simulate the power transients is not quite satisfactory and the properties of the desired final model are described and discussed. General correlations between $P_{(tm)}$ (maximum power during a transient), $E_{(tm)}$ (energy generated up to maximum power), and $1/T_0$ (the reciprocal initial startup period) are presented.

INTRODUCTION

A large amount of experimental and theoretical work investigating the self-limiting nature of boiling-water-reactor power excursions has been performed.^(1,2,3) The prime interest in all of these investigations was reactor safety, and accordingly, the studies were carried out under simulated accidental conditions, that is, the step and ramp-induced power periods studied varied from 10 sec well into the millisecond range. Such periods will not be encountered under normal reactor operations, and thus we have a situation where the most common, what one could call standard, operating reactor transients are the least well understood. There is one significant difference which complicates and distinguishes the longer period transients; namely, whereas for the short period transients the energy generated stays in the core and raising the core temperature serves to terminate the excursion, for the longer period excursions (< 2 sec), part or even most of the energy is removed from the core and does not contribute to the shutdown mechanism. This effect can be very significant as illustrated, for example, by the heat capacities of the EBWR core and included water which is 5.6 Mw-sec/ $^{\circ}\text{C}$ and the heat capacity of the total water content in the EBWR vessel which is ~ 60 Mw-sec/ $^{\circ}\text{C}$. Thus, it makes a very large difference whether the energy generated during a power excursion serves to heat up the core alone or, through well established natural circulation, heats up the total water contained in the reactor vessel.

Besides the general academic interest, there are two good practical reasons why, notwithstanding the additional complications, the longer period transients should be studied and understood.

1. Safety

Understandably, power reactors cannot be operated as prompt critical facilities; thus, the estimated fast period behavior of such reactors will remain academic (barring an accident) unless methods are found to extrapolate from the longer to the shorter periods. Since, for periods just short of prompt critical, the shutdown mechanism is practically identical, one would have only to understand and be able to eliminate the effect of energy removal in order to extrapolate into the short period range. This would put accident evaluation on a semi-experimental basis.

2. Convenience

The startup transient provides the most common and easily obtained data regarding the dynamic behavior of the reactor, further, since such transients will be run as part of the operating procedure all through reactor life, the long-term change in such important parameters as temperature coefficient and delayed neutron fraction could be monitored by means of these transients.

PROCEDURE

Two series of tests (A and B) were conducted for each of three start-up transients. The conditions for series A included a clean core with the control rod bank positioned ~ 19 in. from the bottom of the core. These conditions maximize the temperature coefficient since the control rods contribute a strong negative coefficient. In series B, conversely, the attempt was made to achieve conditions of the lowest possible temperature coefficient. For this purpose 3.66 gm of H_3BO_3 were added per gallon of reactor water and the critical control rod bank position was increased to ~ 35 in. from the bottom of the core.

The startups were conducted in a standard way except that pains were taken to insure equilibrium core conditions prior to startup and that additional instrumentation was used. The startup periods postulated were 15, 25, and 40 sec and in practice did not depart more than ± 1 sec from the desired values.

Specifically, the procedure used is as follows:

1. The reactor was brought to a bank critical with reactor power at ~ 10 kw.
2. Rod No. 9 was inserted and the flux allowed to decay.
3. The control rod bank was raised by an increment dependent upon the desired period.

4. Rod No. 9 was brought up flush to the control rod bank, making the reactor supercritical.

Subsequently, the control rods were not touched for 15-20 min. The power was allowed to rise and even-out by itself. The total power rise was ~ 7 decades above the source level, and since the energy-reactivity feedback did not become appreciable until the last decade, there was ample time for an asymptotic period to be reached and timed. Consequently, the reactivity insertion due to movement of rod No. 9 could be considered as a step function.

The parameters monitored were: power, core inlet temperature, inlet and outlet flows of the instrumented fuel assembly, and pressure. The power traces as measured by the L & N recorder (activated by a fission chamber located just outside the reactor vessel) are shown in the odd numbered Figs. 120A-1 to 17. A Sandborn recording of flow rate, temperature, and power for test 3-B is given in Fig. 120A-18.

POWER MEASUREMENT

As pointed out in Reference 4, the neutronic instrument response will vary not only with power (neutron flux) but also with core changes such as temperature rise, poison addition, and control rod movement. At operating conditions, the neutronic instruments can be normalized to read in terms of power from such direct power measurements as steam and feedwater flow. However, this could not be done during the transient conditions of the test since the powers reached were too low. Further, even full power measurements, at the test condition ($\sim 160^\circ\text{C}$), could not be scaled down sufficiently since they necessarily would have been obtained with voids in the core, while under the test conditions none or extremely few voids were present. Thus, it was necessary to obtain a separate power normalization (because of the different core conditions), for test series A and B. For this purpose, the pressure rise occurring after equilibrium conditions have been attained ($> 300 \text{ Mw-sec}$) was found to yield the most consistent results. An attempted check utilizing the temperature recording was unsuccessful. The reason for this is that only a local temperature, and possibly an atypical one, was recorded. Since the conditions for a series of tests remained constant, the relative powers within a series were represented accurately, the extent of uncertainty concerns only the absolute value of the power which could amount to $\sim 20\%$ (estimated from the discrepancy of several calculational approaches.) A direct demonstration that the absolute powers might be in error is presented in Fig. 120A-19 which gives the temperature versus energy plots for the six tests. As seen in the figure, the slopes in the asymptotic region for the series A tests differ from those for series B. This, however, is not a proof of discrepancy, since the temperature measured is only a local one and the difference could also be attributed to changed flow patterns (because of gross control rod and power distribution shifts) between the series A and B

tests. Additional temperature instrumentation or possibly in-core neutronic instruments should improve the absolute power measurements in future tests.

SHUTDOWN MODELS

General Approach to Problem

The purpose of this analysis is to find the simplest possible mathematical model which would reproduce the experimentally determined positive period transients. A semi-theoretical "curve-fitting" approach is used. That is, simple but physically plausible energy-reactivity coupling equations are stipulated and used in conjunction with the kinetics equations to generate power transients. The comparison of the model with the experiment matches such important physical features as the maximum power, energy generated up to maximum power, and the initial and final energy-reactivity coefficients. Ideally, it would be desirable to find a model with just one empirically fitted parameter, so that from a single experiment, for a certain period, the model could be normalized and subsequently generate the power trace for any other period. This method was employed fruitfully for the analysis of fast period transients.^(1,2,3) However, the models developed in those studies cannot be used without modifications since they do not consider energy removal from the core.

EBWR Heat Capacity Systems

For the present analysis, the EBWR is considered as being composed of a sequence of concentric heat capacity systems, each distinguished by its characteristic time delay. This model is best illustrated by considering power transients of widely differing periods.

1. Millisecond Range - For periods in the millisecond range, no significant amount of heat is transferred to the reactor water, and the energy generated serves to raise the temperature of the fuel matrix and clad only. The fuel matrix and clad are here referred to as heat capacity system I.
2. Second Range - In units of time, on the order of seconds, significant amounts of heat are transferred to the water immediately adjacent to the fuel plates. Thus, the energy being generated enters into a system with a larger heat capacity; namely, into the fuel matrix, clad, and core water (system II), and more energy per degree of temperature rise is required.
3. Tens of Seconds - In tens of seconds a temperature rise in the water will be followed by natural circulation which will carry energy away from the core. Under equilibrium natural

circulation, the generated energy will be absorbed by the whole core-core water-vessel water system (system III), the heat capacity of which is an order of magnitude larger than that for systems I and II.

4. Tens of Minutes - In tens of minutes the reactor vessel, shielding plug, and associated shielding will absorb significant amounts of heat. Heat can be transferred by conduction only into these massive metal parts. Thus the times associated with this heat capacity system (system IV) are correspondingly larger.

A summary of the characteristics of the four heat-capacity systems is presented in Table 120A-I. The required calculations are shown in Attachment A.

Table 120A-I

EBWR HEAT-CAPACITY SYSTEMS

System No.	Description	Heat Capacity, Mw-sec/°C	Time Delay	Energy Required to Shutdown $k_{ex} = 1\%$, Mw-sec
I	Core metal (clad + fuel matrix)	1.17	0	1
II	Core metal + core water	5.56	~0.25 sec	4.2
III	Core metal + core water + vessel water	63	~5 sec	48
IV	Core + core and vessel water + vessel	108	~10 min	80

In Table 120A-I the column labeled "Time Delay" does not give strictly defined time constants, since calculations and even definitions of a time constant are difficult for the larger systems, but rather, quantities which are proportional to the time constants. Also listed in the table is the amount of energy required to compensate 1% of k_{ex} if the temperature coefficient is $-0.1\%/^{\circ}\text{C}$ (a theoretically calculated value for EBWR at 170°C) and if the energy generated serves to raise uniformly the temperature of the system in question.

As seen, especially in the "Time Delay" column, the systems differ by orders of magnitude. The well studied short periods (> 2 sec SPERT work⁽³⁾) encompass systems I and II and their interrelation. The periods analyzed in this report deal with the transition (establishment of natural circulation) from system II to system III. For extremely long periods, system IV would also come into play, but in the present report this is disregarded.

Evidence for Importance of Natural Circulation

Significant natural circulation can be established in a surprisingly short time, for example in the SPERT data⁽³⁾ the power traces for periods of < 2 sec show fluctuations which can only be due to circulation of the core water. The SPERT analysis circumvents this problem by concentrating efforts on the shorter periods and by considering only the initial power peak for the long period transients. Since, in that case, the temperature coefficient was significantly larger (on the order of $-10\text{¢}/^\circ\text{C}$ instead of $\sim 1\text{¢}/^\circ\text{C}$), the total energy generated by the time peak power reached was ~ 3 Mw-sec, and it can be argued (though it was not demonstrated) that only systems I and II (the core and core water) absorb that energy. This is clearly not the case for EBWR. For example, even for the 15-sec period, ~ 150 Mw-sec were introduced into the system at peak power (see Table 120A-II) and if this energy had been absorbed by system II only, the temperature of the core and core water would have risen $\sim 30^\circ\text{C}$. Such a temperature rise would have been followed by "significant" natural circulation within ~ 2 sec. That natural circulation sets in before the peak power is reached is shown in Fig. 120A-18. The graph labeled "Flow Rate" is a qualitative indication of flow (the flowmeters are calibrated to measure flow at operating conditions when water velocities are much higher). As seen in the figure, flow is definitely measured 60 sec before the peak power when the energy input has been ~ 40 Mw-sec. Calculated on the basis of system II again, such an energy input would raise the temperature of the core water $\sim 7^\circ\text{C}$ which, in turn, would lead to significant circulation within ~ 4 sec. The conclusion thus is that for periods of 15 sec, and probably quite shorter (down to 2-3 sec), natural circulation will start in the EBWR before the peak power is reached and is therefore an important and integral part of the shutdown mechanism.

General Model

An energy-reactivity coupling equation successfully used for the short period SPERT transients was :

$$\frac{1}{\phi(t)} \frac{d\phi(t)}{dt} = \frac{1}{T_0} - \gamma \left\{ \int_0^t \phi(t) dt \right\}^n = \frac{1}{T_0} - \gamma E^n \quad (1)$$

where:

ϕ = neutron flux = power

T_0 = initial asymptotic period

n = an empirically fitted constant found to be ~ 2 .

The reactivity feedback in this model is proportional to energy raised to the n -th power. Both of the parameters γ and n were fitted empirically: n was found to be quite constant for a number of different SPERT cores and for the short periods measured it was ~ 2 ; γ , of course, was heavily dependent on the core in question (i.e., the temperature coefficient of the five cores varied from -0.25 to $-1.8\text{ }^\circ\text{C}^{-1}$ at 20°C) and also on the period. The later dependency is due to the fact that the periods which were tried (millisecond to second range) involve heat capacity systems I and II and the transition region between them. Thus the shutdown coefficient is dominated by metal expansion for the extremely short period transients and by water expansion for the longer period transients.

For the present analysis it is assumed that γ applies only to system II (core + core water). The expansion of water outside of the core is assumed not to contribute any reactivity effects. With such a picture, the model must be modified to account for the removal of energy from system II. The general form of the modified equation is

$$\frac{1}{\phi} \frac{d\phi}{dt} = \frac{1}{T_0} - \gamma \int_0^t F(E) \phi(E) dt \quad (2)$$

where: $F(E)$ is the fraction of the generated power remaining in the core.

Since the energy removal mechanism is dependent upon the temperature rise of the core water, it appears logical that F should be a function of $E(t)$, the energy generated up to time t .

In this model, n is taken to be equal to 1, since conversely there would be too many parameters to be fitted. Further, since $F(E)$ is an unspecified function, dependent on E , it should be possible to choose it so that the functional dependence of E^n would be realized by the product $F(E)E$.

Constancy of γ

For short period transients, the shutdown mechanism depends on temperature rises of the core metal, core water, and various combinations

of both; however, for the long period transients arguments can be advanced that γ should be independent of initial startup period. As shown in the SPERT analysis,⁽³⁾ even for periods > 0.2 sec, the shutdown coefficient is $\sim 90\%$ dependent on the expansion of core water. For the EBWR core this percentage is even higher since the removal of water moderator from the core increases absorption in the U^{238} resonance region (SPERT cores were fully enriched), which, besides the increased fast leakage, is the main reactivity decreasing mechanism. The temperature increases required for shutdown periods in the range tested are modest (around 10°C), and the expansion of water over such a temperature range is very close to linear. Thus, since a single mechanism predominates in the shutdown effect, and since that mechanism remains constant with energy, the conclusion would be that, for any given constant reactor conditions, γ should be constant.

Constancy of Reactor Conditions

Test series A and B should, of course, be characterized by significantly different γ -s. Both the addition of H_3BO_3 and the withdrawal of control rods serve to make the temperature coefficient less negative. That this is so is shown (Table 120-A-II) by the higher peak powers of series B tests and by the lower series B test temperature coefficients. Strictly speaking, even for a single series experiment, the reactor conditions are not entirely constant, since the different initial periods are caused by slightly different control rod bank positions, however, the variation in all cases was within 0.5 in. and can be neglected.

Table 120A-II

PARAMETERS USED IN ANALOG SIMULATION OF EBWR POWER TRANSIENTS

<u>Test</u>	<u>$\gamma_{(\text{Initial})}'$ $\phi/\text{Mw-sec}$</u>	<u>$\gamma_{(\text{Final})}'$ $\phi/\text{Mw-sec}$</u>	<u>$E_{(\text{Cutoff})}'$ Mw-sec</u>
1-A	0.3	0.021	163
2-A	0.42	0.021	130
3-A	0.65	0.021	100
1-B	0.14	0.011	230
2-B	0.28	0.011	180
3-B	0.37	0.011	163

Possible Forms of the Function $F(E)$

Some of the requirements of $F(E)$ are that it should:

1. Be equal to 1 at $E = 0$. This means that initially all of the energy enters system II.
2. Decrease until at some energy E_c (referred to as the cutoff energy) it would assume a constant value which would represent the fraction of the power remaining in the core at equilibrium natural-circulation conditions, that is, at conditions where the energy added serves to heat up system III uniformly. Since the heat capacity ratio of systems III to II is ~ 10 , the final constant value of $F(E)$ was taken to be 0.1.

Initially, the simplest possible function fitting such requirements was tried, specifically.

120A-SK-1

Where: k is a parameter to be fitted (units are reciprocal seconds), and it governs the energy at which natural circulation becomes fully established.

This model was investigated on the analog computer (Attachment B) and proved inadequate. The first objection to this model was that the transition from a decreasing to a constant heat capacity system was too abrupt, causing a perturbation in the kinetics equations which showed up as an additional peak in the power trace. Such a peak, of a much smaller nature, is shown in one of the final analog power traces (Fig. 120A-12), but as can be

seen from Figs. 120A-1-12 this peak is completely absent in any of the experimental curves.*

Fig. 120A-1. EBWR Startup Transient for 40.5-sec Startup Period (No H_3BO_3)

Fig. 120A-2. Analog Trace of Test 1-A Startup Transient

* In the long period (2-10 sec) SPERT data,⁽³⁾ such a secondary bump is quite prominent though it appears to be more flattened, especially compared to the secondary peaks produced by the $F(E) = 1 - KE$ model.

Fig. 120A-3. EBWR Startup Transient for 24.5-sec Startup Period (No H_3BO_3)

Fig. 120A-4. Analog Trace of k_{ex} , Power, and Energy Time Dependence for Test 2-A

Fig. 120A-5. EBWR Startup Transient for 15-sec Period (No H_3BO_3)

Fig. 120A-6. Analog Trace of k_{ex} , Power, and Energy Time Dependence for Test 3-A

Fig. 120A-7. EBWR Startup Transient for 40-sec Startup Period (3.66 gm/gal H_3BO_3)

Fig. 120A-8. Analog Trace of k_{ex} , Power, and Energy Time Dependence for Test 1-B

Fig. 120A-9. EBWR Startup Transient for 26.5-sec Startup Period (3.66 gm/gal H_3BO_3)

Fig. 120A-10. Analog Trace of k_{ex} , Power, and Energy Time Dependence for Test 2-B

Fig. 120A-11. EBWR Startup Transient for 15-sec Period (3.66 gm/gal H_3BO_3)

Fig. 120A-12. Analog Trace of k_{ex} , Power, and Energy Time Dependence for Test 3-B

The second attempt at $F(E)$ therefore made the transition at E_c a smoother one. The function tried was

120A-SK-2

This model proved satisfactory as far as the flux shape around the peak was concerned: a secondary flux rise occurred only for the 15-sec 3-B test simulation. In another important respect, however, the model proved faulty; namely, it was impossible to simulate all three experimental periods with the same energy coefficient, γ . In general, the initial γ (see Figs. 120A-13 and 15 where γ times the heat capacity is the slope of the δk_{ex} versus E curves) appeared to be higher for the shorter periods and became equal for all three periods only in the asymptotic region after $\sim 300 \sim 300$ Mw-sec. Further, the initial and final slopes of the δk_{ex} versus E curves turned out to be more in a ratio of 200:1 than the theoretically plausible 10:1. This aspect of the data is puzzling and could be due to three reasons:

1. System I functions more heavily in the initial shutdown than assumed.
2. System IV functions more heavily in the final shutdown than assumed.
3. The shutdown mechanism itself is of a different nature in the initial and equilibrium stages. (This could be due to such effects as local bubble formation, etc.)

Fig. 120A-13. Compensated Reactivity Versus Energy

Fig. 120A-14. Analog Trace of Compensated Reactivity Versus Energy

Fig. 120A-15. Compensated Reactivity Versus Energy

Both for theoretical and practical considerations it would be desirable to eliminate reason No. 3 and continue with a constant γ . Theoretically it does not seem probable (and was discounted for SPERT periods down to 0.1 sec) that local heating could occur to such an extent that bubbles would be formed. Other mechanisms which would operate differently for shorter than for longer periods, which would be different during a power rise than during a power decline, are difficult to postulate. Therefore, it is hoped that a form of $F(E)$ can be found which will successfully incorporate these apparent changes and which would allow γ to remain constant. The direction in which $F(E)$ should be modified to accomplish this will be pointed out subsequently. For the present study, however, the available time was insufficient and the model used therefore was:

$$\frac{1}{\phi} \frac{d\phi}{dt} = \frac{1}{T_0} + \gamma(E_c) \int_0^t (1 - KE)^2 \phi(r) dt \quad (3)$$

where: $\gamma(E_c)$ is a step function, changing values to a constant equilibrium γ at E_c .

The parameters which were used to force agreement with experimental data are given in Table 120A-II.

Fig. 120A-16. Analog Trace of Compensated Reactivity Versus Energy

Fig. 120A-17. Temperature Versus Excess Reactivity

Table 120A-III

COMPARISON OF EXPERIMENTAL AND ANALOG POWER TRANSIENTS

Test	Period, sec	Type	H ₃ BO ₃ , gm/gal	P _(tm) , [*] Mw	E _(tm) , [†] Mw-sec	k _{ex(tm)} , [‡] φ	Δk _{ex(tm)} , [§] φ	t _m , sec	Δk _{ex(to)} , [#] φ	(dk/dE) _F , ^{**} φ/Mw-sec	(dk/dE) _o , ^{††} φ/Mw-sec	(dk/dT) _F , ^{‡‡} φ/°C
3-A	15	Exp.	0	1.12	70	5.5	25.2	275	30.7	-0.0027	-0.51	-0.17
3-A	15	Analog	—	1.13	62	7.2	24.3	273	31.5	-0.02	-0.51	-0.13
2-A	24.5	Exp.	0	0.91	90	1.7	14.5	415	23.9	-0.0054*	-0.35	-0.34
2-A	25	Analog	—	0.91	77	4.1	19.9	387	24	-0.02	-0.37	-1.3
1-A	40.5	Exp.	0	0.775	102	0.9	17.7	630	18.6	-0.009*	-0.22	-0.57
1-A	40	Analog	—	0.768	98	2.5	15.5	610	18.0	-0.02	-0.19	-1.3
3-B	15	Exp.	3.66	1.84	155	3.4	29.7	315	33.1	-0.002	-0.48	-0.34
3-B	15	Analog	—	1.84	105	6.5	25	286	31.5	-0.011	-0.36	-0.7
2-B	26.5	Exp.	3.66	1.32	166	1.5	21.7	460	23.2	-0.002	-0.30	-0.34
2-B	25	Analog	—	1.32	120	3.5	20.5	425	24	-0.011	-0.26	-0.7
1-B	40	Exp.	3.66	1.16	180	1.1	17.7	645	18.8	—	-0.17	—
1-B	40	Analog	—	1.17	160	2	16	633	18	-0.011	-0.17	-0.7

* Maximum power.

† Energy generated up to maximum power.

‡ Remaining excess reactivity at maximum power.

§ Reactivity compensated at maximum power.

|| Time from the initiation of transient to maximum power.

Total initial k_{ex} (obtained from graphs).

** Final energy-reactivity coefficient obtained from the final slopes (> 300 Mw-sec) of Figs. 120A-13 through 16.

†† Initial energy-reactivity coefficient. An estimated average energy coefficient between 0 and 40 Mw-sec.

‡‡ The final energy coefficient expressed in terms of temperature, where the heat sink is considered to be system III (core + total vessel water).

Comparison of Experimental and Analog Data

The experimental data for the six tests are shown on the odd numbered Figs. 120A-1 through 17. The energy and δk_{ex} curves in the figures were calculated from the power distribution by a code described in Attachment C. The equivalent analog curves are shown on the even-numbered Figs. 120A-2 through 16. The analog simulation of the model is presented in Attachment B. Table 120A-III compares, side by side, the more important features of the experimental and analog power transients.

From safety considerations the most important parameters of a power transient are the peak power and the energy generated at the time of peak power. The analog model was adjusted to give the experimental peak power; therefore, the agreement of $P_{(\text{tm})}$ is forced for each analog transient separately.

Ultimate Form of Function $F(E)$

Some of the required properties of a satisfactory $F(E)$ function can be deduced from an inspection of the analog and experimental data. Specific suggestions are:

1. The initial rate of drop should be rapid with $F(E)$ reaching ~ 0.5 at an energy input of ~ 40 Mw-sec.
2. The rate of decrease should slow thereafter and $F(E)$ should approach 0.1 to 0.2 in about 100 Mw-sec.
3. The asymptotic value (~ 0.005 of initial value) should be reached very gradually after an energy input of ~ 300 Mw-sec.

The problem of finding this $F(E)$ function can be summarized as follows: it has to be that function of E which would modify the curves of Figs. 120A-13 through 16 (the total energy generated) so that δk_{ex} versus E (now the energy remaining in the core) would yield a straight line. Since by inspection of Figs. 120A-13 through 16 we see that in general shape the curves are similar it would appear that such a function exists.

CONCLUSIONS

Most of the specific deductions concerning the experimental and analog tests are given in the discussion of the shutdown model. The general conclusions which can be advanced at this stage are as yet based on insufficient data, however, at least two trends in the important parameters $P_{(\text{tm})}$ (peak power) and $E_{(\text{tm})}$ (energy generated up to peak power) are already quite apparent. Both parameters are illustrated in Fig. 120A-20 and correlated to $1/T_0$, the reciprocal of the initial period.

Fig. 120A-18. Sandborn Record of Test 3-B Startup Transient

Fig. 120A-19. Temperature Versus Energy

Fig. 120A-20. $P_{(tm)}$ and $E_{(tm)}$ Versus Reciprocal Startup Period

As seen in the figure, $P_{(tm)}$ is directly proportional to $1/T_0$ and the constant of proportionality is dependent on the temperature coefficient (difference in the slopes of the A and B series tests). This agrees with the SPERT data⁽³⁾ correlations, though it is probable that the slope of the lines would change as periods are approached where natural circulation is absent.

In Fig. 120A-20 $E_{(tm)}$ is seen to be inversely proportional to $1/T_0$, but the proportionality is not definite enough that a deduction about the temperature coefficient effect could be made. In general, this does not agree with the SPERT correlations which show a remarkably constant $E_{(tm)}$ for periods below prompt critical. (Above prompt critical $E_{(tm)}$ increases rapidly with increasing $1/T_0$.) This $E_{(tm)}$ dependency in the long period region is not surprising and just illustrates that during the slower approaches to peak power more energy is removed from the core by natural circulation. Further, the increase in energy removal rate is steeper than the decrease in reactivity to be compensated.

Attachment A

HEAT CAPACITIES AND TIME DELAYS OF THE EBWR SYSTEMS

For heat-transfer purposes the EBWR is envisioned as consisting of four concentric heat capacity systems, each characterized by a time constant. The schematic given below identifies the separate components of each system. Each of the higher order (longer time constant systems) include the shorter time constant systems below them.

Where:

- System I = Core matrix + clad.
- System II = System I + core water.
- System III = System II + vessel water.
- System IV = System III + vessel and shielding.

Estimates of the heat capacities and time constants are calculated for each system. The calculations are quite simple but sufficiently accurate for present purposes since, in general, the parameters differ by orders of magnitude, and the mode of analysis used would not be sensitive to second place discrepancies.

System I - Fuel Matrix + Clad

1. Heat Capacity

Volume of Fuel Matrix	$3.77 \times 10^5 \text{ cm}^3$	$\rho = 16.8 \text{ gm/cm}^3$
Volume of Zirconium Clad	$1.32 \times 10^5 \text{ cm}^3$	$\rho = 6.4 \text{ gm/cm}^3$
Weight of Fuel Matrix	$6.33 \times 10^6 \text{ gm}$	
Heat Capacity	$\sim 0.035 \text{ cal/(gm)} (^\circ\text{C}) [0-800^\circ\text{C}]$	
Weight of Zirconium Clad	$8.45 \times 10^5 \text{ gm}$	
Heat Capacity	$\sim 0.018 \text{ cal/(gm)} (^\circ\text{C}) [0-100^\circ\text{C}]$	
Total Heat Capacity of Core	$2.8 \times 10^5 \text{ cal/}^\circ\text{C} = \underline{1.17 \text{ Mw-sec/}^\circ\text{C}}$	

For accident evaluation purposes the energy inputs required to destroy the core are estimated:

Total energy required to melt core = $\sim 2100 \text{ Mw-sec}$;
Total energy required to vaporize core = $19,000 \text{ Mw-sec}$.

2. Time Constant - Instantaneous.

System II - Core + Included Water

Approximately 280 gal (1060 liters) of water are included directly within the core. Heat capacity of core water = $1.06 \times 10^6 \text{ cal/}^\circ\text{C} = 4.43 \text{ Mw-sec/}^\circ\text{C}$.

- Heat capacity = $1.33 \times 10^6 \text{ cal/}^\circ\text{C} = 5.56 \text{ Mw-sec/}^\circ\text{C}$
- Time constant - Reference 5 gives a water temperature time constant for the EBWR core of 0.25 sec. To relate this to the time delays of higher order systems a sudden 10°C temperature rise in the fuel is postulated. This would then raise the temperature of core water by 6.3°C within 0.25 sec.

System III - Core + Total Reactor Water

1. Heat capacity - During the present tests, the reactor contained ~ 4000 gal (15,000 liters) of water. This volume can vary considerably ($\sim$ a factor of 3) and in effect could be made a variable parameter in some future studies. Heat capacity of vessel water = 1.5×10^7 cal/ $^{\circ}\text{C}$ = 63 Mw-sec/ $^{\circ}\text{C}$.
2. Time constant - For this system, the primary concern is not a strictly defined time constant (the time in which the system III temperature rise will reach ~ 0.63 fraction of its equilibrium value) but rather in an estimate of the time in which natural circulation will become significant. Since the water volume of the vessel is so much larger than the water volume of the core, the later time will be much shorter than an exact (and difficult to calculate) time constant.

Specifically, one might ask how soon after a step temperature rise of the core water will $\sim 1/2$ of the energy input be removed from system II. Roughly, this will correspond to the time when $1/2$ of the water will be expelled from the core.

Approximating the temperature dependence of water density around $130\text{-}170^{\circ}\text{C}$ by a straight line the result is:

$$\rho_{\text{H}_2\text{O}} = 0.00096 T + 1.0613$$

where

$$T = ^{\circ}\text{C}.$$

Taking the driving force that a uniform density change in a 122-cm length channel provides and neglecting friction (the initial velocities are low) the result for the time required in this report is:

$$t_{(1/2)} = \sqrt{\frac{131}{\Delta T}}$$

where:

ΔT = Temperature difference between channel and surrounding water in $^{\circ}\text{C}$.

$t_{(1/2)}$ = Time in seconds from the initiation of ΔT to removal of $1/2$ of channel water.

For the 6.3°C temperature rise postulated for system II, $t_{(1/2)}$ would be ~ 5 sec. The exit water velocity in this time would be ~ 22 cm/sec (~ 0.7 ft/sec).

For some other ΔT we have:

$\Delta T = 1^\circ\text{C}$	$t_{(1/2)} = 11.5 \text{ sec}$	exit velocity = 10 cm/sec
$\Delta T = 20^\circ\text{C}$	$t_{(1/2)} = 2.6 \text{ sec}$	exit velocity = 47 cm/sec

It can be seen, from the above, that even for quite modest temperature rises natural circulation becomes an important phenomenon within 3-10 sec.

System IV - Total Reactor Vessel System

1. Heat Capacity - The total heat capacity of the reactor vessel, included water, and core, closely associated piping, shielding plug, etc., is given in Reference 6 as $\sim 26 \times 10^6 \text{ cal/}^\circ\text{C} = 108 \text{ Mw-sec/}^\circ\text{C}$.
2. Time Constant - The heat capacity system which is added at this stage (vessel + shield plug + associated hardware) is quite complex and a calculation of an exact time constant would be difficult. However, heat-transfer into these massive metallic blocks is by conduction only and therefore quite slow. For example, to raise the vessel wall temperature by 1°C when a temperature gradient across the wall of 5°C is maintained requires ~ 6 min. Time delays into the more massive top shield plug would be longer.

Attachment B

ANALOG SIMULATION

The analog simulation was performed on the Argonne Pace Computer. The equations to be simulated were:

$$\frac{1}{n} \frac{dn}{dt} = \frac{\delta k - \beta}{\ell} + \frac{1}{n} \sum_{i=1}^6 \lambda_i C_i \quad (\text{B-1})$$

$$\frac{1}{n} \frac{dC_i}{dT} = \frac{\beta_i}{\ell} - \lambda_i C_i \quad (\text{B-2})$$

$$\frac{dk}{\beta} = \frac{\delta k_o}{\beta} - \gamma(E) \int_0^t (1 - KE)^2 n(t) dt \quad (B-3)$$

where: Eqs. (B-1) and (B-2) are the standard six delay group kinetics equations and Eq. (B-3) is the final energy-reactivity coupling model used in this report.

The analog simulation of Eqs. (B-1) and (B-2) is standard and is given in Reference 7. Since at the present time the fissionable isotopes in the reactor include an appreciable quantity of Pu^{239} , the delayed neutron fraction is modified. The calculated λ 's and β 's for the present EBWR core (8800-Mwd reactor burnup, 20 fresh fully enriched spikes in core) are given in Table 102A-IV.

In the experiment a step reactivity input produced a power rise which continued for ~ 7 decades on an asymptotic period until the energy feedback became appreciable and modified the initial reactivity. Because of scale limitations such a power rise cannot be duplicated on the analog; however, the first five decades are of no interest, the only important aspect of the five-decade power rise is that at their completion the reactor is on a well-defined asymptotic period, and the delayed neutron groups are in a steady state ratio to each other. Thus, for analog purposes just the two decades in the Mw range are simulated, and the equilibrium delayed neutron concentrations are supplied as input data. The required potentiometer settings, for the three periods simulated, are given in Table 120A-V.

The circuit simulating the coupling equation Eq. (B-3) and the tie-in with the reactor kinetics simulator of Reference 7 is given in Fig. 102A-21.

Table 120A-IV

DELAYED NEUTRON FRACTIONS OF 20-SPIKE EBWR CORE

Delayed Neutron Group	λ	β
1	0.01252	0.0001986
2	0.03059	0.0013901
3	0.11429	0.0012782
4	0.30726	0.0026229
5	1.1542	0.000884
6	3.0578	0.0003155

$$\beta_{\text{total}} = 0.0066893$$

Table 120A-V

POTENTIOMETER SETTINGS FOR
DIFFERENT PERIOD TRANSIENT SIMULATION

No.	<u>T = 15</u>	<u>T = 25</u>	<u>T = 40</u>
1	0.00079	0.00147	0.001670
2	0.001575	0.002165	0.002750
3	0.00316	0.003705	0.004105
4	0.00411	0.004425	0.004625
5	0.004725	0.004835	0.004895
6	0.00049	0.000494	0.000496
7	0.00500	0.00500	0.00500
9	0.6320	0.4820	0.3600

Fig. 120A-21. Circuit Simulating the
Coupling Equation [Eq. (B-3)]

Attachment C

RE-171

Several methods are available for the calculation of instantaneous reactivity from flux-time data. All the methods consist of fitting an analytical expression (polynomial or an exponential) to a small number of the experimental points, obtaining the derivative numerically, and calculating the instantaneous period. Two such codes were tried and found unsatisfactory for the data in this experiment. The difficulty lay in the numerical differentiation routine. The data available were somewhat noisy and their differentiation amplified the noise to an extent that the calculated k_{ex} fluctuated unreasonably. Several methods were tried to dampen the noise, including the conversion of the kinetic equations to double integral equations; as it worked out the error in approximating integrals, though smaller, tended to be cumulative and the resulting answers progressively became worse as time increased. The most promising and subsequently used method was to first smooth the raw data by using a least-squares fit code, spanning a large numerical differentiation code. The two codes were then combined into a single one to form the RE-171 code. This code performed very satisfactorily and the k_{ex} curves displayed in Figs. 120A-1 through 13 were thus calculated.

The code requires for its input:

- prompt neutron lifetime
- fractions and decay constants of delay groups
- initial reactivity and period
- flux in given time steps.

The output includes:

- smoothed flux data
- the differential and the integral of the smoothed data
- excess reactivity and its first and second derivatives.

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TEMPERATURE AND H_3BO_3 EFFECT ON EBWR SUBCRITICAL COUNTING RATE

(Test Report No. 108C, dated June 19, 1961,
by K. Almenas and T. Nomura)

INTRODUCTION

During shutdown periods, the reactivity of the reactor is monitored by taking hourly 10-minute counts with the two BF_3 chambers. The count rate shows long and short time range fluctuations. In this report count rate changes due to temperature and H_3BO_3 concentration changes are illustrated and analyzed. Correction factors for the changes of core-to-counter shielding are presented and the possibility of using the subcritical counting rates for temperature coefficient measurement is investigated. For the H_3BO_3 free case the all-rod-in cold temperature coefficient is found to be $\sim 1.5 \times 10^{-4} \rho/^\circ C$. The uncertainty of the measurement is not analyzed here but is probably quite large.

THEORY

The subcritical counting rate $C_{(T)}$ of a counter is related to reactor parameters as follows:

$$C_{(T)} = G_{(T)} R_{(T)} E_{(T)} \left[\frac{S \ell^*}{1 - k_{eff}} \right] \quad (1)$$

where:

S = source strength (neutrons/sec)

ℓ^* = effective neutron lifetime (sec)

$E_{(T)}$ = counter efficiency

$R_{(T)}$ = counter shielding

$G_{(T)}$ = a factor dependent on core-counter geometry.

Since, in all cases, relative rather than absolute counting values are of interest, any of the three factors E, R, or G which remain constant over a range of variable changes (for example, temperature T) will be omitted.

$E_{(T)}$ Counter Efficiency

Counter efficiency for a counter activated by a $1/v$ absorber (B^{10} , in this case) is independent of temperature if the total number of thermal neutrons (not flux) is maintained constant. Epithermal activation can, in this case, be safely ignored, contributing less than 3% to the total counting rate.

Thus,

$$E_{(T)} \propto \left[\tilde{\phi}_2 \tilde{\Sigma} a_2 \right]^{-1}, \quad (2)$$

where:

$$\tilde{\phi}_2 = \text{thermal flux at counter} = nv_{(T)} = nv_{(0.0253)} \sqrt{\frac{T}{293}}$$

$$\tilde{\Sigma} a_2 = \text{absorption cross section of counter} =$$

$$N^B \sigma_a^B (0.0253) \frac{2}{\sqrt{\pi}} \sqrt{\frac{293}{T}}$$

and

$$v_{(0.0253)} = \text{neutron velocity at 0.0253 ev (220 w/sec)}$$

$$\sigma_a^B (0.0253) = \text{microscopic boron cross section at 0.0253 ev}$$

$$T = \text{temperature in question (}^{\circ}\text{K)}$$

$$N^B = \text{density of boron atoms in counter.}$$

From the above

$$\tilde{\phi}_2 \tilde{\Sigma} a_2 = N^B \sigma a^B (0.0253) \frac{2}{\sqrt{\pi}} n v (0.0253) = \tilde{\Sigma} a (0.0253) \tilde{\phi} (0.0253) \quad (3)$$

and $E_{(T)}$ is seen to be independent of T , if n (the number of thermal neutrons) is constant with T .

$G_{(T)}$ Geometric Factor

$G_{(T)}$ and $R_{(T)}$ are related factors. The specific definition of their product being:

$$G_{(T)} R_{(T)} = \frac{\int_{\text{counter}} \phi_{2(\underline{r})} dv}{\int_{\text{core + reflector}} \phi_{2(\underline{r})} dv} = \frac{\tilde{\phi}_2}{\int_{\text{core + reflector}} \phi_{2(\underline{r})} dv} \quad (4)$$

They [$G_{(T)}$ and $R_{(T)}$] are separated because the effect of temperature (or poisons) on the neutron density can be determined in the reflector but is very difficult to account for in a subcritical core. The ratio of the neutron density at the core-reflector interface to the total neutron density [the term $G_{(T)}$] is thus assumed to be temperature independent. This is done primarily for computational ease, since in the subcritical state the EBWR flux is grossly unsymmetric (the source is at the edge opposite to the counters), and only 2-dimensional calculations could approximate the changes in core neutron flux.

The above assumption is probably the largest single weakness in the theory presented.

$R_{(T)}$ - Counter Shielding

The question posed by the $R_{(T)}$ factor can be formulated as follows: If the neutron population is kept constant at the core-reflector interface while temperature changes are occurring, what will be the changes in counter

response? Or conversely - given a counter reading and reflector temperature - what is the neutron density at the core-reflector interface?

In the present core, the counters are located just behind the pressure vessel and are shielded from the core by ~ 33 cm of reflector water. Since the changes in density and neutron absorption rates in the shielding steel are almost constant with temperature, they are disregarded; also since the radius of curvature of the core is large, a slab geometry is assumed.

The geometric picture is as follows

Fig. 108C-SK-1

The solution to the two group diffusion equations in a non-multiplying medium is:

$$\phi_1(x) = C e^{-k_1 x} \quad (5)$$

$$\phi_2(x) = G e^{-k_2 x} + \frac{\Sigma_{s1}}{\Sigma_{a2} - D_2 K_1^2} C e^{-k_1 x} .$$

In order to achieve the objection of a constant thermal neutron density at $x = 0$, the following boundary conditions are used:

$$\phi_{2(0)}^R = \phi_{2(0)}^C = n_{(0)}^C v_{(0.0253)} \sqrt{\frac{T}{293}} = \sqrt{\frac{T}{293}} . \quad (6)$$

Where: superscripts R and C refer to reflector and core, respectively, and $n_{\text{Ov}}^{\text{C}}(0.0253)$ is normalized to 1.

$$\phi_{1(0)}^{\text{R}} = \phi_{1(0)}^{\text{C}} = \left[\frac{\phi_{1(0)}^{\text{C}}}{\phi_{(0.0253)}^{\text{C}}} \right] \quad (7)$$

The indicated ratio can be calculated by 2-group reflected reactor theory and found equal to $\phi_{1(0)}^{\text{C}}$ since, as noted above, $\phi_{(0.0253)}^{\text{C}}$ was normalized to 1. For the present calculations the above ratio was available for an ET-element EBWR core and is given with some other pertinent data in Table 108C-I.

With the indicated normalization, the constants of Eq. (5) become

$$C = \sqrt{\frac{T}{293}} \quad G = \sqrt{\frac{T}{293}} - \frac{\Sigma_{s1}}{\Sigma a_2 - D_2 K_1^2} \sqrt{\frac{T}{293}} \quad (8)$$

Now the variation of neutron density at the counter, due to variation in reflector properties, can be calculated and a correction factor, which eliminates $R_{(T)}$ from Eq. (1), obtained.

Defining a hopefully, temperature independent term

$$\left[S\ell^* \right] = G_{(T)} E_{(T)} S\ell^* \quad , \quad (9)$$

and correcting $C_{(T)}$ for the variation in $R_{(T)}$, Eq. (1) can be rewritten as follows:

$$C_{(T)\text{corr}} = \frac{\left[S\ell^* \right]}{1 - k_{\text{eff}}} \quad (10)$$

Where $\left[S\ell^* \right]$ is a source term in units of counter response and would be that count if no multiplication existed ($k_{\text{eff}} = 0$).

Then for any two different subcriticalities:

$$k_{\text{eff}_2} - k_{\text{eff}_1} = \left[S^{\ell*} \right] \left[\frac{1}{C_1} - \frac{1}{C_2} \right] , \quad (11)$$

where: C_1 and C_2 are the two corrected counting rates at conditions of subcriticality, k_{eff_1} and k_{eff_2} .

CALCULATIONS

$R_{(T)}$

The temperature dependence of the basic constants used was as follows:

$$\tau = \frac{31.27}{q} , \quad \Sigma_{\text{TR}_1} = 0.4667 q , \quad \Sigma_{\text{TR}_2} = 2.278 q \sqrt{\frac{293}{T}} , \quad (12)$$

$$\Sigma_{a_1} = 0.000566 q , \quad \Sigma_{a_2} = 0.01908 q \sqrt{\frac{293}{T}} ,$$

where:

$$q = \frac{q_{\text{H}_2\text{O}}(T)}{q_{\text{H}_2\text{O}}(20^\circ)} .$$

The derived constants then become:

$$\Sigma_{s1} = \frac{D_1}{\tau_1} = 0.0228 q , \quad K_1 = \sqrt{\frac{\Sigma_{s1} + \Sigma_{a_1}}{D_1}} = 0.1814 q , \quad (13)$$

$$K_2 = \sqrt{\frac{\Sigma_{a_2}}{D_2}} = 0.361 q \sqrt{\frac{293}{T}}$$

Some intermediate products and the final result of the H_3BO_3 free calculation are given in Table 108C-I.

Table 108C-I

 H_3BO_3 FREE CALCULATION

Temp., °K	$\left(\frac{\phi_{1(0)}^C}{\phi_{2(0)}^C}\right)$	K_1	K_2	e^{-33K_1}	$\phi^2(T)\sqrt{\frac{293}{T}}$	Correction Factor *
293	7.22	0.1814	0.361	0.00257	0.02971	1
323	7.141	0.1794	0.340	0.00269	0.03344	0.8885
373	7.119	0.1742	0.3072	0.00318	0.04454	0.6670
423	7.237	0.1676	0.2775	0.00397	0.06342	0.4685
473	7.551	0.1577	0.2469	0.00549	0.1012	0.2936
523	8.078	0.1453	0.2163	0.00827	0.1770	0.1679
573	9.044	0.1275	0.1814	0.0149	0.3736	0.07952
623	11.638	0.1038	0.1417	0.0826	1.0288	0.02888

* Normalized to 20°C.

Applying the correction factor listed in Table 108C-I to any count at a given temperature T gives the count that would have been obtained if the reflector temperature would have been 20°C. This correction factor is plotted in Fig. 108C-1. for the H_3BO_3 free case, and in Fig. 108C-2, for a concentration of 3.6 gm/gal of H_3BO_3 in reactor water.

[Sl *]

Fig. 108C-1. Counter Correction Factor (Normalized at 20°C. No H_3BO_3)

Fig. 108C-2. Counter Correction Factor (Normalized to the Zero H_3BO_3 , 20°C Condition)

On May 9, 1961, before and after adding H_3BO_3 , the data in Table 108C-II was obtained.

Table 108C-II

COUNTS VERSUS H_3BO_3 CONCENTRATION

	Average Count (10 five-min counts)	Temp., °F	Correction Factor	Corrected Count
No H_3BO_3	1550	80	0.747	1158
3.6 gm/gal H_3BO_3	260	65	1.34	349

As obtained from Reference 1, the worth of $H_3BO_3 = 0.0159 \rho / (\text{gm/gal})$ or a total $\Delta\rho$ between the two counts = 0.0572.

From the definition of ρ one obtains

$$k_{\text{eff}2} - k_{\text{eff}1} = \frac{-\Delta\rho k_{\text{eff}1}^2}{1 + \Delta\rho k_{\text{eff}1}}, \quad (14)$$

so that a $k_{\text{eff}1}$ value is needed to obtain Δk_{eff} . Generally, this is a difficult parameter to measure; however, the calculated value is ~ 0.95 for the H_3BO_3 free case. An error in the value of $k_{\text{eff}1}$ of $\pm 0.01 \rho$ results in an error of only $\sim 2\%$ of the value of Δk_{eff} .

Now using Eq. (11), the result is

$$[S\ell *] = \frac{\Delta k_{\text{eff}}}{\frac{1}{C_1} - \frac{1}{C_2}} = \frac{-0.049}{-0.002002} = 24.5 \text{ counts/min.}$$

Temperature Coefficient

A temperature coefficient can now be obtained from the subcritical counting rates as follows:

$$\text{Temp. Coef.} = \frac{\sim \Delta k_{\text{eff}}}{\Delta T} = \frac{[S\ell *]}{\Delta T} \left[\frac{1}{C_1} - \frac{1}{C_2} \right]. \quad (15)$$

From Figs. 108C-3 and 4 approximate average temperature coefficients were obtained as follows:

$$\text{for } \text{H}_3\text{BO}_3\text{-free case} = 1.4 \times 10^{-4} \rho/^{\circ}\text{C}$$

$$\text{for } 3.6 \text{ gm } \text{H}_3\text{BO}_3 \text{ per gal} = -3.0 \times 10^{-4} \rho/^{\circ}\text{C}$$

Fig. 108C-3. Counting Rates Versus Temperature (No H_3BO_3)

Fig. 108C-4. Counting Rates Versus Temperature (~ 3.6 gm/gal H_3BO_3)

DISCUSSION

As shown by the temperature coefficient calculation, the method presented is far from accurate (with H_3BO_3 in the reactor water, the temperature coefficient should become more positive). The reasons for the inaccuracy are not hard to find; two major reasons are: (1) statistical variation at the low counting rates prevailing with H_3BO_3 (Fig. 108C-4); and (2) limitations of theory, especially the disregarding of core flux variation during subcritical reactivity changes. The purpose of this report, however, is not to present accurate results and methods but rather to interpret qualitatively that data obtained routinely, as part of the operating procedure. During long shutdown periods, the only information of any sensitivity, regarding reactivity of the core, is obtained from the hourly 10-min subcritical counts, any changes in core reactivity will first be recorded during these counts. It is, therefore, important to understand standard expected changes in core reactivity and the counting rate so that a deviation from the standard can be recognized.

Again taking Eq. (1)

$$G_{(T)} = G_{(T)} E_{(T)} R_{(T)} \left[\frac{S\ell^*}{1 - k_{\text{eff}}} \right]$$

The parameters which can change the counting rate are:

1. $S\ell^*$ - This will introduce changes for the following reasons:
 - a. Source decay. During prolonged shutdown the source loses strength, having a half-life of ~ 61 days.
 - b. Source reactivation. During power runs the $S\ell^*$ is reactivated by the neutron flux and as a result the strength of the source before and after power runs will not be identical.
 - c. Fission product source. Immediately after shutdown an appreciable fraction of neutrons will be liberated by the γ -n reaction of the deuterium in water, with the hard γ provided by short lived fission products. This source decreases with a half-life of a few hours.
2. k_{eff} - The count will decrease when k_{eff} becomes smaller. This can occur for the following reasons:
 - a. Rise in temperature. With all the rods in, the temperature coefficient will be strongly negative, even for high H_3BO_3 concentrations.

- b. Rise in H_3BO_3 concentration.
 - c. Rise in Xe concentration. After a power run the count will decrease gradually until ~ 10 hours after shutdown when it will reach a minimum and start to increase.
3. $R_{(T)}$ - The count will increase as the shielding provided by the reflector water becomes less, this can occur for the following reasons:
- a. Increase in temperature, decreasing the density of reflector water.
 - b. Decrease in H_3BO_3 concentration.

Usually several of the above mentioned effects will be operating at once, sometimes in opposite directions, a good illustration of this is provided by Figs. 108C-3 and 4.

In the H_3BO_3 -free case (Fig. 108C-3), the count clearly decreases with temperature; this means that the negative effect of increased subcriticality overrides any decreasing counter shielding. After addition of H_3BO_3 , the two effects reverse; though the subcriticality is still increasing as the temperature rises, the shielding of the borated reflector water decreases correspondingly and the count rate is seen to rise (Fig. 108C-4).

It would be possible to increase the accuracy of the measurements by taking counts at a higher rod position, with the reactor closer to critical. Under these circumstances, the net count could be much higher and the statistical error, correspondingly less. In effect, this would enable temperature coefficients to be measured with a less reactive reactor (present loading), which measurements could be then applied to a more reactive core (100-Mw full loading). That is, control rod positions, which have a major influence on temperature coefficient, could be duplicated for the two cases, with the difference that the rod configuration would be subcritical with the present loading and super or just critical with the 100-Mw full loading. In practice, opportunities to obtain such data, not taking up any additional reactor time, do exist. Namely, heating-up operations, because of temperature stress limitations, take several hours, further, in-core temperature monitors are now available. The only change necessary in the heating-up procedure would be to start heating up with the control rods not all the way in, but rather at some intermediary level (from all in to critical position), and then to take subcritical counts and temperature readings simultaneously. The effective source term $[Sl^*]$ then should be obtained at that control rod position. It could be determined by obtaining a bank critical and following the flux decrease as the control rods are lowered to the previously determined intermediary level.

REFERENCES

- (1) EBWR Test Report No. 100D.

EXPERIMENTAL EBWR TEMPERATURE COEFFICIENTS

(Test Report No. 108D, dated June 30, 1961, by K. Almenas and J. Haugsnes)

SUMMARY

The temperature coefficient of the 20-spike EBWR core was found to be $-(6.0 \pm 0.6) \times 10^{-5} \rho/^{\circ}\text{C}$ at 65°C and $-(5.0 \pm 0.5) \times 10^{-5} \rho/^{\circ}\text{C}$ at 40°C . The control rod bank was at 16 in. during the measurements and there was no poisoning due to xenon or H_3BO_3 .

A number of experiments performed with an H_3BO_3 concentration of 3.6 gm/gal of reactor water were inconclusive, as the obtained value ($-1.0 \times 10^{-5} \rho/^{\circ}\text{C}$) was within the experimental error margin. Most of the uncertainties arose from effects which were underestimated and could have been avoided. A procedure is outlined in the ERROR ANALYSIS section of this report, with which temperature coefficients in the range 1×10^{-5} could be measured with an accuracy of $\pm 15\%$.

EXPERIMENTAL PROCEDURE

During the present experiment several procedures were initiated which proved satisfactory and should contribute to accurate temperature coefficient measurements in the future.

Mixing of Reactor Water

As shown in Reference 1 the water in the EBWR core tends to stratify, and the circulation provided by the cleanup loops is unable to break up this stratification. Thus for any experiments which require uniform temperatures or uniform H_3BO_3 concentration, some additional agitation of the reactor was required. Fortunately this could be provided by introducing compressed air through three available nozzles at the bottom of the reactor vessel. The mixing seemed to be quite thorough and complete within ~ 10 min. Further, no agitation of settled crud was evident; the water remained perfectly clean at all times and rod drop speeds were unchanged, indicating that no appreciable crud settled into the control rod thimbles. The general conclusion is that mixing by compressed air proved satisfactory and is recommended for future use. One unwelcome aspect (more rapid evaporation of reactor water) could not be avoided but it could be monitored and corrected for in the measurements. (See section on ERROR ANALYSIS.)

In-Core Temperature Measurements

Figures 108D-1 through 3 show the temperature history of the reactor during the experiments.

Fig. 108D-1. EBWR Core Temperatures on May 8, 1961

Fig. 108D-2. EBWR Core Temperatures on May 9, 1961

Fig. 108D-3. EBWR Core Temperatures on May 10, 1961

Thermocouples, Nos. 2, 3, and 4, were chromel-alumel with zirconium oxide insulation. They were swaged in an Inconel sheath having an OD of 1/16 in. The couples were held in the core by inserting them in 1/4-in. OD (1/16-in. wall) steel support tubes. The steel support tubes extended between the fuel plates and were positioned in the upper fuel handling fittings by means of spiders. Couple No. 7 was also chromel-alumel. The thermocouple locations are listed in Table 108D-I.

Table 108D-I

THERMOCOUPLE LOCATIONS

Thermocouple No.	Location in Core	Distance (in.) below Top Edge of Fuel Plates	
		Support Tube	Thermocouple
2	101-54	29	34-5/8
3	101-266	1	3
4	101-160	45	51-1/8
7	In downcomer	--	--

The voltages developed by the thermocouples were measured with a Leeds and Northrup, Type K-2, potentiometer and a D'Arsonval galvanometer. Voltages were determined to an estimated accuracy of ± 0.003 mv ($\pm 0.1^\circ\text{C}$). These voltages were then corrected, for the ice bath temperature and for the thermocouple calibration, to get temperature readings which are estimated to be accurate to $\pm 0.2^\circ\text{C}$.

The thermocouples, Nos. 2, 3, and 4, were calibrated in a thermostatically controlled circulating water bath against a Hg-in-glass thermometer. The thermometer used had a smallest scale division of 0.1°C and had been calibrated by the National Bureau of Standards. The NBS certificate showed the largest thermometer correction to be 0.11°C . The thermocouples were immersed in the bath to a depth of 44 in. and the thermometer was totally immersed. At any given temperature, the three thermocouple voltages differed by less than 0.02 mv (0.5°).

Thermocouple No. 7 was part of another experiment and could not be removed for calibration. Its calibration was arbitrarily taken to be that of thermocouple No. 3.

Measurement of Reactivity Changes at Constant Rod Position

Previously, reactivity was measured by correlating it to changes in the bank critical position. Besides incorporating all the uncertainties inherent in the rod calibration, this method had the additional disadvantage that the rod movement actually changed the core configuration so that the reactivity change in two slightly different cores was measured.

The procedure adopted during this experiment was as follows:

1. The bank critical was established.
2. No. 9 rod was dropped.
3. The desired temperature change was performed.
4. No. 9 rod was raised to its original position.

The reactivity change due to Step 3 was then measured, if positive, by the positive period resulting from Step 4. If the reactivity change was negative, the bank position (Step 1) would be raised to put the reactor on a positive period, and then the change in periods measured. It was estimated that, with due care, reactivity changes as small as $\pm 0.2 \times 10^{-5} \rho$ could be measured by this method.

CALCULATIONS

Source Correction

Period measurements, in a reactor with an immovable source, must be done under two opposing limitations:

1. The experiment should be performed at the highest acceptable power, in order to minimize the source influence.
2. The experiment must be performed at a sufficiently low power, so that nuclear heating does not introduce reactivity changes.

In the present series of experiments, limitation No. 1 was underestimated necessitating a source correction.

The influence of a source on an asymptotic period is evaluated as follows: Allowing sufficient time elapse for the decay of all transients after a step reactivity change, the neutron population increases as follows:

$$\frac{dn(t)}{dt} = \frac{n(t)}{T} + S \quad (1)$$

where

T = asymptotic period

S = neutron source (in neutrons/sec)

If we now plot dn/dt versus $n(t)$, a straight line with slope $1/T$ and intercept S results. This is shown in Fig. 108D-4 for two quite slow power rises (2600 and 1600 sec periods). (In Fig. 108D-4 the abscissa is shown in units of the linear flux recorder on the 10^{-10} scale.) From Fig. 108D-4 it was determined that $S = 0.0048 \times 10^{-10}/\text{sec}$. This value was used throughout the experiment. This is a reasonable approximation since the time for the experiment was short (3 days), no gross rod movements and no large temperature and H_3BO_3 concentrations were made, and thus the source contribution to the instrument reading probably stayed fairly constant.

Fig. 108D-4. dn/dt Versus $n(t)$

Asymptotic Period Calculation

The solution of Eq. (1) is

$$n_{(t)} = (n_0 + ST)e^{t/T} - ST \quad (2)$$

This solution involves the initial condition ($n = n_0$ at $t = 0$) which actually is a fictitious point, since at $t = 0$ the transients predominated and Eq. (1) did not apply. Therefore, for computational purposes n_0 was eliminated and time was used only in a relative sense. For two different readings at times t_1 and t_2 the result was:

$$e^{\Delta t/T} = \frac{n_{(t_2)} + ST}{n_{(t_1)} + ST} \quad (3)$$

where

$$\Delta t = t_2 - t_1$$

This equation was solved iteratively and the asymptotic periods obtained are given in Column 8 of Table 108D-II.

Reactivity

Reactivity associated with the periods obtained was read from a graphed solution of a six-delay group inhour equation. The total delayed neutron fraction (changed due to the presence of Pu^{239}) was 6.55×10^{-5} .

H_3BO_3 Concentration

The boric acid concentration was sampled and analyzed three times during the experiment; however, the evaporation rate (accentuated by air bubbling) was much higher and more variable than anticipated so that no complete picture of H_3BO_3 changes could be obtained from the three samples. The additional required points were calculated using the continuously recorded data of the reactor water level indicator. The curve of the boric acid concentration (normalized to fit the analytically obtained points) is given in Fig. 108D-5.

Fig. 108D-5. Boric Acid Concentration during Temperature Coefficient Tests

Table 108D-II

SUMMARY OF CRITICALS DURING TEMPERATURE COEFFICIENT MEASUREMENTS
(May 10, 1961)

Test	Time	Description	1	2	3	4	5	6	7	8	9	10	11	12
			Temp., °C	H ₃ BO ₃ , gm/gal	Period No.	P ₁ × 10 ⁻¹⁰	P ₂ × 10 ⁻¹⁰	Δ T, sec	Period, sec	Asymptotic Period, sec	ρ × 10 ⁵	Δ ρ × 10 ⁵	Δ t, °C	ρ × 10 ⁻⁵ /°C
5	10:25	Rods banked at 34.1 in. Reactor on positive period. Maximum power 13.6 × 10 ⁻¹⁰ . No. 9 rod inserted after test.	68.2	3.5760	B C D	5 8.46 4.08	13.56 13.56 8.12	219 115 135	253 274 235	254 ± 20	27.8 ± 2	+ 6.5	-8.6	-0.75
	12:00	No. 9 rod withdrawn to bank position. Reactor on positive period. Maximum power 24 × 10 ⁻¹⁰	59.6	3.5825	A B C	3 9 5	8.12 24.36 13.56	125 140 135	144 147 147	146 ± 1	44.7 ± 0.25			
		Rods inserted as a bank up to 33.7 in. to stop positive period. Final power. 8.7 × 10 ⁻¹⁰ Long negative period. No. 9 rod inserted.			A	8.82	8.64	180		-1500	-5.04			
6	13:25	No. 9 rod withdrawn to bank position 33.7 in. Power evened out at 1.18 × 10 ⁻¹⁰ .	68.3	3.5895							-30.9 ± 1.2	-14.66	+8.7	-1.6
	13:35	No. 9 rod withdrawn 2 in. to 35.7 in. Reactor seems to be exactly critical over a range of 10 min (~ 2 ft chart travel no deviation from linear power increase can be observed).			A B C D	2.62 4.08 2.62 3.34	5.68 5.68 4.08 4.83	480 240 240 240	2600 2585 2600 2750	2.660 ± 100	2.86 ± 0.11			
	13:38	No. 9 rod was pulled another inch to 36.7 in. Reactor now on period, exponential character of curve observable.			A B	6.39 7.78	10.59 10.59	165 105	404 415	409 ± 6	17.8 ± 0.25			
	13:43	No. 9 rod moved to 35.5 in. Power is linear at 10.2 × 10 ⁻¹⁰ (checked over 8 min).									- 3.6 ± 0.14			
7	16:15	No. 9 rod brought up to bank position (33.7 in.) Power evened out at 1.29 × 10 ⁻¹⁰ .	54.3	3.5980							-28.3 ± 1.1	+15.48	-14	-1.1
	16:20	No. 9 rod brought out to 35.7 in. Power rising slightly above linear rate.			A B C D	2.35 3.96 2.35 3.1	5.84 5.84 3.96 4.87	480 240 240 240	1625 1605 1645 1537	1603 ± 50	4.74 ± 0.13			
	16:29	No. 9 rod brought out to 36.7 in. Power rising more steeply.			A B	6.1 8.19	10.5 10.5	120 75	255 358	306 ± 50	23.6 ± 3.7			
	16:35	No. 9 rod lowered to 35.2 in. Power decreasing slightly at 10 × 10 ⁻¹⁰ level.			A	10.26	10	360			- 3.6 ± 0.14			

Temperature Coefficient

A temperature coefficient was calculated (columns 9 to 12 in Table 108D-III) for all conditions for which control rod settings were identical and reactor temperatures were different. A sizable correction for H_3BO_3 changes was required for most cases. The worth of H_3BO_3 used was 0.0159 $\rho/\text{gm/gal}$ (Reference 2).

ERROR ANALYSIS

Unfortunately, some error sources were underestimated during the experiment, and the results suffered accordingly. In the subsequent analysis all error sources (Table 108D-III) will be taken in turn, their contribution to the present experiment estimated, and ways to minimize the errors involved outlined.

Table 108D-III

THE ERROR SOURCES, THEIR CONTRIBUTIONS TO THE PRESENT EXPERIMENT AND EXPECTED MINIMIZED VALUES

(Given in percentage of the reactivity difference
 $15.48 \times 10^{-5} \rho$ in this case.)

Error Source	Present Contribution, %	Minimized Contribution, %
1. Uncertainty of H_3BO_3 concentration	± 50	± 6
2. Uncertainty of H_3BO_3 worth	± 24	± 3
3. Source uncertainty	± 3	± 0.2
4. Transient uncertainty	± 2	± 0.2
5. Repositioning of control rod	± 9	± 1
6. Reading errors (time, flux, and reactivity)	$+ 3$	± 1
7. Temperature measurement errors	± 3.6	± 3.6
TOTAL	± 90	± 15

As an illustrated example, for tests 6 and 7 (Table 108D-II), the conditions were as follows:

	at $t_1 = 13:35$	at $t_2 = 16:20$
Temperature, $^{\circ}\text{C}$	68.3	54.3
H_3BO_3 Concentration, gm/gal	3.5895	3.5980
Period Worth, ρ	2.86×10^{-5}	4.74×10^{-5}

Rod positions, in both cases, were identical.

$$\begin{aligned} \therefore \Delta\rho &= \text{reactivity in } t_2 \text{ period} - \text{reactivity in } t_1 \text{ period} + \text{worth of in-} \\ &\quad \text{creased } \text{H}_3\text{BO}_3 = (4.74 - 2.86) \times 10^{-5} + [0.0085 \times 0.0159] \\ &= 15.48 \times 10^{-5} \rho. \end{aligned}$$

$$\therefore \frac{\Delta\rho}{\Delta T} = 1.1 \times 10^{-5} \rho / ^{\circ}\text{C}.$$

Uncertainty of H_3BO_3 Concentration

The exact H_3BO_3 concentration would be of no importance if it could be kept exactly constant. However, at the elevated temperatures existing volumes of water were evaporated, and the H_3BO_3 became more concentrated.

The analysis was performed by titrating reactor water samples with 0.05 N NaOH; the total volume of titrant being 6-10 ml. Burette readings could not be improved much beyond ± 0.1 ml in this case or $\sim \pm 1.5\%$ of the total value. Further, only three samples were taken during the experiment, the balance of data for Fig. 108D-5 had to be filled out with normalized calculated points as obtained from the reactor water level indicator. The absolute accuracy of this instrument (as compared to other water level indicators) is about $\pm 2\%$ of total water volume (~ 3000 gal). Both of these errors exceed the difference in the two H_3BO_3 concentrations (~ 0.24); thus the maximum error contributed could, theoretically, be over 100% of the measured value. In practice, differential rather than absolute water level readings were used. Accuracy has not been checked against any other instrumentation but it seems to be $\sim \pm 0.5$ in. of water level or about ± 12 gal. This introduced an uncertainty in the difference of the H_3BO_3 concentrations of $\sim \pm 60\%$. In terms of measured reactivity difference, this uncertainty becomes $\pm 50\%$ (Table 108D-III).

Minimization - Fortunately, this source of error can be minimized effectively, the following steps are suggested:

1. More accurate H_3BO_3 analysis. Titration accuracy could be improved by a factor of 10 or more using more dilute NaOH solutions, going to larger samples, or even using gravimetric methods.
2. Differential water level could be measured accurately and simply, by lowering a graduated meter stick against some rigid and prominent core structure. A wetted stick could be read to within ± 0.3 cm or ~ 2.5 gal of H_2O .
3. Use a full reactor, that is ~ 600 gal of water total instead of the 3000 gal used, to reduce total relative evaporation (around 17 gal in this case) to half.
4. Decrease cooling time. Only one pump, in the cleanup system, was available for this experiment. With both pumps on the line, cooling, and evaporation time would be cut in half.

To sum up, Steps 3 and 4 would decrease the extent of the relative concentration change by a factor of four. Step 2 would give differential volume measurements with a total relative accuracy (6000 gal volume) of 0.05% and the total improved accuracy would be $\pm 6\%$ of the total reactivity difference ($15.5 \times 10^{-5} \rho$).

Uncertainty in H_3BO_3 Worth

As shown in Reference 3, the worth of H_3BO_3 ($0.0159 \rho/\text{gm/gal}$) is known to within $\pm 30\%$ (maximum possible error). This could have contributed a deviation of $\pm 24\%$ to the present experiment.

Minimization - Steps 3 and 4 would reduce this error by a factor of four; to decrease it further the H_3BO_3 worth would have to be established more accurately. Increasing the accuracy of analysis by a factor of 10 would decrease the maximum uncertainty in H_3BO_3 worth to $\pm 15\%$. Thus, error from this source should not exceed $\pm 3\%$.

Source Uncertainty

Though it is difficult to evaluate what absolute uncertainty existed in the source value used ($0.0048 \times 10^{-10}/\text{sec}$), a guess of 50% is certainly an outside estimate. Such an error would effect the reactivities as follows:

If source was 50% high

$$\Delta \rho \text{ (periods)} = 1.96 \times 10^{-5} \rho$$

If source was 50% low

$$\Delta \rho \text{ (periods)} = 1.10 \times 10^{-5} \rho$$

Total absolute error = $\sim 6\%$ or $\pm 3\%$.

NOTE

The error appears small here because the bulk of the reactivity difference is due to the change in H_3BO_3 concentration. Disregarding that part, total relative error due to the quoted source uncertainty would be $\pm 30\%$.

Minimization - As shown in Figs. 108D-1 through 3, no heating occurred during the criticality measurements. The periods consequently could have been taken safely at a higher power. Increasing the power by a factor of 10 reduces the error in approximate accord. As shown in Reference 3, the power range of 5×10^{-9} (~ 10 kev) is a satisfactory one, the error here would be $\sim \pm 0.2\%$.

Transient Uncertainty

In the present case, the periods were measured ~ 4 min after rod movement stopped. According to B. Toppel's⁽⁴⁾ analysis, a total contribution of $\sim 10\%$ existed, at such a time, due to transients (for periods of above 1000 sec). This contributed a relative error of $\sim 2\%$ to the present measurement, a factor of 10 higher if the H_3BO_3 worth were not included.

Minimization - Theoretically, this error could be minimized to zero. However, this would require waiting times, in the range of tens and even hundreds of minutes, and the patience of the experimenters; flux levels and other considerations would hardly permit it. In practice, it probably is better to estimate the reactivity steps so that periods below 100 sec result. Under these conditions, waiting times of 5 min will give accuracies of 5-1% or better (depending on initial conditions).

Repositioning of Control Rod

The rod position can be read easily to within ± 0.02 in. In the present case, an error of this magnitude would be worth $\sim \pm 0.6 \times 10^{-5} \rho$ or $\pm 4\%$.

Minimization - The initial and final position of the movable control rod (No. 9) does not have to be flush with the control rod bank (where its worth is appreciable, $\sim 40 \times 10^{-5} \rho/\text{in.}$); but the movable rod could be an appreciable distance above the control rod bank position, where its differ-

ential worth is much less. Thus, we find that at 5 in. above the bank position the differential worth has decreased to $\sim 10 \times 10^{-5} \rho / \text{in.}$, and the relative error would decrease accordingly.

Reading Errors

Under this category are the errors due to reading (or misreading) of the time, power, and reactivity values. In Reference 3, the total error due to these causes is estimated to be $\pm 3\%$ of the reactivity being measured.

Minimization - Since these are random errors, they can be reduced by taking a number of readings. Thus for a total of nine measurements the error decreases as the square root of n and would become $\pm 1\%$.

Temperature Measurements

As seen from Figs. 108D-2 and 3, the temperatures of the differently located thermocouples agreed, at most times, to within $\pm 0.5^\circ\text{C}$. Taking this to be a measure of the precision would give an uncertainty to the final result of $\pm 3.6\%$. It is doubtful whether this measurement could be improved appreciably, because as it was, great care was taken to shield and calibrate the couples and to mix the reactor water.

CONCLUSIONS

As shown by the discussion, the largest uncertainties in the present and minimized case are introduced by the variable H_3BO_3 concentration. If experiments without H BO are performed, accuracies of $\pm 6\%$ are quite possible. All of the errors quoted herein are reasonable maximum errors, so that experimental care and hopefully random cancelling of positive and negative errors should reduce the total further.

REFERENCES

- (1) EBWR Test Report No. 108A.
- (2) EBWR Test Report No. 109D.
- (3) EBWR Test Report No. 100E.
- (4) B. J. Toppel, "Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements," Nuc. Sci. and Eng. 5, (February, 1959), pp. 88-98.

VOID COEFFICIENT MEASUREMENTS IN EBWR

(Test Report No. 124, dated August 27, 1962, by H. P. Iskenderian)

GENERAL

Measurements were made to estimate void coefficients of the cold EBWR for varying concentrations of H_3BO_3 .

The test facility consisted of four void simulator subassemblies, which could be fitted in the ET elements of the core. These void simulators could produce 70% void when air replaced water in the simulators.

The reactivity effects of these void simulators were measured at a number of locations or regions in the core (Figs. 124-1 a-e). The results of these measurements were then averaged over the area of the core containing ET elements. By an extrapolation of data, that concentration of H_3BO_3 was estimated which would yield a zero void coefficient for a core consisting of all ET elements. This extrapolation is limited in its use in that it refers to two criticality positions of the rods; and it may not apply for other positions of the rods.

H_3BO_3 Concentration, 0.5 gm/gal (cold).

When void simulators filled with water, reactor critical with rods, Nos. 1-9, at 10.78 in. except rod No. 2 down.

When simulator(s) filled with air, following data obtained:*

Air in Simulators	(1)	(1) + (2)	(1) + (2) + (3)**	(1) + (2) + (3)** + (4)
Critical Position Rod No. 2	2.95	4.45	4.45	5.55
$\Delta\rho/\beta$	0.044	0.089	0.089**	0.134

*Rods, Nos. 1-9, in position 10.78 in., except for rod No. 2 as indicated above, for criticality.

**Void simulator inoperative.

Fig. 124-1a. Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements

H_3BO_3 Concentration, 0.5 gm/gal (cold).

When void simulators filled with water, reactor critical with rods, Nos. 1-9, at 10.81 in. except rod No. 2 down.

When simulator(s) filled with air, following data obtained:*

Air in simulators	(1)	(1) + (2)	(1) + (2) + (3)	(1) + (2) + (3) + (4)
Critical Position Rod No. 2	2.95	4.42	5.12	6.1
$\Delta\rho/\beta$	0.044	0.086	0.114	0.16

*Rods, Nos. 1-9, in position 10.81 in., except for rod No. 2 as indicated above, for criticality.

Fig. 124-lb. Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements

H_3BO_3 Concentration, 0.5 gm/gal (cold).

When void simulators filled with water, reactor critical with rods, Nos. 1-9, at 10.81 in. except rod No. 2 down.

When simulator(s) filled with air, following data obtained:*

Air in Simulators	(1)	(1) + (2)	(1) + (2) + (3)	(1) + (2) + (3) + (4)
Critical Position Rod No. 2	-	4.32	5.25	6.19
$\Delta\rho/\beta$	-	0.085	0.121	0.166

*Rods, Nos. 1-9, in position 10.81 in., except for rod No. 2 as indicated above, for criticality.

Fig. 124-lc. Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements

H_2O Concentration, 3.37 gm/gal (cold)

When void simulators filled with water, reactor critical with rods, Nos. 1-9, at 18.29 in. except rod No. 2 down.

When simulator(s) filled with air, following data obtained:*

Air in Simulators	(1)	(1) + (2)	(1) + (2) + (3)	(1) + (2) + (3) + (4)
Critical Position Rod No. 2	3.3	5.11	6.01	7.1
$\Delta\rho/\beta$	0.031	0.06	0.081	0.113

*Rods, Nos. 1-9, in position 18.29 in., except for rod No. 2 as indicated above, for criticality

Fig. 124-1d. Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements

H_2O Concentration, 3.37 gm/gal (cold).

When void simulators filled with water, reactor critical with rods, Nos. 1-9, at 18.29 in. except rod No. 2 down.

When simulator(s) filled with air, following data obtained:*

Air in Simulators	(1)	(1) + (2)	(1) + (2) + (3)	(1) + (2) + (3) + (4)
Critical Position Rod No. 2	-	5.12	6.31	7.5
$\Delta\rho/\beta$	-	0.0605	0.090	0.1265

*Rods, Nos. 1-9, in position 18.29 in., except for rod No. 2 as indicated above, for criticality.

Fig. 124-1e. Void Coefficient Measurements. Void Simulators Inserted in Four Fuel Elements

Fig. 124-2. Calibration of Rod No. 2. Differential Worth of Rod as Indicated

MEASUREMENTS

Calibration of Rod No. 2 by Period Method

A calibration of rod No. 2 was obtained by the period method, for two concentrations of H_3BO_3 (0.5 gm/gal and 3.37 gm/gal), when the No. 2 rod was done (see Fig. 124-2), and the remaining rods were at bank critical positions.

Accuracy of Calibration

The errors of calibration were greater for small reactivities, when long periods were involved, and when the reactor "source" effects became appreciable. It is estimated that these errors may have been as large as $\pm 0.4\%$ for small ρ/β readings of about 4.0% which occur when the void effect of one simulator was measured.

Void Effect Measurements

The reactivity effects of voids in simulators 1, 1 + 2, 1 + 2 + 3, and 1 + 2 + 3 + 4, located as shown on Figs. 124-1 a-e, were obtained.

In making these measurements, rods Nos. 1-9 were retained in position except for rod No. 2 and another rod. It was estimated that the latter two rods could be reset within a maximum total positional error of ~ 0.05 in., resulting in a corresponding error in ρ/β of 0.14% to 0.24% .

Fig. 124-3. Thermal Neutron Flux (Integrated for Each Element) Radial Distribution

elements are in relatively low thermal flux areas, but S elements have appreciable ϕ_{th} values (see Fig. 124-3).

The results of measurements, indicated in Figs. 124-1 a-e, are summarized below in Table 124-I.

Table 124-I

($\Delta k/\beta$) VALUES (in cents)

70% VOIDS IN ET ELEMENTS TESTED

H_3BO_3 Concentration	0.5 gm/gal		3.37 gm/gal	
	3 Simulators	4 Simulators	3 Simulators	4 Simulators
Region 1a (Fig. 124-1 b and d)	11.4	16.0	8.1	11.3
Region 1b (Fig. 124-1 c and e)	12.1	16.6	9.0	12.6
Region 2 (Fig. 124-1 a)	13.4	----	---	----
<p>Note: Δk void (Region 1b) > Δk void (Region 1a) Δk void (Region 2) / Δk void (Region 1a) = 1.17 from above data</p> <p>From Fig. 124-3 $\int \text{Region 2 } \phi_{th} dV / \int \text{Region 1a } \phi_{th} dV = 1.11$.</p>				

DISCUSSION OF RESULTS

The purpose of these tests was to check the calculated values of the void coefficient of core 1A, resulting from temperature-void effects in the ET elements, also, to determine the concentration of H_3BO_3 that would result in a zero value of this coefficient.

Four void simulators were used in the present system, instead of the previous arrangement of styrofoam fillers in the core, to shorten the time of these tests.

As already stated, the void effects could be measured in the ET elements only, due to physical limitations. $\Delta k/\Delta v$ values should be greater in the EH elements, and especially in S elements. The EH

These void reactivity measurements made in localized areas and in different regions indicate:

1. $\Delta k / \Delta v \propto$ approximately to ϕ_{th}
2. Some cross-streaming of neutrons between void channels.
(Compare $\Delta k / \beta$ values in Table 124-I, for Regions 1a and 1b.)
3. $\Delta k / \Delta v$ varies with the concentration of H_3BO_3 (and the control rod bank settings).

Measurements (Nos. 1 and 2 above) lack high accuracy due to the small Δk 's involved but are qualitative check measurements.

Measurements (No. 3 above) were obtained for corresponding settings of the bank of rods Nos. 1-9 (except rod No. 2) at 10.8 in. and 18.3 in. An extrapolation of the data to other concentrations of H_3BO_3 is limited in its use in that it refers to the two criticality positions of the bank settings of the control rods given above.

Rod No. 2 was chosen as the void reactivity measuring stick, in view of the minimized effects of voids on the calibration of this rod.

The void reactivity effect of the 56 ET elements was obtained from the measured values of $\Delta k / \beta$ given in Table 124-I.

For a 0.5 gm/gal concentration of H_3BO_3 :

$$-\frac{1}{\beta_{eff}} \frac{\Delta k}{\Delta v} (ET) = + \left[\frac{36 \text{ elements in Region 1}}{4 \times 0.70} \times \$0.166 + \frac{20 \text{ elements of Region 2}}{3 \times 0.70} \times \$0.134 \right]$$

$$\text{Taking } \beta_{eff} = 6.95 \times 10^{-3} \quad \frac{\Delta k}{\Delta v} (ET) = -0.0237$$

Similarly, for a 3.37 gm/gal concentration of H_3BO_3 :

$$-\frac{\Delta k}{\Delta v} (ET) = \left[\frac{36}{4 \times 0.70} \times 0.126 + \frac{20}{4 \times 0.70} \times \$0.113 \right] \times$$

$$6.95 \times 10^{-1} = 0.0168$$

The void coefficients obtained above are plotted in Fig. 124-4 and the results are compared with the calculated values for ET elements given in Reference 1 (see Table 124-II).

Fig. 124-4. EBWR Void Coefficient Measurements for Regions Containing ET Elements Only ($\Delta \rho / \Delta v$ Versus H_3BO_3 Concentration)

Table 124-II

MEASUREMENT OF REACTIVITY VOID COEFFICIENTS

	$\Delta k / \Delta v$ Values		
	0	1.33	2.66
H_3BO_3 Concentration (gm/gal)			
Measured for 56 ET elements	-0.025	-0.013	-0.0016
Reference 1 for total of Core	-0.119	-0.085	-0.045

An estimate of $\Delta k / \Delta v$ (cold zero H_3BO_3) = -0.077 was obtained for the core, by normalizing the calculated data in Reference 1 to agree with measured $\Delta k / \Delta v$ value for the ET elements.

It is seen that under the specified conditions the reactor will have a negative void coefficient for a H_3BO_3 concentration of a 10 gm/gal.

REFERENCES

- (1) R. Avery, et al., "EBWR Core 1A Physics Analysis," ANL-6305 (February 1961).

RELATIVE REACTIVITY MEASUREMENTS OF EBWR FUEL ELEMENTS

(Test Report No. 100D, dated December 21, 1961, by K. Almenas and V. Kolba)

SUMMARY

Relative reactivity measurements of EBWR fuel elements were made in two radial core directions. The results were compared to those derived from theoretical reactivity versus burnup calculations. The experimental and theoretical procedures were found to be attractively simple and a qualitative check between the two was obtained; however, for measurements requiring preciseness, the method was found to be unsuitable in the EBWR case.

NOMENCLATURE

f = fast fission factor

p = resonance escape probability

Σ_a = absorption cross section

Σ_f = fission cross section

Σ_{sl} = slowing down cross section

$\bar{\phi}_{(E)}$ = average flux in element

$\bar{\phi}_{(R)}$ = average flux in reactor

$I_1(\kappa L), I_0(\kappa L), K_0(\kappa L), K_1(\kappa L)$ - Bessel function of the first and second kind

L = radius of cylindrized fuel element

THEORY

The reactivity "worth" of an element in a core depends on the element itself and on the "importance" of its position within the core. To separate the two effects, relative rather than absolute reactivity measurements were made. That is, the worth of three elements were tested in a particular position - two of known composition and reactivity (standard unused elements) which bracketed the reactivity of the unknown (burned)

element. If all of these elements would perturb the flux in an identical manner, the reactivity of the unknown element could be obtained by simple interpolation. Unfortunately, this is not the case, both the resonance and thermal fluxes change appreciably as the elements are interchanged and a parameter has to be found which would take this into account. For this purpose the following parameter was defined:

$$\Sigma^* = \left[\epsilon \nu \Sigma_{f_3} - \Sigma_{a_3} \right] \frac{\bar{\phi}_3(E)}{\bar{\phi}_3(R)} - \Sigma_{a_2} \frac{\bar{\phi}_2(E)}{\bar{\phi}_3(R)} \quad (1)$$

where

$\bar{\phi}(E)$ = average flux in the element

$\bar{\phi}_3(R)$ = average reactor thermal flux

$$\Sigma_{a_2} = \frac{1-p}{p} \Sigma_{sl_2}$$

Subscripts 2 and 3 refer to epithermal and thermal groups respectively.

Subscripts E and R refer to element and reactor respectively.

Σ^* is thus an artificial fission cross section which in the unperturbed reactor flux would produce the same reactivity effect as the combination of fission and absorption cross sections of the actual element. The experimentally unproved (but theoretically consistent with second-order perturbation theory) assumption is that Σ^* is directly proportional to the reactivity effect of the element in the core. In the definition of Σ^* the following simplifications were made:

1. That thermal, resonance, and fast neutrons have the same importance.
2. That no epithermal fission and absorption (except resonance absorption) occurs.

For slightly enriched uranium systems, the second assumption has been found to be quite reasonable. The first assumption depends largely on the size of the core. That is, since fast neutrons leak preferentially they have a lower importance, i.e., for EBWR the total percent leaked is on the order of 3 to 5%. Conceivably this could be corrected for, however, the correction is position dependent and was not attempted.

Some additional simplifications were made for the calculation of the required flux ratios. These simplifications were:

1. That the square element can be represented by a cylinder with the same cross-sectional area.
2. That the rest of the reactor (in comparison with the test element) is infinite and composed of ET elements of 10,000 Mwd irradiation.
3. That the fast flux (> 180 kev) is flat.

Given the above assumptions the solutions for the fuel element to reactor flux ratios are relatively simple:

$$\frac{\bar{\phi}_{2(E)}}{\bar{\phi}_{3(R)}} = \frac{2 [q_{2(R)} - q_{2(E)}]}{L \kappa_{2(E)} q_{3(R)}} \left\{ \frac{I_1 [\kappa_{2(E)} L]}{I_0 [\kappa_{2(E)} L] + \frac{\kappa_{2(E)} K_0 [\kappa_{2(R)} L]}{\kappa_{2(R)} K_1 [\kappa_{2(R)} L]} I_1 [\kappa_{2(E)} L]} \right\} + \frac{q_{2(E)}}{q_{3(R)}} \quad (2)$$

where:

Subscripts 2 and 3 refer to neutron energy groups.

Subscripts E and R refer to element and reactor respectively.

$$q_2 = \frac{\Sigma_{s1_1}}{\Sigma_{s1_2} + \Sigma_{a_2}} \quad (3)$$

$$(\kappa_2)^2 = \frac{\Sigma_{s1_2} + \Sigma_{a_2}}{D_2}$$

L = radius of cylindrized element = 6.09 cm.

The thermal flux ratio is obtained in a identical manner, except the source terms q have an epithermal flux dependence.

$$\frac{\bar{\phi}_3(E)}{\bar{\phi}_3(R)} = \frac{2[q_{3(R)} - q_{3(E)}]}{L \kappa_{3(E)} q_{3(R)}} \left\{ \frac{I_1 \kappa_{3(E)} L}{I_0 [\kappa_{3(E)} L] + \frac{\kappa_{3(E)} K_0 [\kappa_{3(R)} L]}{\kappa_{3(R)} K_1 [\kappa_{3(R)} L]} I_1 [\kappa_{3(E)} L]} \right\} + \frac{q_{3(E)}}{q_{3(R)}} \quad (4)$$

where:

$$q_{3(E)} = \bar{\phi}_2(E) \frac{\Sigma_{s1_2}}{\Sigma_{a_3}}$$

$$q_{3(R)} = \bar{\phi}_2(R) \frac{\Sigma_{s1_2}}{\Sigma_{a_3}} = q_{2(R)} \frac{\Sigma_{s1_2}}{\Sigma_{a_3}} = \frac{\Sigma_{s1_1} \Sigma_{s1_2}}{(\Sigma_{s1_2} + \Sigma_{a_2}) \Sigma_{a_3}}$$

The flux ratios calculated by the above expressions compare favorably with equivalent PDQ calculations; however, the comparison with actual reactor conditions is more uncertain since such strong local flux disturbances as control rods and boron-stainless steel strips have not been taken into account.

Fig. 100D-1. Location of Mapped Elements In Northwest Quadrant of EBWR Core

Fig. 100D-2.

Integral Worth of No. 9 Rod with Remaining Eight Rods Banked at 21.1 in.

Fig. 100D-3.

Σ^*_{ET} Versus Reactor Mwd Burnup

Σ^* can now be calculated for known element compositions and its theoretical change with burnup can be obtained. This was done for the ET element (assuming 15% average voids at 20 Mw) and is shown in Fig. 100D-3.

PROCEDURE

Experimental

The experimental procedure consisted of three steps:

1. The burned element to be tested was moved to the edge of the core.
2. A standard element (thin natural and thick enriched in turn) was substituted in place of the element to be tested.
3. The reactivity difference was measured by obtaining a Critical and noting the change (from the condition previous to Step 1 above in the No. 9 rod position).

Fuel element moves were made as listed in Tables 100D-I and II to accomplish the core mapping. Figure 100D-1 indicates the fuel element locations which were involved in the mapping process. The stainless steel dummy elements were transferred from the reactor to the storage pit using the auxiliary winch on the overhead crane and the rigid fuel handling tool. The new "standard" fuel element was placed in the core and later moved to the storage pit using the above transfer arrangement. Fuel moves in the core were made using the "A" frame on the indexing plug and the rigid tool. Proper hazards monitoring was maintained on all fuel element moves. All control rods were banked at 6 in. during fuel element moves.

Table 100D-I

HANDLING MOVES TO AND FROM THE CORE AND STORAGE
(5/23 - 5/25/61)

Move	Date and Time	Fuel Element No.	From - Location	To - Location
1	5/23/61 11:10	Ran a Critical (C) S.D.	C.I. 0-122	S.P. I-1
2	11:15	S.D.	C.I. 0-143	S.P. I-2
3	16:17	Ran a Critical (D) ET-5	C.I. 129-370	C.I. 0-122
4	16:26	T-19	NEW	C.I. 129-370
		Ran a Critical (E)		
1	5/24/61 09:30	Ran a Critical (E _a) T-19	C.I. 129-370	C.I. 0-145
2	09:40	ET-5	0-122	129-370
3	09:47	ET-21	109-311	0-122
4	09:54	T-19	0-145	109-311
		Ran a Critical (F)		
5	10:58	T-19	C.I. 109-311	0-145
6	11:04	ET-21	0-122	109-311
7	11:09	ET-27	95-294	0-122
8	11:15	T-19	0-145	95-294
		Ran a Critical (G)		
9	12:39	T-19	C.I. 95-294	0-145

Table 100D-I (Cont'd.)

Move	Date and Time	Fuel Element No.	From - Location		To - Location	
10	12:44	ET-27	C.I.	0-122	C.I.	95-294
11	12:51	ET-40	↓	59-259	↓	0-122
12	12:56	T-19	↓	0-145	↓	59-259
		Ran a Critical (H)				
13	13:40	T-19	C.I.	59-259	↓	0-145
14	13:46	ET-40	↓	0-122	↓	59-259
15	13:52	EH-31	↓	34-237	↓	0-122
16	13:58	T-19	↓	0-145	↓	34-237
	5/24/61	Ran a Critical (I)				
17	14:36	T-19	C.I.	34-237	C.I.	0-145
18	14:43	EH-31	↓	0-122	↓	34-237
19	14:48	ET-48	↓	101-266	↓	0-122
20	14:53	T-19	↓	0-145	↓	101-266
		Ran a Critical (J)				
21	15:26	T-19	C.I.	101-266	↓	0-145
22	15:32	ET-48	↓	0-122	↓	101-266
23	15:37	ET-31	↓	80-237	↓	0-122
24	15:41	T-19	↓	0-145	↓	80-237
		Ran a Critical (K)				
	5/25/61	Ran a Critical (K _a)				
1	09:24	T-19	C.I.	80-237	C.I.	0-145
2	09:28	ET-31	↓	0-122	↓	80-237
3	09:32	EH-44	↓	47-203	↓	0-122
4	09:37	T-19	↓	0-145	↓	47-203
		Ran a Critical (L)				
5	10:33	T-19	C.I.	47-203	S.P.	A-2
6	10:39	S.D.	S.P.	1-2	C.I.	0-145
7	10:44	EH-44	C.I.	0-122	↓	47-203
8	10:53	S.D.	S.P.	1-1	↓	0-122
		Ran a Critical (D _a)				

Table 100D-II

HANDLING MOVES TO AND FROM THE CORE AND STORAGE PIT
(9/1, 9/5 - 9/7/61)

Move	Date and Time	Fuel Element No.	From - Location	To - Location
	9/1/61	Ran a Critical (1-A) Removed 500 gal H ₃ BO ₃ water and added 500 gal "clean" demineralized water to reactor.		

Table 100D-II (Cont'd.)

Move	Date and Time	Fuel Element No.	From - Location	To - Location
	9/5/61	Ran a Critical (1-B)		
		Ran a Critical (1-C)		
1	09:42	EH-IFA-II	Rotated 90 degrees counterclockwise	
2	09:55	S.D.	C.I. 0-122	S.P. 1-1
3	10:05	S.D.	C.I. 0-143	S.P. 1-2
		Ran a Critical (1-D)		
4	11:20	ET-5	C.I. 129-370	C.I. 0-122
5	11:58	EHS-58	NEW	129-370
		Ran a Critical (1-E)		
6	13:11	EHS-58	C.I. 129-370	0-145
7	13:19	ET-5	0-122	129-370
8	13:27	ET-21	109-311	0-122
9	13:47	EHS-58	0-145	109-311
		Ran a Critical (1-F)		
10	14:35	EHS-58	C.I. 109-311	0-145
11	14:42	ET-21	0-122	109-311
12	14:47	ET-27	95-294	0-122
13	14:53	EHS-58	0-145	95-294
		Ran a Critical (1-G)		
14	15:41	EHS-58	C.I. 95-294	0-145
15	15:47	ET-27	0-122	95-294
16	15:54	ET-40	50-259	0-122
17	16:00	EHS-58	0-145	59-259
		Ran a Critical (1-H)		
	9/6/61	Ran a Critical (1-H _a)		
1	13:06	EHS-58	C.I. 59-259	C.I. 0-145
2	13:13	ET-40	0-122	59-259
3	13:24	EH-31	34-237	0-122
4	13:32	EHS-58	0-145	34-237
		Ran a Critical (1-I)		
5	14:14	EHS-58	C.I. 34-237	0-145
6	14:20	EH-31	0-122	34-237
7	14:26	ET-31	80-237	0-122
8	14:49	EHS-58	0-145	80-237
		Ran a Critical (1-K)		
9	16:17	EHS-58	C.I. 80-237	0-145
10	16:23	ET-31	0-122	80-237
11	16:29	EH-44	47-203	0-122
12	16:34	EHS-58	0-145	47-203
		Ran a Critical (1-L)		
	9/7/61	Ran a Critical (1-L _a)		
1	09:35	EHS-58	C.I. 47-203	S.P. A-3
2	09:44	S.D.	S.P. 1-2	C.I. 0-145
3	09:50	EH-44	C.I. 0-122	47-203
4	09:56	S.D.	S.P. 1-1	0-122
		Ran a Critical (1-D _a)		
5	10:45	EH-IFA-II	Rotated 90 degrees clockwise.	

Table 100D-III

SUMMARY OF EXPERIMENTAL DATA OBTAINED DURING MOVEMENT OF FUEL

Critical No.	Remarks	ks	1	2	3	4	5	6	7	8	9	10	11	12
			No. 9 Rod Position, in.	Power, $\times 10^{-10}$	Asymptotic Period, sec	$\rho \times 10^5$ in Period	Δ Rod Position, in.	ρ of Δ Rod Position	$\delta\rho$ (ET $\rightarrow$ T)	$\delta\rho$ (EH $\rightarrow$ T)	S	$(1-S)\Sigma_T^*$	$S\Sigma_{EH}^*$	$\Sigma_{f(ET)}^*$
D	Initial condition. Only change from standard loading is transfer of S.D. to S.P.		11.65	50	3500	2.16	0	0	2.16	-----	-----	-----	-----	-----
1-D	Initial condition. (Same as D with different H_3BO_3 concentration.)		16.27	55.8	∞	-0.04	0	0	-----	-----	-----	-----	-----	-----
E	T element substituted in position 129-370. TE to position 0-122.		14.01	51.1	∞	-0.47	-2.36	-126	-126.5	-----	0.532	-0.0041	0.01395	0.00972
1-E	KE element substituted in position 129-370. TE to position 0-122.		13.53	56	∞	-0.04	2.74	144	-----	0.261				
F	T element substituted in position 109-311.		15	52	8000	0.9	-3.33	-178	177	-----	0.527	-0.0043	0.01331	0.0090
1-F	KE element substituted in position 109-311.		13.29	56	4000	+1.9	+2.98	+156	-----	336				
G	T element substituted in position 95-294.		14.91	51-53	30,000	>0.5	-3.26	-173.8	-173.8	-----	0.509	-0.0045	0.01286	0.00836
1-G	EK element substituted in position 95-294.		13.12	54	-4000	-0.2	+3.15	166	-----	342				
H	T element substituted in position 59-259.		14.41	50.5	∞	-0.5	-2.74	-147.5	-148	-----	0.552	0.0041	0.01394	0.00984
1-H	KE element substituted in position 59-259.		14.01	56	3400	+2.2	+2.26	+118	-----	268				
I	T element substituted in position 34-237.		13	50.3	∞	-0.5	-1.33	-98.6	-99.1	-----	0.656	0.0031	0.01657	0.0135
1-I	KE element substituted in position 34-237.		15.35	55-65	1300	+5.8	+0.92	+47	-----	151				
J	T element substituted in position 101-266.		15.11	50.5	6000	+1.3	-3.46	-184.7	-186	350	0.531	0.0043	0.01341	0.0092
1-J	None.		---	---	---	---	---	---	-----	-----				
K	T element substituted in position 80-237.		13.90	50.6	∞	-0.5	-2.25	-120	-120	-----	0.494	-0.0046	0.01248	0.0079
1-K	KE element substituted in position 80-237.		13.97	55	<4000	$\sim$ +2	+2.30	+121	-----	+243				
L	T element substituted in position 47-203.		12.26	50	8600	-0.9	-0.61	-37	-37.9	-----	0.442	-0.0051	0.0116	0.00606
1-L	KE element substituted in position 47-203.		15.41	58	2000	3.8	+0.86	+44	-----	+85.7				
E _a	Repeat of test E.		14.12	49.9	∞	-0.48	-2.46	-118	120	-----	---	-----	-----	-----
1-H _a	Repeat of test 1-H.		14	55	∞	-0.04	+2.27	+119	119	-----	---	-----	-----	-----
1-L _a	Repeat of test 1-L.		15.23	55	∞	-0.04	+1.04	+54	+54	-----	---	-----	-----	-----
1-D _a	Repeat of test 1-D.		16.22	55	∞	-0.04	+0.07	+3.2	+3.16	-----	---	-----	-----	-----
K _a	Repeat of test K.		13.91	50	∞	-0.5	-2.25	-120.5	-121	-----	---	-----	-----	-----

Calculations

A summary of the experimental data and its interpretation is presented in Table 100D-III. The significance of the columns in the table are discussed;

- Column 1. No. 9 Rod Position - The reactivity differences were determined from the No. 9 rod position. (See Reference 1 for calibration data.) To allow the No. 9 rod to operate in its calibrated range (8-20 in. from the bottom of the reactor), H_3BO_3 had to be added to the reactor water. The two series of tests (substitution of thin natural and subsequently thick enriched elements) had different H_3BO_3 concentrations, thus the No. 9 rod zero position for the two series [Criticals (E) and (1-E)] are different.
- Column 2. Power - The scale on the L&N linear recorder at which the Criticals were obtained, in general, was on the 50×10^{-10} scale or about a factor of 10^5 above the source level.
- Column 3. Asymptotic Period - Experimental time could be saved if instead of adjusting the rod to get a steady power, slight variations were recorded and the period calculated (see Reference 1). Periods above 10,000 sec were considered to be infinite.
- Column 4. Reactivity in Period (Given in units of $\rho \times 10^5$ corresponding to $\sim 0.14\%$) - Where the period was infinite the indicated reactivity was due to the source contribution.
- Column 5. Δ Rod Position - The displacement of the No. 9 rod from the zero position [Criticals (D) and (1-D)].
- Column 6. Reactivity Worth in $\rho \times 10^5$ of the Rod Displacement - Obtained from the integrated worth of the No. 9 rod, Fig. 100D-2.
- Column 7. $\frac{\delta\rho}{(ET \rightarrow T)}$ - The total reactivity difference when a thin enriched (ET) element is replaced by a thin natural (T) element.
- Column 8. $\frac{\delta\rho}{(EH \rightarrow T)}$ - Reactivity difference between the thick enriched (EH) and thin natural (T) elements.
- Column 9. $S = \frac{(ET \rightarrow T)}{(ET \rightarrow T)}$ - This ratio is useful since we assume Σ^* of the three elements to be directly proportional to their reactivity effect. Σ_{ET}^* is thus obtained by the formula:

$$\Sigma_{ET}^* = (1 - S)\Sigma_T^* + S\Sigma_{EH}^*$$

Column 10. $(1 - S)\Sigma_T^*$ - Where $\Sigma_T^* = -0.0093$ and has been obtained as outlined in the section titled THEORY. The pertinent parameters in the calculation are given in Table 100D-IV.

Table 100D-IV
CONSTANTS FOR CALCULATING Σ_T^*

Group No.	Σ_{rem}	D	κ	q	$2[q_{(R)} - q_{(E)}]$	$\frac{q_{(E)}}{q_{3(R)}}$	M^*	$\frac{\bar{\phi}_{(E)}}{\bar{\phi}_{3(R)}}$
2	0.06254	0.485	0.359	0.7275	-0.04672	1.826	-0.0213	1.805
3	0.0592	0.196	0.55	0.6003	-0.411	1.5205	-0.1639	1.352

where:

$$M = \frac{2[q_{(E)} - q_{(R)}]}{R \kappa_{(E)} q_{3(R)}} \left\{ \frac{I_1 [\kappa_{(E)} L]}{I_0 [\kappa_{(E)} L] + \frac{\kappa_{(E)} K_0 [\kappa_{(R)} L]}{\kappa_{(R)} K_1 [\kappa_{(R)} L]} I_1 [\kappa_{(E)} L]} \right\}$$

Column 11. $S\Sigma_{EH}^*$ (Where $\Sigma_{EH}^* = +0.0253$) - The parameters in the calculation are given in Table 100D-V.

Table 100D-V
CONSTANTS FOR CALCULATING Σ_{EH}^*

Group No.	Σ_{rem}	D	κ	q	$2[q_{(R)} - q_{(E)}]$	$\frac{q_{(E)}}{q_{3(R)}}$	M	$\frac{\bar{\phi}_{(E)}}{\bar{\phi}_{3(R)}}$
2	0.07988	0.607	0.363	0.5086	+0.2752	1.288	+0.144	1.4321
3	0.1005	0.2395	0.647	0.3254	+0.1388	0.8242	+0.0466	0.8708

Column 12. Σ_{ET}^* - The calculated value of Σ_{ET}^* versus burnup in Mwd is given in Fig.100D-4. The latter calculation was obtained in the

standard (though tedious) way of going through the depletion calculations and evaluating the over-all cross sections at 2000 reactor Mwd steps. Again, simplifications are almost unavoidable and the one used here is that the flux in the element does not change with burnup and so the flux ratios in the computation of Σ_{ET}^* are equal to 1.

Fig. 100D-4. Experimentally Measured EBWR Burnup (Radial)

DISCUSSION

The method tried in this experiment is an attractively simple one - namely, to measure the worth of fuel elements by using their actual core positions as a reactivity measuring facility. Further, in a qualitative sense the experiment was moderately successful, that is, it did measure reactivity differences between ET elements in rough agreement with previous determinations; (2) and the average burnup of 2400 Mwd/ton compares favorably with the calculated average of 2200 Mwd/ton. However, the purpose of the experiment was not such qualitative comparisons but an attempt to find an experimental method exact enough so that theoretically calculated burnups (i.e., by the TURBO code) could be checked out. This was not achieved, and, at least for the EBWR, the inherent experimental inaccuracies and the necessary calculational simplifications limit this method to rough reactivity approximations. This need not be the case for other reactors, so the main error sources are described and methods suggested to minimize these errors.

Principal Error Sources

Reactivity Measurement - As described in a previous report (1) the maximum reactivity measurement error under most favorable measurement conditions for the EBWR is $\sim 0.003\% \rho$ (this estimate does not include possible nonlinearities in the flux measuring instrument nor errors in the as-indicated versus actual rod position). These favorable conditions were

not met completely for the present series of experiments since, in order to permit fuel transfer, the whole rod bank had to be moved between criticality determinations. This introduces errors due to repositioning of eight more rods which errors, if cumulative, could amount to $\sim 0.007\% \rho$.

A statistical measure of the errors in this experiment is provided by the five duplicate Criticals which were run. A maximum disagreement of $0.0065\% \rho$ was obtained for tests E and E_a, the arithmetical average of all of the differences being $\sim 0.003\% \rho$. These values are rather close to the estimated possible maximum of $0.01\% \rho$, and considering the fact that to accumulate such an error would necessitate a coincidence of about ten events (rods being displaced in the same direction, etc.), the suspicion arises again that the unknown errors due to instrumentation might be appreciable.

The errors above are due to the mechanical aspects of the measurements; unfortunately, in the present experiment three more error sources exist, they are:

1. The change in the No. 9 rod differential worth as elements are changed in the reactor (especially pertinent for the central elements).
2. The reactivity effect of the element changes at the edge of the reactor.
3. The difference in the effects of Σ_f and Σ_a on over-all rod worth.

To estimate the first uncertainty, the No. 9 rod would have to be recalibrated each time an element change was performed. Because of time limitations this was not done. The second uncertainty was tested experimentally but the reactivity difference measured was within the average statistical ($0.003\% \rho$) error spread and consequently uncertain in the absolute sense.

The third uncertainty, in effect, put the assumption of direct proportionality between Σ^* and the reactivity effect in question for reactors containing control rods. In defining Σ^* the fission and absorption cross sections are accorded equal importance (aside from preferential leakage of fast neutrons this is valid within the perturbation theory); however, the worth of control rods react differently to local changes in Σ_f or Σ_a . This can be demonstrated with some control models (i.e., absorption area), and can also be shown in a qualitative sense. The difference lies in the fact that absorption, being mostly thermal, affects a rather small area and produces severe but localized flux perturbations. Fission, however, since it produces fast neutrons, affects a larger area but produces less severe local disturbances. We say, the effect of a local change in Σ_f "spreads" further. The control rods of course are strong local absorbers and thus it can be reasoned that a flux at a control rod will be influenced more by a nearby change in Σ_a than

by an equivalent change in Σ_f . Thus, one could expect the total control rod worth to change differently for different Σ_f/Σ_a ratios. As pointed out in Reference 3 such changes in total rod worth can have a very severe influence on the validity of reactivity measurements.

Summary

1. For the reactivity measurements a maximum error of $\sim 0.015\% \rho$ could exist. This amounts to a 12 - 40% variance (depending on the absolute value of the reactivity difference) of the reactivity measured, and in terms of final Mwd/ton for the ET elements represents an uncertainty of 400 to 1200 Mwd/ton. Estimating on the basis of the average statistical error ($0.003\% \rho$), this becomes ~ 100 to 240 Mwd/ton.
2. The calculated parameter Σ^* has several weaknesses: (a) it is obtained as a difference of numbers usually $\sim$ five times larger; (b) it required theoretically obtained flux ratios; and (c) it effects control rod worth unpredictably.

The first weakness is serious in the absolute sense, that is, in the calculation of an a priori Σ^* , an error in the basic constants could be magnified by a factor of 5 in the end result. For relative measurements, as done in this case, this is not important since the same constants are used for computing both of the standards (Σ_T^* and Σ_{EH}^*) and in the interpolated end result (Σ_{ET}) the errors would nearly cancel. However, since the T and EH elements have different spectra and different macroscopic constants, the error does not completely cancel. This is a second order error and is neglected in this experiment.

The calculations of the flux ratios are beset by all the weaknesses that a simplified theory in a complex reactor can inherit. As is well known, in such cases accuracy goes up only as the fourth root of the effort expended. This would especially be true in this case since in order to improve accuracy significantly one would not go on to higher order approximations (i.e., P_3 , etc.) but would have to reproduce the actual element geometry more exactly, taking into account control rods, finiteness of reactor, and so on. This is clearly an unattractive proposition. Errors in the flux ratios (both $\bar{\phi}_3(E)/\bar{\phi}_3(R)$ and $\bar{\phi}_2(E)/\bar{\phi}_3(R)$) are directly proportional to errors in Σ^* . Thus an error of 10% in a flux ratio could cause an uncertainty of ± 300 Mwd/ton in the final experimentally obtained burnup.

Minimization of Uncertainties

The uncertainties outlined could be reduced by an order of magnitude, which would make this type of reactivity measuring method a quantitative one. The required procedures are not applicable to the EBWR because of complexity and set safety standards. Some of the possibilities are:

For flux-ratios:

1. Do the required detailed geometry calculations.
2. Make flux measurements along with the reactivity measurements.
3. Get a bigger and better (more uniform) reactor, in which the simplifying assumptions made are more accurate.

For reactivity measurement errors:

1. Keep rods Nos. 1-8 stationary while changing fuel elements and move No. 9 rod only.
2. Oscillate the elements being tested and measure the flux modulation rather than the reactivity directly.
3. Design a low worth special control rod for fine reactivity measurements.

REFERENCES

- (1) EBWR Test Report No. 100E.
- (2) "EBWR Test Reports," ANL-6229, (November 1960), p. 172.
- (3) EBWR Test Report No. 100F.

REACTIVITY EFFECT OF ROTATING THE BORON-STAINLESS STEEL STRIP CONTAINING SPIKE ELEMENTS

(Test Report No. 100F, dated November 24, 1961, by K. Almenas)

SUMMARY

The present 20-spike loading was readjusted to conform to the final 1A core loading by rotating 9 of the spikes in place. In all cases the purpose of the rotation was to reduce the cold, rods-in reactivity so that the 8-rod subcriticality condition could be met in the fully loaded 1A core. This was achieved for all of the rotations, the total reactivity loss being 0.14% ρ .

PROCEDURE

The 9 spikes were rotated in 3 steps, the reactivity loss after each rotation being measured by determining the critical No. 9 rod position. (Remaining rods banked at 21.1 in., see Reference 1.) The four criticals obtained are listed in Table 100F-I.

Table 100F-I

EXPERIMENTAL CONDITIONS OF CRITICALS DURING SPIKE ROTATION

	Average Power $\times 10^{-10}$	Asymptotic Period (T), sec	$\rho \times 10^5$ in T	No. 9 Rod Position, in.	Δ No. 9 Rod Position	$\rho \times 10^5$ in Δ Rod Position
Critical No. 1 Prior to rotation of spikes rods Nos. 1-8 at 21.1 in. ~ 1.35 gm/gal H_3BO_3 present	150	∞	0	18	0	0
Critical No. 2 After rotating 4 three-strip spikes	~ 150	5050	1.5	19.11	1.11	51.1
Critical No. 3 After rotating 2 two-strip spikes	~ 150	-5000	-1.6	20.29	2.29	$51.1 + 51.3 = 102.4$
Critical No. 4 After rotating 3 two-strip spikes	~ 150	- 870	-9.0	21.4	3.4	$102.4 + 44 = 146.4$

The purpose of the rotations was to reduce the cold, rods-in reactivity. For example, the three-strip spikes were rotated by 180 deg so that their final position would be as shown below:

This reduced the proximity of the middle strip to the control rod and decreased their mutual self shielding, thus in the rotated configuration both strip and rod were more effective. The loss in reactivity was $0.013\% \rho$ per rotation of this type.

The two strip elements were rotated by 90 deg to obtain final configurations as shown below:

The main reactivity loss for these cases resulted from the more direct introduction of the poison strips into the high importance spike region. The reactivity loss was $0.016\% \rho$ per this type of rotation.

DISCUSSION

Total reactivity lost from spike rotation amounted to $0.14\% \rho$ and conformed to pre-experimental estimates.

Since there is some confusion concerning the meaning and use of such reactivity measurements, the number above is defined here as to what it does and what it does not mean and the importance of the distinction is illustrated.

The number does mean that if the rods were at their pre-rotational position (reactor critical) and somehow, suddenly, the spikes were rotated, the reactor would become subcritical by $-0.14\% \rho$, and its flux would decrease with a corresponding negative period (~ 110 sec in this case). The measurement has a practical use since it tells how the rods will move when, as during the loading of core 1A (the rods were close to the bottom of the core during fuel loadings), the spikes are rotated. Specifically, it tells how the spikes should be rotated so that the rod and boron-stainless steel strip control system will absorb the maximum amount of reactivity so that the 8-rod shutdown condition could be met.

The number does not mean that the reactivity of the core has been decreased by $0.14\% \rho$, nor that the pre-rotational core had a higher total reactivity by $0.14\% \rho$ than the present configuration.

Before the above statements appear like outright contradictions it should be realized that too often two quite different reactivity aspects of a reactor are used interchangeably. The two separate aspects being:

1. The reactivity of the core-control rod system.
2. The reactivity of the core itself.

How different these two concepts are can be illustrated by a somewhat idealized example, the parameters of which, however, apply quite well to the present case.

Given a reactor with a control rod having a perfectly constant differential worth from the top to the bottom of the reactor. Suppose that an experiment of the type described in this report is performed, that is

1. Establish a critical rod position at condition on (C-I).
2. Perform a change on the core (i.e., move an element).
3. Establish a new critical rod position at condition two (C-II).

Choosing parameters in line with the present experiment we might have

Rod worth $[\Delta \rho / \Delta x]$ at C-I = $0.1\$/in.$

Calculating the reactivity change from Step 1 to Step 3 above in a common sense way we get

$$\Delta \rho (\text{Step 2}) = (\Delta \rho / \Delta x) \Delta x_{(I \rightarrow II)} = 0.1\$$$

An additional complication which is inevitable is that the differential rod worth itself changes somewhat due to the changes performed in the core during Step 2. This change might be small, say -5% and as such within the measurement error of the differential rod worth. Suppose, however, it was

measured exactly; this would affect the just obtained results in a small way only, namely now the reactivity change could be interpreted in two ways:

$$\Delta \rho \left[\begin{array}{l} \text{from C-I, rod at 30 in.} \\ \text{to C-II, rod at 30 in.} \end{array} \right] = -0.095\%$$

$$\Delta \rho \left[\begin{array}{l} \text{from C-II, rod at 29 in.} \\ \text{to C-I, rod at 29 in.} \end{array} \right] = +0.1\%$$

The absolute difference of 0.005% being due to the changing worth of $\Delta x = 1$ in.

Defined in the way above both of the reactivity changes are correct and refer to the asymptotic periods which would result after performing the indicated change with the rod at either 29 or 30 in.; however, many times reactivity is interpreted in a more absolute sense - namely it is taken to be the total reactivity of the rod free core. As seen immediately such an interpretation yields very different results.

$$\begin{array}{lcl} \text{Total reactivity held down by rod at C-I} & = 30 \times 0.1\% & = 3.0\% \\ \text{Total reactivity held down by rod at C-II} & = 29 \times 0.095 & = \underline{2.755\%} \\ \text{Difference in core reactivity from C-I to C-II} & & = -0.245\% \end{array}$$

As seen, the result is well over twice as large. This again is no contradiction, what we have obtained is simply the reactivity difference at completely different (rod free) conditions. A more important aspect of this last result is that it is almost 100% uncertain, since it was obtained as a difference of two large numbers both of which, even under optimum conditions, have uncertainties of $\sim 5\%$.

In the comparison above all of the parameters employed were completely "reasonable" and could very well apply to this experiment. The single rod of constant worth feature just makes the difference more self-evident, the same conclusions would apply for multiple rods of changing differential worth. To further draw on the experiment for an illustration, part of the reactivity loss was achieved by changing configuration C-I to configuration C-II.

100F-SK-4

The reason for loss of reactivity with the control rod in place is quite obvious, namely the middle strip and the control rod blade shield each other in configuration C-I; however, this shielding is reduced in configuration C-II. Taking the control rod away removes the prime reason for reactivity loss and thus obviously we cannot think of this loss as being the property of the reactor but only in the context that it was measured, namely, as the property of the reactor-control rod system.

Attachment A

SOURCE STRENGTH CALCULATIONS

The experiments reported here afforded an opportunity to make criticality measurements in the presence of a strong source (a newly re-activated source had been installed three weeks previously). Also, two independent methods of measuring source strength could be checked against each other.

Dynamic (Reference 1)

If sufficiently long periods are measured, the source strength can be obtained from the asymptotic period kinetics equation:

$$\frac{dn(t)}{dt} = \frac{n(t)}{T} + S ,$$

where:

T = asymptotic period

S = source strength (n/sec) or $\left[\frac{\text{units of measuring instrument}}{\text{sec}} \right]$

If $dn(t)/dt$ is now plotted versus $n(t)$, a straight line with slope $1/T$ and intercept S results. This is shown for 3 long positive periods (163, 193, and 300 sec) on Fig. 100F-1. The units used are readings of the linear flux recorder on the 10^{-10} amp scale.

Fig. 100F-1. $dn(t)/dt$ Versus $n(t)$ for Positive Periods

The three values of S obtained are: $0.041 \times 10^{-10}/\text{sec}$ and $0.03 \times 10^{-10}/\text{sec}$ and $0.028 \times 10^{-10}/\text{sec}$. Average $S = 0.033 \times 10^{-10}$.

Subcritical (Reference 2)

If power levels at two different subcriticalities are measured, and if the change in subcriticality is known, S can be calculated as follows:

$$[S\ell^*] = \frac{\Delta k_{\text{eff}}}{(1/n_1) - (1/n_2)}$$

where

$$\Delta k_{\text{eff}} = \frac{-\Delta\rho k_{\text{eff}}^2}{1 + \Delta\rho k_{\text{eff}}}$$

$\ell^* = \text{effective neutron lifetime} = 0.079 \text{ sec}$

$n_1 = \text{total neutron population at subcritical condition 1, assumed to be proportional to measuring instrument response.}$

Two subcritical power levels were held during Critical No. 1, and the reactivity difference (obtained from displacement of the No. 9 rod) was measured. The data was

$$n_1 = 1.4 \times 10^{-10} \quad \text{No. 9 rod at 14 in.}$$

$$n_2 = 3.1 \times 10^{-10} \quad \text{No. 9 rod at 16 in.}$$

$$\Delta \rho = 0.00107 \quad k_{\text{eff}} \approx 0.99$$

$$\therefore \Delta k_{\text{eff}} = 0.00105$$

$$[S\ell^*] = \frac{0.00105 \times 10^{-10}}{(1/1.4) - (1/3.4)} = \frac{0.00105 \times 10^{-10}}{0.3916} = 0.00268 \times 10^{-10}$$

$$S = 0.034 \times 10^{-10} / \text{sec.}$$

REFERENCES

- (1) EBWR Test Report No. 100D.
- (2) EBWR Test Report No. 108C.

COMPLETION OF LOADING 1A, WITH THE ADDITION OF
FOUR PERIPHERAL SPIKES AND ONE ET ELEMENT

DESTRIPPING OF EIGHT SPIKES BY ONE STRIP EACH

VERIFICATION OF THE REACTOR MEETING THE
COLD SHUTDOWN REQUIREMENTS

A PARTIAL CALIBRATION OF THE CONTROL ROD SYSTEM

(Test Report No. 100J, dated March 8, 1962, by H. P. Iskenderian)

A constant concentration of H_3BO_3 was maintained during most of the above operations for economy of time. See Table 100J-I for information on operations performed, concentrations of H_3BO_3 , rod positions, and changes in reactivities.

Table 100J-I

COMPLETION OF LOADING 1A AND DESTRIPPING OPERATION

Operation	H ₃ BO ₃ Concentration (gm/gal of H ₂ O)	Rod Position for Bank Crit. (in.)	Change in Reactivity, %
	0.38	12.05	
	Zero	11.2*	
	1.78	15.48	-1.523
Add 2 Spikes (Peripherally) Plus 1 ET	1.78	15.0	+0.17
Add 2 Spikes (Peripherally)	1.78	14.7	+0.12
Add 0.74 gm/gal H ₃ BO ₃	2.52	16.9	-0.786
Destrip S-28 and S-7 by 1 Strip Each	2.52	16.2	+0.23
Destrip S-15 and S-8 by 1 Strip Each	2.52	15.7	+0.23
Destrip S-16 and S-10 by 1 Strip Each	2.52	14.82	+0.23
Add 0.95 gm/gal of H ₃ BO ₃	3.47	17.2	-0.992
Destrip S-11 and S-12 by 1 Strip Each	3.47	16.5†	+0.23
Check 8-Rod Shutdown Requirement for all Rods	3.47		
Reduce H ₃ BO ₃ Content to -	2.36	13.7	+1.161
	1.68	12.3	+0.724
	1.135	11.15	+0.589
	0.80	10.46	+0.365
	0.57	10.0	+0.252
	0.42	9.65	+0.165
	Zero	8.8*	+0.165

* Extrapolated

† Rod No. 9 at 17.1 in.

COMPLETION OF LOADING 1A

Retaining a constant concentration of H₃BO₃ of 1.78 gm/gal of H₂O, four spikes with two and three strips per pair were added to the periphery of the core; and one ET element was also added to the core. These additions were made in two steps with accompanying bank criticals, and the results were in near agreement with anticipated reactivity gain or bank control rod movements. The gain in reactivity was 0.29% for the above operation (see Table 100J-I).

DESTRIPPING OF EIGHT SPIKES

Destripping operations were performed taking bank criticalities after every destripping of two spikes. The worth of a strip was measured to be $0.115\% \rho$ on the basis of the control rod calibration of 1/22/62. At the end of eight destripping operations, the reactor was bank critical at 16.5 in.; and with 3.47 gm/gal of H_3BO_3 , the reactor met the 8-rod shutdown requirement. It was anticipated that it could also meet the 9-rod cold shutdown requirement; however, having stored $\sim 1.2\% \rho$ in the above operations alone, the 9-rod cold shutdown was verified as follows.

VERIFICATION OF 8 AND 9-ROD COLD SHUTDOWN

The H_3BO_3 content was increased to a value of 3.47 gm/gal of H_2O and the 8-rod shutdown requirement was checked (see Table 100J-I).

In order to verify that the present loading would also meet the 9-rod shutdown requirement with zero H_3BO_3 , the above H_3BO_3 content of 3.47 gm/gal was reduced by the addition and drainage of water in steps of 1000 gallons. The actual values of H_3BO_3 concentrations, bank positions (for criticality), and corresponding reactivity changes are all listed in the lower part of Table 100J-I. From Fig. 100J-1 it is also seen that the zero H_3BO_3 extrapolation corresponds to a control rod position of 8.7 in. (estimated value was ~ 8.5 in.).

The reactivity available in the present core with its ring of 28 spikes, having an average of 1.57 strip per spike, is estimated as follows.

Prior to the above operations and additions of $1.2\% \rho$, the reactor was bank critical, for a zero concentration of H_3BO_3 , at an extrapolated position of 11.2 in. In June 1961, the reactor was in operation at a 20-Mw power level for two hours, at a bank position of 21 in. for the control rods. The differential of about 10 in. in control rod position corresponds to $\sim 3.4\% \rho$. Adding the reactivity equivalent of $\sim 0.3\%$ for Xe buildup in two hours, the present core will have a reactivity in excess of that in June, 1961, by $\sim 3.7 + 1.2 = 4.9\%$ plus the differential rod worth of the present system over the old system with the fueled stainless steel followers, for the indicated positions of the rods. This available reactivity should take us near 100 Mw, if a linear extrapolation could be made.

Fig. 100J-1. EBWR Bank Criticalities Forming the Pattern of Core Loading 1A

PARTIAL CALIBRATION OF THE CONTROL RODS

A partial calibration of the bank of rods was obtained, as a function of the H_3BO_3 in the reactor and its reactivity effects (see Table 100J-I). The present calibration (2/24/62) refers to a core somewhat different from that when the last calibration (1/20/62) was obtained. However, the two calibrations agree well within the errors of the measurement, and there is some overlap (see Fig. 100J-2).

Calibration of No. 9 Rod by Period Measurements

A two point calibration of the central rod No. 9 (3 in. and 5 in. above the bottom of the core), when rods Nos. 1-8 were at a 10.5-in. position, was obtained (see Fig. 100J-3).

Equations:

$$n = \frac{dn}{dt} = \frac{n(t)}{T} + S \quad (1)$$

where

S = Fixed Source in Reactor.

T = Asymptotic Period in Seconds.

Fig. 100J-2. Bank Calibration of
EBWR Control Rods
(Zircaloy Followers)

From Eq. (1),

$$T = \frac{n_2 - n_1}{\dot{n}_2 - \dot{n}_1} \quad (2)$$

Calibration: Rods Nos. 1-8 @ 10.5 in.

(a) No. 9 Rod Raised to 3 in.

t, sec	40	50	60
$10^9 n$	52	60	68.5
$10^9 \dot{n}$	0.71	0.82	0.933

$$T = \frac{n_2 - n_1}{\dot{n}_2 - \dot{n}_1} \quad \text{----} \quad 72.7 \quad 75.2$$

From Inhour Curve (ρ/β versus T),

ρ/β	1.2	1.16
ρ	0.0786	0.076
Average ρ	0.077

PERIOD MEASUREMENTS: $\beta = 6.55 \times 10^{-3}$ USED

Fig. 100J-3. Worth of Central Rod No. 9 When Rods
(Nos. 1-8) Are at 10.5 in. (Above the
Bottom of the Core)

From Eq. (1), and calculated T, $S \approx 0.07 \times 10^{-10}$

(b) No. 9 Rod Raised to 5 in.

Periods, 24.4, 25.4 and 25.94 sec

Asymptotic Period 26 sec.

From Inhour Curve (ρ/β versus T),

$$\rho/\beta = 0.238$$

$$\rho = 0.157$$

Worth of Central Rod No. 9

With all rods at a 10 in. position, the worth of central rod as it is moved by about 0.5, 1.0, and 1.6 in. was also obtained (see Table 100J-II).

Table 100J-II

WORTH OF CENTRAL ROD NO. 9

When all Rods Nos. 1-8 are at 10 in., No. 9 Rod at 10.35 in.,	Reactor Critical
When all Rods Nos. 1-8 are at 10 in., No. 9 Rod at 10.85 in.,	$\delta\rho = 0.02\%$
When all Rods Nos. 1-8 are at 10 in., No. 9 Rod at 11.4 in.,	$\delta\rho = 0.05\%$
When all Rods Nos. 1-8 are at 10 in., No. 9 Rod at 12 in.,	$\delta\rho = 0.08\%$

Comments

The above control rod measurements are not complete as they had not been planned. The data was obtained incidental to the operations described in this report.

PHYSICS TERMINATION TESTS ON EBWR, CORE LOADING 1A

(Test Report No. 100N, dated January 14, 1963, by H. P. Iskenderian)

Concluding physics measurements were made on core loading 1A to obtain:

1. Power coefficient values over the range 40-60-80 Mw, with all of the rods out.

2. A partial calibration of the control rod system for the cold reactor, to complete previous partial calibrations, for purposes of making final reactivity calculations.

Power Coefficient Measurements

Power coefficient tests, over the range of 40-60 Mw, were made to check previous results which had been obtained with the central control rod at a 30 in. position over this (40-60 Mw) power range. For the purpose of completeness, results of earlier measurements (8/13/62) made over the power range 1-40 Mw have also been included. Results of power coefficient measurements from 60-80 Mw are also given.

In Table 100N-I are the details of measured data and calculated values of power coefficient,

$$\frac{(\Delta \rho / \Delta P)}{\left| \begin{array}{l} P_2 \\ P_1 \end{array} \right|} = \frac{\Delta \rho_{(CR)} + \Delta \rho_{(H_3BO_3)} + \Delta \rho_{(Xe)}}{\Delta P = P_2 - P_1}$$

Table 100N-I

CALCULATIONS OF POWER COEFFICIENT AT 40-60 Mw
(Water Level at 15 ft. 0 in. and all Rods Out; Measurements of 12/6/62)

Hour	Power, Mw	Conc. of Acid* in Hot Un- voided Coolant, gm/gal	Steam Void [†] Fraction in Coolant	$\alpha(1-\alpha)^{**}$	Change in Reactivity Due to				
					Xe poison	Acid Poison	$\Delta \rho_{total}$	ΔP	$\Delta \rho / \Delta P$
					$\Delta \rho_{Xe}$	$\Delta \rho_{(H_3BO_3)}$			
		(α)	(α)		(%)	(%)	(%)	(Mw)	(% ρ /Mw)
12:30	63.4	0.816	0.21	0.64					
14:05	45.6	2.656	0.13	2.31	-0.18	-1.67	-1.85	17.8	0.104
13:00	58.7	1.544	0.20	1.23					
14:05	45.6	2.656	0.13	2.31	-0.14	-1.08	-1.22	13.1	0.093
13:00	58.7	1.544	0.20	1.23					
15:20	35.1	3.624	0.10	3.26	-0.24	-2.03	-2.27	23.6	0.096
13:00	58.7	1.544	0.20	1.23					
13:34	41.9	3.016	0.125	2.64	-0.21	-1.41	-1.62	16.8	0.096

*These concentration values were obtained by multiplying the corresponding concentrations at measured room temperatures by the density of hot water (≈ 0.8).

[†]These void fractions are based on early hydrodynamic calculations which did not allow for "carryunder" of steam. Final analysis of the results of the measurements are to appear in a separate report.

**Average concentration of H_3BO_3 in hot voided coolant. One gm of H_3BO_3 per gal of water is worth $\sim 1\%$ of reactivity.

From the above data,

$$\overline{(\Delta \rho / \Delta P)} \bigg|_{40 \text{ Mw}}^{60 \text{ Mw}} \simeq 0.100\% \rho / \text{Mw}$$

Total reactivity required 0 Mw to 60 Mw $\simeq 0.10 \times 20 + 3.3$ (see Table 100N-II)
 $\simeq 5.3\%$

where:

$\Delta \rho$ refers to the differences in the reactivity worths of:

1. Control rod settings,
2. H_3BO_3 concentrations,
3. Xe poisonings,

for the power settings of P_1 and P_2 .

In making these calculations, it was known from previous measurements that the reactivity worths of H_3BO_3 was 1% ρ for a concentration of 1 gm/gal.

It is seen from the data of Table 100N-I, that in the power range of 40-60 Mw,

$$\overline{(\Delta \rho / \Delta P)} \bigg|_{40 \text{ Mw}}^{60 \text{ Mw}} \simeq 0.10$$

in agreement with previous results.⁽¹⁾

Fig. 100N-1

EBWR Operation 0-40 Mw (8/13/62)
before Final Destripping of 16 Spikes

Over the range of 60 to 80 Mw, where the reactor operated as a modified dual cycle boiling reactor, with appreciable "carryover," it was found (see Table 100N-Ia):

$$\overline{(\Delta\rho/\Delta P)} \left| \begin{array}{l} 80 \text{ Mw} \\ 60 \text{ Mw} \end{array} \right. \approx 0.055$$

For the sake of completeness, power coefficient measurements at lower powers of about 6-40 Mw have also been included in Table 100N-II (also see Fig. 100N-1). It was found:

$$\overline{(\Delta\rho/\Delta P)} \left| \begin{array}{l} 40 \text{ Mw} \\ 5.8 \text{ Mw} \end{array} \right. \approx 0.075 ,$$

which is in near agreement with the previous value.⁽¹⁾

Table 100N-Ia

CALCULATIONS OF POWER COEFFICIENT AT 60-80 Mw
(Water Level at 15 ft-3 in.; Measurements of 12/4/62)

Hour	Power, Mw	α	a	$\alpha(1-a)$	Bank of Rod Settings, in.	$\Delta\rho_{(CR)} 41.0 \text{ in.}$	$\Delta\rho_{(CR)} 41.0 \text{ in.}$	$\Delta\rho/\Delta P^*$
						cold 33.5 in.	hot 33.5 in.	
						%	%	
10:22	61.1	0.14	0.20	0.11	33.5			
11:13	81.1	0.14	0.30	0.10	41.0	0.83	1.2	0.06

*Effect of Xe has been neglected; its inclusion should reduce 0.06 further.

These power coefficient values in the lower power range 10-40 Mw are lower than the corresponding values at 40-60 Mw, as previously noted.⁽¹⁾

The reactivity versus power characteristic obtained from the above data is shown on Fig. 100N-2. It will be noted from Fig. 100N-2, that without carryover, the system operating as a direct cycle boiling reactor, $(\Delta\rho/\text{Mw})$ has an increasing slope with power.

Table 100N-II
 CALCULATION OF POWER COEFFICIENT AT 10-40 MW
 Measurement of 8/13/62 (see Fig. 100N-1)

Hour	Power Mw	Bank of Control Rod Position	$\Delta\rho$ (CR) cold	$\left. \begin{array}{c} P_1 \\ \hline P_2 \end{array} \right \Delta\rho$ hot	(1)	Conc. of H_3BO_3 Hot not Voided (gm/gal)	Steam* Void Fraction in Coolant	Conc. of H_3BO_3 in Hot Void Coolant	(2)	(3)	(1)+(2)+(3) ↓ $\Delta\rho$ total	$\Delta\rho$	$\Delta\rho/\Delta\rho$
		(in.)	(%)	(%)	(%)	(α)	(a)	$\alpha(1-a)$	(%)	(%)	(%)	(Mw)	% ρ /Mw
12:00	<1	23.6				3.88	small	3.88					
13:30	5.8	25.7	0.5	0.68		3.92	~ 3	3.80	0.08		0.76	5.8	0.131
13:30	5.8	25.7				3.92	~ 3	3.80					
14:55	21.7	29.8	0.85	1.15		4.13	7.3	3.82	-0.02		1.13	15.9	0.071
14:55	21.7	29.8				4.13	7.3	3.82					
15:47	29.8	32.8	0.40	0.58		4.13	9	3.76	0.06	- small	0.64	8.1	0.079
15:47	29.8	32.8				4.13	9	3.76					
16:15	40	36.6	0.55	0.72		4.13	12	3.64	0.12	-0.04	0.80	10.2	0.078

Loss of reactivity 0 to 40 Mw 3.33% ρ

Average power coefficient 10 to 40 Mw 0.075% ρ /Mw

* These void fractions are based on early hydrodynamic calculations which did not allow for "carryunder" of steam. Final analysis of the results of the measurements are to appear in a separate report.

Fig. 100N-2. Reactivity Versus Reactor Power

Calibration of Bank of Control Rods Nos. 1-9

A cold calibration of the control rod system was completed as shown on Fig. 100N-3. This calibration is based on bank criticality and period measurements obtained previously, and on the present calibration shown on Table 100N-III.

Fig. 100N-3. Bank Calibration of Control Rods with Core Loading 1A

Table 100N-III

PARTIAL CALIBRATION OF BANK, RODS NOS. 1-9
CORE WATER AT 44.4°C (112°F)

Bank Criticality Settings:

Concentration of H_3BO_3 , gm/gal	4.15	8.64	9.49
Bank criticality settings, in.	16.45	35.2	43.2

Period Runs:

With 9.49 gm/gal of H_3BO_3 concentration at bank criticality at 43.2 in.

When bank, rods Nos. 1-9 → 46 in.	<u>Period</u>	<u>Scale</u>
	43.5	3×10^{-10}
	39.1	10×10^{-10}
	38.3	3×10^{-9}

Asymptotic period $\simeq 38$ sec

$$\Delta\rho/\beta = 0.18$$

$$\Delta\rho = 0.12\%$$

When bank, rods Nos. 1-9 → 48 in.	<u>Period</u>	<u>Scale</u>
	41.2	3×10^{-10}
	39.4	10×10^{-10}
	36.4	3×10^{-9}

Asymptotic period = 36 sec

$$\Delta\rho/\beta = 0.19$$

$$\Delta\rho = 0.13\%$$

REFERENCES

- (1) EBWR Test Report 100M.

REACTIVITY ASPECTS OF EBWR CORE 1B

(Test Report No. 100H, dated February 1, 1962, by K. Almenas)

SUMMARY

The EBWR 1B core (24 centralized spike elements) was loaded and tested in the cold critical condition. The following answers regarding its reactivity properties were obtained:

1. The loading is subcritical by $\sim 2.4\% \rho$ with all the rods in and no H_3BO_3 in the reactor water.
2. The loading has 8-rod subcriticality with an H_3BO_3 concentration above 0.6 gm/gal.
3. The loading gained $\sim 2.7\% \rho$ in the cold critical state above the previous 20-spike loading. Since the present loading contains less boron-stainless steel strips, the gain at hot operating conditions will be larger.
4. It is estimated that the prescribed shutdown conditions can certainly be met after removal of 8 additional boron-stainless steel strips and could quite probably be met after removal of 12 or even 16 boron-stainless steel strips.
5. The optimized 1B loading thus could have a cold excess reactivity $\sim 4.5\%$ above that of the previous 20-spike loading.

PROCEDURE

Some practical aspects on the effectiveness of the loading and H_3BO_3 dilution methods are presented here.

Spike Loading

An attempt was made to keep a running check on the subcriticality of the reactor during the removal of boron-stainless steel strips by monitoring the subcritical count rate. Thus Fig. 100H-1 shows the combined effects of shifting 4 ET elements outward, removing 4 boron-stainless steel strips, and then shifting 4 spikes inward. The uncertainty inherent in the method is

Fig. 100H-1. Reactivity Changes during Loading of Core 1B

demonstrated by the fluctuation and the discrepancy in the reactor subcriticality as obtained by the two separate counters. Especially for the high subcriticalities $> 7\%$ where the count rates approached 100 counts/min, the statistical fluctuation becomes quite pronounced. The accuracy of these measurements could be improved substantially by taking longer counts (only 1 minute counts were taken) or by installing a stronger source. However, during important experimental phases when conditions closer to critical exist and consequently the count rates increase, the results of both of the counters are shown to be in satisfactory agreement. After the lowering of each spike, a general reactivity increasing trend can be seen and is estimated at $\sim 0.16\% \rho$ per boron-stainless steel strip removed and per spike shifted; this agrees quite well with estimated reactivity increases.

Removal of H_3BO_3 by Dilution

The final steps of the dilution procedure were as follows:

1. Diluent water was added in batches of ~ 250 gal only (about equal to the volume of water in the core and $\sim 7.5\%$ of the total water in the vessel).
2. Subcritical count rates were taken before and after each addition.
3. The change in subcriticality was monitored so that the prescribed maximum reactivity addition rate of $0.002\% \rho/\text{sec}$ would not be exceeded. As shown by Fig. 100H-2 this condition was met by a safe margin.

Fig. 100H-2. Dilution Rate Versus $1-k_{eff}$ (~ 250 -gal Water Added/5 min.)

4. Air was bubbled continuously to ensure fast mixing of the diluent water. This was found to affect the subcritical counts only slightly.
5. After addition of ~ 150 gal, critical rod positions were obtained and the concentration of H_3BO_3 analyzed. The data obtained for the four criticals are given in Table 100H-I; the change of subcriticality with H_3BO_3 concentration is given in Fig. 100H-3.

Table 100H-I

SUMMARY OF CRITICALS DURING REMOVAL OF H_3BO_3

Critical	H_3BO_3 Conc., gm/gal	Zero Rod Position	ρ @ Zero Pos. ($\times 10^{-5}$)	Super- critical Rod Posi- tion, in.	Asymp- totic Period, sec	ρ of Period ($\times 10^{-5}$)	Diff. Rod Worth $\rho \times 10^{-5}/in.$	Subcritical Count with Rods @ 0 in.	
								No. 1 Counter	No. 2 Counter
(A) After loading of 24 spikes	1.9	14.6	- 3	14.9	+22.7	167	567	140	280
(C) After removing 0.6 gm/gal H_3BO_3 by dil.	1.3	13.2	-32	13.5	+25.7	157	630	179	326
(H) After removing 0.42 gm/gal H_3BO_3 by dil.	.88	12.35	+18.5	12.6	+22.5	167.8	627	262	456
(N) After removing 0.34 gm/gal H_3BO_3 by dil.	.54	11.6	- 3	11.8	+33.7	131	670	344	549

Total change in H_3BO_3 concentration = 1.36 gm/gal

Total increase in rod worth = (3 in.) (0.624) = 1.87% ρ

Worth of H_3BO_3 = 1.4% ρ /gm/gal

Source Strength Estimation

An intrinsic and difficult part of monitoring $1-k_{\text{eff}}$ through the subcritical count rate is the estimation of effective source strength. The main difficulty lies in the fact that what is required is a measurement proportional to the integrated neutron flux; what is measured, however, is the response of a stationary (two in this case) instrument. Thus any spatial shift in the neutron flux will result in a changed instrument response, regardless of whether the total neutron flux has changed or not. These changes can be quite drastic, especially when such gross effects as control rod bank movements are involved. For example, the $[Sl*]$ (source strength times

Fig. 100H-3. $1-k_{\text{eff}}$ Versus H_3BO_3 Concentration

effective neutron lifetime) term in units of the No. 2 counter is estimated as ~ 66 counts/min with the rod bank at ~ 17 in. and as 15 counts/min with the rods fully in. It is thus evident that a source strength applies only at the rod position at which it was determined.

The most direct way of obtaining the source strength is by observing the change in a long period at a power low enough so that the source can exert a significant influence. This can be done only with a

critical reactor and consequently, does not apply to the subcritical rod position. For a subcritical rod position, a change in the count rate with an accompanying known change in k_{eff} must be available.

During the present experiment such data was obtained by monitoring count rates at different H_3BO_3 concentrations. Thus, from the data of Table 100H-I we obtain

$$[Sl*]_{\text{No. 2 counter}} = \frac{0.0169}{[1/(280)1.22 - 1/549]} = \frac{0.0169}{0.00115} = 14.7 \text{ counts/min}$$

Where $k_{\text{eff}} = \sim 0.96$ and the method used is outlined in Reference 1.

A further change in source strength, even at constant rod position, results from the change in H_3BO_3 concentration itself. This is due to the fact that a decrease in H_3BO_3 concentration reduces the shielding between the core and the instrument. Thus $[Sl*]$ would be expected to increase as H_3BO_3 concentration went down; this effect is shown in Fig. 100H-2 where $1-k_{\text{eff}}$, as determined from the count rates, lies consistently below the more direct determination involving the control rod worth. Since this results in a more conservative subcriticality estimate, no correction was made for this effect.

REFERENCES

- (1) EBWR Test Report No. 108C.

DESTRIPPING OF 16 MORE SPIKES FOR A GAIN OF $\sim 2\%$ REACTIVITY -
CHECK OF 9-ROD COLD SHUTDOWN REQUIREMENT

(Test Report No. 110R(B), dated October 8, 1962, by H. P. Iskenderian)

The results presented in Reference 1 indicated the need for further increases in core reactivity, in order to obtain reactor powers above the 60-Mw level. It was decided to obtain greater core reactivities by the destripping of spikes, without violation of the 8- and 9-rod cold shutdown requirements.

The destripping of spikes, resulting gains in reactivity, bank criticality, and other data are shown in Fig. 110R(B)-1. The same information is also summarized in Tables 110R(B)-I and II. A reactivity gain of 2.07% was obtained as indicated in Table 110R(B)-I and Fig. 110R(B)-1.

A check of the 9-rod cold shutdown requirement was made in the usual manner, by dilution of H_3BO_3 and by taking precaution not to overestimate the worth of rods below 7 to 8 in. from the bottom of the core.

In evaluating the rod worth below 7.5 in. the usual methods of "rod drive" or "rod drop" were not suitable, due mainly to relatively high uncertainties in the time scale, and limitations of instrumentation.

A method was afforded for estimating the worth of the bank, rods Nos. 1-9, below 7.5 in., from the differential worth data of the bank, rods Nos. 1-8, with rod No. 9 out. Moving out the central No. 9 rod (either completely or to a 40-in. position) was expected to have little effect on the differential worth of the remaining rods, Nos. 1-8; this allowed moving the bank, rods Nos. 1-8, down as low as 2 in., where the differential worth of the bank was determined.

The calibration of the banks, rods Nos. 1-8 and 1-9, are presented in Fig. 110R(B)-2. The worths of these banks are equal at the 7.3- and 9.05-in. positions, and again at the 12.35- and 12.95-in. positions. The remainder of

Table 110R(B)-I

DESTRIPPING OPERATION OF 16 SPIKES BY 1 STRIP EACH [SEE LOADING NO. 63⁽²⁾],
GAINS IN REACTIVITY ρ , BANK CRITICALITIES, AND DIFFERENTIAL WORTHS INDICATED

Date of Operations	9/19	9/19	9/20	9/20	9/20	9/21	9/21	9/26	9/26	9/26
Concentration of H_3BO_3 (gm/gal)	4.1	4.1	4.1	3.78	3.78	3.78	3.78	3.78	3.78	4.26
Spiques Destripped	S-3; S-4	S-1; S-2	S-30; S-25 S-26	----	S-16; S-18	S-29 S-13	S-14	S-7 * S-16	S-8	S-15
Resulting Gain in Reactivity (%)	0.24	0.24	0.33	----	0.30	0.21	0.12	0.26	0.15	0.22
New Bank Critical Position (in.)	18.85	18.3	17.51	16.71	16.08	15.6	15.37	14.8	14.5	15.19

* Spikes S-7 and S-10 were interchanged.

Period Runs

Rod Positions (in.): 15.37-15.75
T (sec) = 18
 $\rho/\beta\%$ = 0.302
 $\rho\%$ = 0.191
 $\rho/\text{in. } \%$ = 0.504

Period Runs

Rod Positions (in.): 15.9-15.6
T (sec) = 16
 $\rho/\beta\%$ = 0.305
 $\rho\%$ = 0.20
 $\rho/\text{in. } \%$ = 0.487

Total gain in reactivity after 16 destrippings of 16 spikes, 2.07%.

Table 110R(B)-II

DILUTION OF H_3BO_3 TO CHECK THE 9-ROD SHUTDOWN REQUIREMENT
 CALIBRATION OF BANKS RODS NOS. 1-9 AND 1-8 $\beta_{eff} = 6.34 \times 10^{-3}$

<u>Bank of rods (Nos. 1-9)</u>					
Concentration of H_3BO_3 (gm/gal)	4.26	3.2	2.4	1.325	0.7
Bank Critical (in.)	15.19	12.95	11.16	9.05	7.49
Period Runs: Rod Positions (in.)	15.19 → 15.6		11.16 → 11.36	9.05 → 9.30	7.49 → 7.80
T (sec)	16		40	24.5	35
$\rho/\beta\%$	0.322		0.192	0.258	0.209
$\rho\%$	0.204		0.121	0.163	0.132
$\rho/in. \%$	0.50		0.60	0.65	0.42
<u>Bank of rods (Nos. 1-8) Rod No. 9 out unless otherwise indicated.</u>					
Concentration of H_3BO_3 (gm/gal)	3.2		1.325	0.7	
Bank Critical (in.)	12.35		7.3	2 (Rod No. 9 @ 40 in.)	
Period Runs: Rod Positions (in.)	12.35 → 12.6		7.3 → 7.6	2 → 4	
T (sec)	29		40	35	
$\rho/\beta\%$	0.233		0.19	0.209	
$\rho\%$	0.147		0.121	0.133	
$\rho/in. \%$	0.59		0.40	0.066

Fig. 110R(B)1. EBWR Destripping Operations of 16 Spikes by One Strip Each

Fig. 110R(B)-2.

EBWR Core 1A (10/1/62)
Cold Criticality

the calibration procedure is based on period measurements (see Tables 110R(B)-I and II).

It is to be noted in Fig. 110R(B)-2 that the differential worth of rods, Nos. 1-8, is quite small at the 2-3-in. positions. Noting also that the worth of rods, Nos. 1-9, will always be greater than the worth of the bank, rods Nos. 1-8, the estimated bank worth of the rods, Nos. 1-9, at 0 to 7.0 in., is obtained. It is estimated that this worth is $\sim 0.9\% \rho$.

Further destrippings of at least four more spikes (to have a symmetrical core) could result in the possible violation of the 9-rod cold shutdown requirement.

The reactor, with the core as of this date, is now estimated to be capable of delivering ~ 80 Mw, if the wattless component of circulating steam, or other causes (such as growth in the fuel plates) do not become excessive.

REFERENCES

- (1) EBWR Test Report No. 100M.
- (2) EBWR Test Report No. 110R(A).

EBWR NEUTRON FLUX DISTRIBUTION DURING COLD CRITICAL OPERATION

(Test Report No. 100K, dated October 23, 1962, by W. G. Knapp)

INTRODUCTION

A study has been made of the neutron flux distribution in the cold, critical EBWR core at approximately 20 kw power, with different reactor control parameters. Changes in reactor control, for different tests, were made on control rod bank positions, on the soluble poison concentration in the reactor water, and in the number and location of boron-stainless steel strips attached to spike elements (see Reference 1). The results of these studies are shown in axial plots from specific assemblies, and isogram plots in specific planes included with this report.

PROCEDURE

Copper wire with a diameter of 0.25 mm was used in all tests. The wire was normal electrical grade, and no variation was encountered when Formvar coated wire was used as compared to bare wire. The activation foil wires were about 1.75 m long, with a soldered bead at one end for reference as described later. They were sheathed in a standard 1/8-in. OD stainless steel tubing, fitted at one end with a handling-sealing fixture made from a Swagelock fitting, a 8-in. length of aluminum pipe, and ring-shaped end fitting to facilitate handling and retrieval from the reactor (Fig. 100K-1); while the opposite end of the tubing was sealed with a rubber cement. The wire protruded about 5 cm from the cemented end.

The fixture was accurately located on the tubing, for each run, by the Swagelock ferrule which was located so that a 3/4-in. washer welded to the Swagelock nut was 6 in. from the upper end of the 1/8-in. tubing. The soldered bead was pulled against the upper end of the 1/8-in. tubing when the wire was pulled taut through the tubing and the cement was applied and dried at the lower end. Assembly of the handling-sealing fixture to the upper end of the tube, with the Swagelock nut-washer assembly, made the sheath water-tight, provided an accurately located surface for positioning the activation foil wires in the fuel assemblies, and, as previously mentioned, provided a handle for manipulation and retrieval.

With the reactor subcritical, and with the soluble poison (boric acid) at the desired concentration, up to 20 foil assemblies were positioned in selected locations in the core. The Swagelock nut-washer assembly was seated on the upper surface of the fuel plates (or rods), and the wire was then located from 6 in. above the upper edge of the plates to 6 in. below the lower edge of the plates. The reactor was brought to a power of about 20 kw, or, more accurately, to a linear flux level of 2.34×10^{-8} on the Keithley, for each run with the rods in (or nearly in) a bank position. Timing of the run was started as the power reached about 2×10^{-8} on the Keithley, and the reactor was scrammed 30 min later. After a few minutes delay the foil assemblies were removed and hung to one side of the reactor containment shell to dry. After several hours the assemblies were removed to the counting facility where each sheath was disassembled and the wire-foil removed and placed in a wire scanning-counting device. Again the soldered bead was used to accurately locate the wire in the counter. One-minute counts were taken at 4 and 2-in. intervals in most cases, starting at 6 in. below the bead, i.e., on a level with the upper edge of the fuel plates during activation of the wire foil.

A 3 x 3 in. NaI(Tl) scintillation detector was used to detect the 511-keV positron annihilation radiation of the 12.8-hr Cu^{64} . The peak was well defined and analysis was performed at a fixed window level and window width setting on an ANL single channel analyzer by counting for a specified length of time - usually one minute. Each determination was corrected only for a decay prior to the plotting of axial neutron distribution curves for each wire.

The normalization of points within a run, or even between two runs at the same conditions, is good. The great change in integrated value of an individual plot for different runs in a given assembly under different reactor control parameters is suggestive of a lesser degree of reproducibility or normalization. This may be due, at least in part, to variation in sensitivity of reactor flux detection instrumentation with the variation in control parameters, and, in part, to variation in analysis parameters. Normalization between runs was not an original requirement of these studies since only relative information was sought from within each run.

Prior to the runs made with the core configuration shown in Fig. 100K-13, the scintillation crystal used for all previous runs suddenly developed poor resolution difficulties, and the single-channel analyzer succumbed to an "electronic illness." Another detector and repair of the analyzer corrected these problems; but the normalization of further runs with the previous runs is not accurate due to the absence of secondary standardization of the equipment.

Prior to the last run, secondary standardization was attempted by 4π counting a selected portion of a wire-foil (following dissolution and deposition of an aliquot portion on a 4π counting disc). As it happened, this

secondary standardization was good only for the wire from which the sample was cut since the interchange of equipment had changed the sensitivity relative to prior runs; and following the first analysis of the last series, the electronic parameters of the single-channel analyzer again shifted, unknown to the technician, so that later wires were also run at a different, though uniform, sensitivity.

The short-lived Cu^{64} isotope does not lend itself to reruns a couple of days later after computations and plots indicate a discrepancy. On the other hand, since the obtaining of only relative information for a given run was the original goal, this has been accomplished. From the one 4π analysis made, which gave a flux of $4.78 \times 10^{10} \text{ n(t)/(cm}^2\text{)(sec)}$ in the area of highest flux [in the corner of the innermost assembly (ET-46) nearest the core center] it appears that the thermal neutron flux is accurately represented by the activated wire-foil. The data from this one wire-foil is plotted in Fig. 100K-18.

RESULTS

Both axial and isogram neutron flux distribution plots are used to convey the data obtained. The plots given in this report do not include all the plots obtained from the positions indicated with an "X" on Figs. 100K-6 and 13. The reactor control parameters can be ascertained by reference to Figs. 100K-6 or 13, as applicable, and from indicated control rod positions and boric acid concentrations given on the isogram plots.

Figure 100K-2 isograms were determined from the axial plots of Fig. 100K-7 in a vertical plane indicated by line A of Fig. 100K-6, with axial plot No. 1 located near the center of the core in ET-46, etc. Individual wire-foil positions in the isogram plot then correspond to a vertical line of plotted points. In addition, Fig. 100K-7 shows axial plots for the following locations:

- I. Position 101-160, east center, in assembly ET-10.
- II. Position 80-131, west center, in assembly ET-7.

Not shown are plots for the following:

1. Position 80-131, east center, in assembly ET-7.
2. Position 101-160, west center, in assembly ET-10.
3. Position 95-113, west center, in assembly ET-8.
4. Position 95-113, east center, in assembly ET-8.
5. Position 0-53, center, in assembly EH-35.

Plots for Nos. 1-3 were similar to plot No. 3, Fig. 100K-7; No. 4 was similar to plot No. 1, Fig. 100K-7; and No. 5 was not accurately located vertically in its position, but does show a flux maxima with a relative value of 7 and is similar to plot No. 7, Fig. 100K-7.

Figures 100K-3 and 4 are composite isogram plots of two almost identical runs made three weeks apart. Correlation between results from identical irradiation locations for the two runs was very good, particularly in view of the fact that the soluble poison concentration was about 30 ppm higher and the control rod bank position was about 0.35 in. higher for the second run. Axial plots are shown as follows:

Figure 100K-7 axial neutron distribution plots correspond to the isograms of Fig. 100K-3 numbered from center out, except as follows: plot No. 2, Fig. 100K-8, is from position 109-205, east center, but is almost identical to 129-264, west center, and 109-205, west center, plots for which are not included on Fig. 100K-7 but are used in Fig. 100K-2; plot 5, Fig. 100K-8, is from position 77-170, east center, but is almost identical to (though slightly higher than) 77-170, west center, which plot though not shown on Fig. 100K-7, is used in Fig. 100K-3.

Figure 100K-9 axial neutron distribution plots correspond to the isograms of Fig. 100K-4, numbered from the center out, except as follows: plot No. 1, Fig. 100K-9, is for position 129-264, northeast corner, but is almost identical to 129-264, southwest corner, which though not shown in Fig. 100K-9, is used in Fig. 100K-4; plot No. 6 may be displaced axially from correct position due to incomplete seating of the irradiation assembly in the fuel assembly. (Irradiation for Figs. 100K-4 and 9 were made in corner locations of plate elements, and a center location in the spike element.)

Not shown in the figures are axial plots of the following:

1. Position 101-160, both east and west center, in assembly ET-10.
2. Position 95-113, both east and west center, in assembly ET-7.
3. Position 80-131, both east and west center, in assembly ET-8.
4. Position 63-101, center, in assembly S-2.
5. Position 77-85, center, in assembly S-9.

Other than low flux maxima in the two spike elements, no unusual characteristic was encountered. The low level of the activity from assemblies S-2 and S-9 are reflected in the integrated values cited later.

Figure 100K-5 is an isogram plot from the axial plots shown in Fig. 100K-10, plot No. 3, Fig. 100K-10, is the position 95-188, east center, but is almost identical to 95-188, west center, not shown in Fig. 100K-10, but used

in Fig. 100K-5. As this was the first run made, several interesting locations were not used, such as in the spike elements. On Fig. 100K-10 plots are included for other locations as follows:

- I. Position 101-160, east center, in assembly ET-10.
- II. Position 80-131, east center, in assembly ET-7, almost identical to position 80-131, west center (not shown).
- III. Position 95-113, east center, in assembly ET-8, similar to position 95-113, west center (not shown).
- IV. Position 0-53, northeast corner, in assembly EH-35, incomplete due to erratic vertical positioning.

Since the ANL single-channel analyzer parameters were the same throughout for all the above runs, and reproducibility of results has been shown by a set of duplicate runs, the information of one run is assumed to be of relatively uniform level, activation-wise, for any other. Neither of the runs reported below are normalized to the above runs, however, due to previously mentioned equipment changes without secondary standardization.

Prior to the following runs the spike elements were partially de-stripped of boron-stainless steel strips as explained in Reference 1.

Figures 100K-11 and 100K-12 show the isograms of neutron flux distribution determined from the axial plots of Figs. 100K-14 and 15, in two different vertical planes of the core as shown in Fig. 100K-12. As usual, the individual plots are numbered from the center out. Not shown is a plot for the position marked "X" in Fig. 100K-13.

Figures 100K-16 and 17 show the isogram plots for the final run in which the analyzer parameters shifted, shortly after analysis was started, for reasons unknown. Since the count rate was significantly lowered by this shift it is difficult to assume a high reliability, particularly since the peak values of any given curve are low especially when compared to the level, for instance, at the upper end of the assemblies. These plots are included, therefore, more for their qualitative representation of the neutron flux distribution in the core than for any quantitative value. Figure 100K-18 does show one axial plot from this run which is not included in the isograms. It is representative of position 129-264, northeast corner, in assembly ET-46. The plot of 129-264, east center, was used for the preparation of Fig. 100K-17, to represent the approximate values expected in that corner. The plot in Fig. 100K-18 was obtained prior to the shift in analyzer parameters previously mentioned, and was standardized by sampling the 4π counting of a section of the wire, so that the plot is shown in terms of neutron flux rather than in relative values as used in all other plots. It is believed that this standardization value, $4.78 \times 10^{10} \text{ n(t)/(cm}^{-2}\text{)(sec}^{-1}\text{)}$ would also be applicable to the peak of curve No. 1, Fig. 100K-9, so that other points can be calculated proportionately.

INTEGRATED AREAS OF AXIAL NEUTRON FLUX PLOTS

The area of each of the axial flux plots was determined and is reported here (Tables 100K-I through IV), again in a relative fashion. This should be an indication of the relative power contribution of each assembly. For simplicity, reference will be made to the applicable isogram plot figure number and axial plot number, and will represent, first, each wire-foil shown on the isogram plots from left to right, followed by other values as labeled. Assembly numbers will also be given for further clarity. Where axial plots were not used in the preparation of isograms, they will be identified as to location, but will be included in the group representing that particular run. No integrated areas were obtained for the final runs shown in Figs. 100K-16, 17, and 18.

It is particularly interesting to note the different values of relative integrated flux for wire foils in identical positions in ET-46 under two different parameters of reactor control.

Fig. 100K-1. Upper End of Flux Wire Shield Assembly

Fig. 100K-2. Neutron Flux Isograms

(S.W. Quadrant along Line A (Fig. 100K-6). Boric Acid Concentration 125 ppm.
Control Rods Banked at 10.25 in.)

Fig. 100K-3. Neutron Flux Isograms

(S.W. Quadrant along Line A (Fig. 100K-6). Boric Acid Concentration 900 ppm.
Control Rods at 16.65 to 17.0 in.)

Fig. 100K-4. Neutron Flux Isograms
(Same As Fig. 100K-3 Except along Line B (Fig. 100K-6))

Fig. 100K-5. Neutron Flux Isograms

(S.W. Quadrant along line A (Fig. 100K-6). Boric Acid Concentration 1900 ppm.
Control Rods at 32.5 in. Except Rod No. 9 at 32.3 in.)

Fig. 100K-6. EBWR Core Loading No. 62, S. W. Quadrant and Adjacent Assemblies (Activations Plotted in Figs. 100K-1 through 5 Made along Lines A and B. Other Wire-Foils Simultaneously Irradiated in Several or All Positions Marked with An "X").

Fig. 100K-7.

Axial Neutron Flux Distribution

Fig. 100K-8.

Axial Neutron Flux Distribution

Fig. 100K-9.

Axial Neutron Flux Distribution

Fig. 100K-10.

Axial Neutron Flux Distribution

Fig. 100K-11. Neutron Flux Isograms

(S.W. Quadrant along Line A (Fig. 100K-13). Boric Acid Concentration 180 ppm.
Control Rods Banked at 7.55 in.)

Fig. 100K-12. Neutron Flux Isograms

(Same As Fig. 100K-11. Except along Line B (Fig. 13))

Fig. 100K-13. EBWR Core Loading No. 63, S. W. Quadrant and Adjacent Assemblies (Activations Plotted in Figs. 100K-11, 12, 16 & 17 Made along Lines A & B, Wire-Foil Also Irradiated in Position Marked "X".)

Fig. 100K-14.

Axial Neutron Flux Distribution

Fig. 100K-15.

Axial Neutron Flux Distribution

Fig. 100K-16. Neutron Flux Isograms

(S. W. Quadrant along Line A (Fig. 100K-13). Boric Acid Concentration 1110 ppm.
Control Rods Banked at 15.4 in.)

Fig. 100K-17. Neutron Flux Isograms

(Same As Fig. 100K-16 Except along Line B (Fig. 13))

Fig. 100K-18. Axial Neutron Flux Distribution (Center of Core)

Table 100K-1

INTEGRATED AREAS OF AXIAL PLOTS SHOWN IN FIG. 100K-7, AND
RELATED PLOTS OF THE SAME RUN (ISOGRAMS OF FIG. 100K-2)

Plot No. (Fig. 100K-7)	Assembly (Fig. 100K-6)	Isogram	Area (units)
1	ET-46	X	33.1
2	ET-46	X	39.5
3	ET-12	X	45.8
4	ET-12	X	40.2
5	ET-26	X	47.0
6	ET-26	X	41.0
7	EH-39	X	26.9
8	Downcomer (est.)	X	6.9
I	ET-10		49.1
West Center	ET-10		46.4
II	ET-7		41.7
East Center	ET-7		41.2
East Center	ET-8		36.9
West Center	ET-8		43.7

Table 100K-II

INTEGRATED AREAS OF AXIAL PLOTS IN FIGS. 100K-8 AND 9,
AND RELATED PLOTS OF THE SAME RUNS

Plot	Fig. No.	Assembly (Fig. 100K-6)	Isogram (Fig. No.)	Area (units)
1	100K-8	ET-46	100K-3	49.3 *
West Center		ET-46	↓	49.6 *
2	100K-8	ET-37		51.5
West Center		ET-37		53.6
3	100K-8	ET-12		58.9
4	↓	ET-12		55.3
5	↓	S-17		43.5
West Center		S-17		41.3
6	100K-8	ET-26		62.5
7	↓	ET-26		47.8
8	↓	EH-39		22.9
9	↓	H-13		16.4
10	↓	Downcomer	↓	6.9
1	100K-9	ET-46	100K-4	48.2 *
		ET-46	↓	48.8 *
2	100K-9	ET-10		47.8
3	↓	ET-10	↓	53.7
4	↓	ET-7		51.2
5		ET-7		53.6
6		S-24		45.0
7	↓	EH-35		12.9
8	↓	Downcomer	↓	8.0
Center		S-2		27.2
Center		S-9		41.3
East Center		ET-10		63.8
West Center		ET-10		55.2
East Center		ET-7		54.1
West Center		ET-7		54.0
East Center		ET-8		49.7
West Center		ET-8		57.2

* The four values for ET-46 are all nearly the same as opposed to similar values for that assembly given in Table 100K-IV.

Table 100K-III
 INTEGRATED AREAS OF AXIAL PLOTS SHOWN IN FIG. 100K-10, AND
 RELATED PLOTS OF THE SAME RUN (ISOGRAMS OF FIG. 100K-5)

<u>Plot</u> (Fig. 100K-10)	<u>Assembly</u> (Fig. 100K-6)	<u>Isogram</u>	<u>Area</u> (units)
1	ET-46	X	37.5
2	ET-46	X	43.6
3	ET-12	X	45.8
	ET-12	X	48.3
4	ET-26	X	39.7
5	ET-26	X	38.8
6	EH-39	X	32.7
I	ET-10		57.4
West Center	ET-10		45.5
II	ET-7		48.8
West Center	ET-7		47.6
III	ET-8		46.8
West Center	ET-8		43.3
IV	EH-35		17.3

Table 100K-IV
 INTEGRATED AREAS OF AXIAL PLOTS SHOWN IN FIGS. 100K-14 AND 15,
 AND RELATED PLOTS OF THE SAME RUN

<u>Plot</u>	<u>Fig. No.</u>	<u>Assembly</u> (Fig. 100K-13)	<u>Isogram</u> (Fig. No.)	<u>Area</u> (units)
1	100K-14	ET-46	100K-11	34.1*
2	↓	ET-46	↓	52.9*
3		ET-37		52.2
4		ET-12		53.1
5		ET-12		49.6
6		S-17		43.8
7		ET-26		56.2
8		EH-39		33.3
9		H-13		30.2
10	↓	Downcomer	↓	20.5
1	100K-15	ET-46	100K-12	37.1*
2	↓	ET-46	↓	49.3*
3		ET-10		57.7
4		ET-7		52.5
5		ET-7		40.2
6		S-24		31.1
7		EH-35		19.2
8	↓	Downcomer	↓	Not Determined
Center		S-16		48.0

* See Table 100K-I, note that the inner values of foils for ET-46 are much lower near the center of the core (plot No. 1, Figs. 100K-14 and 15) than they are at the outer locations in this assembly (plot No. 2, Figs. 100K-14 and 15).

NOTE

The values for this run are normalized within the run, but are not normalized with values of Tables 100K-I, II, and III.

FUTURE PLANS

The present study of cold critical neutron flux distribution in EBWR was undertaken with a minimum of time and preparation. Subsequent studies are being made of flux distribution in the operating reactor at power. In those studies an emphasis has been placed on obtaining reliable flux levels values as well as distribution and spectral information.

CONCLUSION

This study accomplished what it set out to do - to determine as quickly as possible, in a relative fashion, the distribution of neutron flux in a representative quadrant of the EBWR core during cold critical operation with several reactor control parameters. The changes in thermal neutron flux distribution with changes in reactor control parameters has been graphically presented. Comparisons of relative distributions between, for example, Figs. 100K-2 and 11, Figs. 100K-3 and 16, and Figs. 100K-4 and 17, give an indication of the effect of removal of boron shim strips; while comparisons of Figs. 100K-2, 3, and 4 give a normalized indication of the effect of varying soluble poison concentration, and consequently, rod bank positions. Axial plots likewise reflect these parameter changes. The zero and high concentrations of soluble poison do give a larger effective core size than the intermediate concentrations, due to different reasons: the former (low or zero concentration) due to relatively uninhibited scatter of neutrons without a high rate of absorption by the soluble poison; while the latter (high concentration) requires the control rods be withdrawn in order to attain the 20-kw power and therefore, obviously, produces the larger effective core size.

Removal of boron strips did significantly increase the effective core size during the cold critical studies just performed.

REFERENCES

- (1) EBWR Test Report No. 110R(A).

FLUX DISTRIBUTION MEASUREMENTS IN EBWR DURING OPERATION

(Test Report No. 100P, dated January 22, 1963, by W. G. Knapp)

INTRODUCTION

Since the equipment, procedure, and results of the program of flux distribution measurements made during power operation of the EBWR are fully described in Reference 1, the purpose of this report is to present information of a general nature concerning this program and its scope.

EQUIPMENT

The project of flux measurements during cold critical operation (Reference 1) was fairly straightforward in that (a) wires could be used for flux measurements; (b) the wires could be individually located from the top of the open reactor vessel; (c) accurate vertical location was readily possible by referencing each wire assembly from the top of the fuel plates in an assembly; and (d) removal of the wires and radiochemical analysis was a simple operation.

The present program, on the other hand, presented a more challenging problem - that of inserting, activating, removing, and analyzing foils from known locations in the reactor core during power operation. Power operation refers to 600 psi, 489°F reactor operation with power output of 20 to 80 Mw(t). Access was limited to a 3-in. instrument nozzle into the reactor vessel which was specially modified for this project.

Since it was necessary to insert and remove foils during reactor operation without affecting that operation, tubes were used which could be compared to the rabbit tube activation facility in CP-5, except that the inner diameter of these tubes was only 3.56 mm. Six 4.76-mm OD Inconel tubing "loops" were arranged, with special tools, in the reactor core and down-comer area in locations selected for flux distribution studies. The tubes were sized to withstand an external pressure of 1200 psi at up to 490°F. Where the tubes passed through the high pressure flange of the east instrument nozzle a solid 4.76-mm ID, 9.52-mm OD adapter of stainless steel was welded to the tubing. Conax fittings 9.52 mm (3/8 in.) with lava sealant were then used. No distortion of the tubing was experienced with this procedure.

Since tubing was obtained in 10.8-m lengths, it was necessary to weld two pieces together to give the necessary over-all length of 21 m. Stainless steel connectors were welded on the tubing so that the ends of the Inconel tubing were butted together in perfect alignment. The 21 m length was bent on a 3.8-cm (inside) radius in the middle, fitted with a holder at the exact position that would allow the bottom of the curvature to project 8.8 cm below the bottom of the fuel plates, and, with considerable manipulation, was in-

stalled in positions indicated in Fig. 100P-1 in the reactor core. In the case of the loop in the downcomer, this was fastened to existing hardware in the reactor.

With these tubing loops, foils could be irradiated in the reactor core along 10 axial positions, individually or simultaneously. Two procedures were possible - using a continuous chain feeding in one end of a loop and out the other, or shorter lengths merely dropped in one side, and removed from the same side. The latter was used in these studies.

Conventional flat or wire type foils could be inserted into the desired core locations only if they were very small, or could be bent or rolled to fit into the irradiation capsule which had an inside diameter of 2.44 mm. Such bending or folding could yield a condition in which self absorption due to more than one thickness of absorber would become an unpredictable variable. In order to eliminate any such variables, and to simplify foil handling by elimination of individual weighings, a spherical aluminum alloy foil with a diameter of 1.5875 (+0.0000, -0.0127) mm was used. The alloy concentration was chosen to give a sufficiently large quantity of the sensing isotope, in this case Co^{59} , while self absorption effects were kept at a negligible value; 1.01% cobalt alloyed in high purity aluminum was used in these tests.

The foils were encapsulated in sections of stainless steel tubing which had a wall thickness of 0.25 mm. These sections were incorporated into a length of bead chain in any desired spacing, usually 5 cm and 15 cm in these tests. Measured lengths of bead chain were also used to lower the test sections into a precise vertical location in the Inconel tubing loops, and hence, within the reactor core.

Cadmium covered foils were also used. Though in external appearance the encapsulated foils did not differ from the bare foils, they did incorporate a sleeve of cadmium of 0.381 mm wall thickness into which the foil was placed. Both ends of the sleeve were closed with 0.381 mm thick covers. All dimensions of the cadmium cover pieces were held to close tolerance so that during the final encapsulation step, in which the bead chain was attached to the upper end of the capsule (the bead chain having been attached at the lower end prior to any other assembly operation), the cadmium was fitted together into one integral cover completely surrounding the small spherical cobalt-aluminum foil.

A Unimat lathe was modified and enclosed in a specially designed glove box for recovering the foils from the sample chains after irradiation. The lathe was provided with a hollow draw bar spindle and a 3.1 mm precision collet obtained as accessories. The spindle was spring loaded and lever actuated to facilitate one hand operation while accurately locating a capsule to be cut open in the lathe collet. The lathe lead screw and cross feed were operated from outside the glove box. With the lathe mounted in a vertical

Fig. 100P-1. Location of Inconel Tubing "Irradiation Loops" Used for Neutron Flux Distribution Measurements in the Southeast Quadrant of the Core and the Downcomer of EBWR during Power Operation.

position the sample chain was fed down through the hollow spindle, the irradiation capsule clamped in the collet chuck, with the excess chain hanging free below. The cutting tool for opening the stainless steel capsule was ground to allow the cut to be made practically at the end face of the collet chuck. The purpose of the glove box was to contain all radioactive chips and dust, preventing contamination of the working area.

The cadmium-covered foils required additional removal operations in that the capsule had to be cut twice, once above and once below the foil location. The foil was then removed from the cadmium and stainless steel covering with pliers which were provided with a pin projection on the inside of one jaw which coincided with a hole through the other jaw. Due to the very

close tolerances, damaged foils were obtained in 2 to 5% of the sample capsules opened; no trouble was encountered with the remaining samples. Even the damaged foils could be analyzed, but they required individual weighing. No trouble was encountered with the bare foils.

Analysis of the Co^{60} radioisotope was made with a 200-channel analyzer connected to a 7.5 x 7.5 cm NaI(Tl) scintillation detection crystal. Automatic typewriter readout was used. To facilitate counting of over 2,000 foils used in this study, a sample changer was designed especially for the small spherical foils. A vacuum was used to pneumatically transfer the samples from an indexed sample changer disc to a sample holder in a precise position near the scintillation detector. At the end of the analysis, the sample was returned to its assigned location in the disc by air pressure, the disc advanced, and the procedure repeated for the next foil. With this equipment 180 runs could be made without attention to the equipment. Only minor adjustments were then necessary to initiate a second, and then a third series of 180 runs each. Each sample disc had 540 such sample locations. (A later modification allowed unattended analysis of all 540 samples.)

PROCEDURE

Sample chains were constructed as follows: nine capsules, each containing one foil, were located at 15-cm intervals, followed by an additional six capsules at 5-cm intervals, the latter being the lower end of the sample chain during irradiation. Sufficient chain, accurately measured for length, was then attached above this, followed at the uppermost end by a stop that could not enter the Inconel tubing. For a given irradiation, chains were so constructed for from one to six locations (half loop) in the reactor vessel (five in the core, one in the downcomer) and/or one location just outside of the reactor vessel. (Only one side of an irradiation loop was used at a time.)

Prior to the beginning of an irradiation, the chains were dropped into their respective tubings so that about three feet remained between the lower end of the sample chain and the upper surface of the core. Using pre-recorded verbal commands and time signals on a tape recorder, the chains were then fully inserted, using a vacuum line attached to the opposite end of each loop, in order, at one minute intervals. Still using the tape recorder "clock," the chains were removed after each had been irradiated a desired length of time - usually 30 min.

Removal consisted first of pulling (or blowing) the chains up to at least the level used prior to irradiation, and usually to the point where an increase in radiation level could be detected with a radiation monitor instrument which was located near the reactor instrument nozzle through which the tubing loops entered the reactor. A cooling period of usually 30 min was then allowed, after which the excess chain was cut from the sample chains, still in the tubing, and air pressure was used to transfer the sample chains into a lead cave, thence into indexed containers, with a minimum of radiation hazard to personnel.

After a decay period of at least one week, the sample chains were disassembled, using the modified Unimat lathe previously described, and the foils were either placed directly into an indexed location in a sample changer disc, or, as was particularly the case with cadmium covered foils, placed in labeled containers. Cadmium covers were removed with special pliers prior to placing the foils in a sample changer disc location.

The cobalt activation was determined in each case using only 100 channels of the 200-channel analyzer, and by typing out the information automatically. No attention to the equipment was required during up to 180 runs. A background run was made after each series of 15 samples representing the foils from an individual chain. Standard foils calibrated by the Experimental Physics Group, or secondary standard foils were included occasionally to give frequent standardization of the equipment.

Both Co^{60} peaks (1.17 and 1.33 Mev) were summed in the analysis, and the result and sum for a one minute (live time) counting period was multiplied by a factor giving a direct calculation of flux for that particular location, power, etc., under which conditions the foils were activated, and the result corrected for radioactivity decay since irradiation.

The results were also corrected for known position irregularities, and plotted as axial distribution plots. From several of these axial distribution plots, made with the same reactor conditions, flux distribution isograms were prepared.

Irradiations were made with bare and cadmium covered foils in five paired (but significantly different) locations in the core. A pair is dictated by the configuration of the irradiation tubing loop which must be positioned in selected fuel assemblies in the core so that one side of the tubing loop lays against one Zircaloy side plate of the assembly, while the other lays against the opposite side plate.

The reactor parameters for the irradiations are listed in Table 100P-I.

Table 100P-I

REACTOR PARAMETERS

Reactor Power, Mw	Rod Pos., in. (Nos. 1-8)	Rod Pos., in. (No. 9)	H_3BO_3^* , ppm	Foil
21	22	22	25	Co & Co/Cd
41.2	26	26	25	Co & Co/Cd
40	48	43	900	Co
42	48	44½	1200	Co & Co/Cd

Table 100P-1 (Cont'd.)

Reactor Power, Mw	Rod Pos., in. (Nos. 1-8)	Rod Pos., in. (No. 9)	H ₃ BO ₃ *, ppm	Foil
62	30½	30½	25	Co & Co/Cd
58	48	30	120	Co & Co/Cd
60	48	48	925	Co
79	48	48	25	Co
72	48	48	225	Co
75	40½	30	290	Co
80	48	48	540	Co
* Concentration measured in a cold leg. The concentration, on a volume basis, in the core, would depend on the decrease in water density of 0.8, but in the ppm would, of course, remain the same.				

COMMENTS

Some of the problems which arose with the irradiations reported here were due to limited time at higher powers and limited access to the reactor building during high power operation. Had the reactor not been scheduled for shutdown the studies using cobalt foils could have been greatly expanded to give more data. The use of detector materials other than cobalt was planned but various delays in deliveries and fabrication of foil materials held up this program.

REFERENCES

- (1) William G. Knapp, "Neutron Flux Studies in the EBWR during Power Operation," ANL-6751 (to be published).

STUDY OF OPERATIONAL DATA FROM EBWR AT 0-60-MW LEVELS
 $\Delta k / \Delta P$ INCREASING WITH POWER

(Test Report No. 100M, dated October 1, 1962, by H. P. Iskenderian)

GENERAL

For the projected operation of the EBWR to power levels of 60 to 100 Mw, after completion of Core 1A, 8 spikes were destripped by 1 strip each gaining a core reactivity increment of $\sim 1.2\%$ (Reference 1). It was estimated that the reactor would go bank critical at a rod position of 8.7 in. with zero H₃BO₃, and would meet the 8 rod shutdown requirement with an H₃BO₃ concentration of about 3 gm/gal. It was estimated that 100-Mw power could

be obtained from the reactor, if a linear extrapolation could be made of power versus reactivity data previously obtained. (Reference 1).

In order to predict the behavior of the reactor at high power levels, from considerations of reactivity, hydrodynamic losses, and reactor stability, a number of power runs at 20, 40, and 60 Mw were made with equilibrium Xe for transfer function measurements.

Control of reactor power above the 40-Mw power level was obtained with control of H_3BO_3 concentration, the control rods being all out. The above power runs amounted to ~ 1100 Mwd, resulting in a reactivity loss of 0.6%.

POWER RUN DATA

A record of power runs made is shown on Figs. 100M-1 through 3. It will be noted from the data of these figures, that there was no unaccountable losses of reactivity.

Since an accurate knowledge of the concentration of H_3BO_3 was essential, samples of the reactor borated water were tested by the Chemistry Division, in addition to the usual tests made at EBWR. A comparison of data on Figs. 100M-2 and 3 indicates mostly a good agreement between values obtained by the Chemistry Division and EBWR values obtained by the titration method.

Fig. 100M-1. EBWR Operation at 20-Mw Power Level (3/30-4/5/62)

Fig. 100M-2. EBWR Operation at 40-Mw Power Level (9/1-9/4/62)

Fig. 100M-3. EBWR Operation at 40-60-Mw Power Levels (9/5-9/10/62)

Cold criticality data of importance were the bank criticality values of 18 in. and 19.5 in. obtained before and after the above power runs amounting to ~ 110 Mwd (see Fig. 100M-4 for control rod calibration).

DISCUSSION OF MEASUREMENTS

Reactivity (or Void) Power Coefficient at 0.60 Mw

From the data of Figs. 100M-1 through 3, reactivity losses with power increases have been determined, allowing for incremental XE buildup. An evaluation of Δk versus reactor power output, or steam void, was made and plotted on Fig. 100M-5.

Fig. 100M-4. Period Measurement Data Converted Using $\beta = 6.55 \times 10^{-3}$

From the data of Fig. 100M-5 it will be noted that over the power range of 10-40 Mw,

$$\Delta k / \Delta P \simeq 0.075\% \rho \text{ per Mw}$$

and over the 40-60 Mw range,

$$\Delta k / \Delta P \simeq 0.1\% \rho \text{ per Mw.}$$

It will also be seen from Fig. 100M-5 that the reactivity tied up in voids at 60 Mw is about $5\% \rho$, the calculated value corresponding to 100 Mw (Reference 2). This reactivity loss was for a 20% average steam void in the core. The calculated loss of ρ and the average steam void concentration at 60 Mw were $\sim 3.75\%$ and 15%, respectively.

Fig. 100M-5. Reactivity Increments Required for Indicated Power Increases

There appeared, thus, a loss of reactivity, ρ , which increased with reactor output power. Assuming that these reactivity losses were all tied up in increased steam voids in the core, there was then, a "wattless" component of steam circulating in the core. This was in agreement with the explanation in Reference 3 of circulating steam carried by the "carryunder" in the down-comer area.

It appeared, then, from the point of view of steam void production, that the EBWR core 1A produced enough steam for 100 Mw power extraction. Due to thermo-mechanical limitations of the system, an appreciable part of this steam was fed back to the core.

Loss of Reactivity with Recent Core Burnup

The power operations referred to above, which were accompanied by transfer function measurements, resulted in a core burnup of ~ 1100 Mwd. The cold bank criticality data before and after 1100 Mwd operation indicated a reactivity loss of 0.6%

It was of interest to calculate if 1100 Mw burnup would result in the loss of reactivity of 0.6%. This was estimated from

$$\left. \frac{\Delta k}{k} \right|_{1100 \text{ Mw}} = \frac{\int_{\text{ET}} \phi_{\text{th}} \Sigma_a^U dV + \int_{\text{S=spike}} \phi_{\text{th}} \Sigma_a^U dV}{\int_{\text{ET}} \phi_{\text{th}} \Sigma_a dV + \int_{\text{S-spike}} \phi_{\text{th}} \Sigma_a dV}$$

The summation was carried over 28 spikes, and 56 ET elements, taking $\bar{\phi}_{\text{th}}(\text{S}) \simeq 0.61 \bar{\phi}_{\text{th}}(\text{ET})$. It was calculated $\Delta k/k \simeq 0.47\%$. This did not allow for Pu buildup in the ET elements, which could reduce the 0.47% ρ , if the effect of Pu^{240} were not high. Previous measured losses (Reference 4) were 3.6% ρ for 10^4 -Mwd burnup, or $\sim 0.4\%$ for 1100-Mwd burnup.

The agreement between measured and roughly calculated values appeared to be fair. If there is an unaccountable reactivity loss in the core, it appears to be small.

Further destripping of spikes was achieved by increasing the core reactivity by $\sim 2\%$ but maintaining the 8- and 9-rod cold shutdown requirements. The reactor is now ready for operation at powers greater than 60 Mw.

REFERENCES

- (1) EBWR Test Report No. 100J.
- (2) R. Avery, et al., "EBWR Core 1A Physics Analysis," ANL-6305 (February 1961).
- (3) EBWR Test Report No. 105A.
- (4) K. Almenas, "Heterogeneous Reactor Temperature Coefficients," ANL-6686 (to be published).

EBWR REACTOR TRANSFER FUNCTION MEASUREMENTS

(Test Report No. 123, 123A, 123B, and 123C, dated January 31, 1963, by W. C. Lipinski, A. Hirsch, C. Hsu, and G. Popper)

The data obtained from the transfer function experiments outlined here are presented in Tables 123-I through XV. All tests were performed with the feedwater and pressure control systems in manual operation.

Table 123-I

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 100, 3/13 and 14/62)

Power	2300 w	Water Level	16 ft-0 in.
Pressure	0 psig	Boric Acid	3.83 gm/gal
Rods Nos. 1-8	18.1 in.	No. 9 Rod Worth	$4.2 \times 10^{-4} \delta k/in.$
Rod No. 9	12.0 in.	Temperature	65° F

			OUT-OF-CORE							
			Analog	Digital	Null	Analog	Digital	Null	Calculated	
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ	db	ϕ
0.0115	1	1/4	+17.8	+18.4	—	-64.5	-64.0	—	+18.1	-70.0
0.0139	1	1/4	+16.9	+17.5	—	-60.9	-60.8	—	+16.8	-67.1
0.0186	2	1/4	+14.1	+14.6	—	-56.5	-57.6	—	+14.9	-62.2
0.0237	3	1/4	+12.6	+13.9	—	-52.9	-54.6	—	+13.4	-58.0
0.0329	3	1/4	+11.6	+12.4	—	-49.8	-50.0	—	+11.8	-52.4
0.0419	4	1/4	+10.4	+11.4	—	-46.2	-47.5	—	+10.7	-48.6
0.0424	4	1/2	+10.7	+11.2	—	-45.2	-45.7	—	+10.6	-48.5
0.0558	5	1/2	+10.1	+10.1	—	-44.6	-44.1	—	+ 9.5	-44.9
0.0751	7	1/2	+ 9.0	+ 9.1	—	-41.7	-41.3	—	+ 8.4	-41.9
0.100	10	1/2	+ 7.6	+ 7.6	—	-40.0	-40.1	—	+ 7.3	-39.4
0.126	13	1/2	+ 6.8	+ 6.7	—	-38.9	-38.4	—	+ 6.5	-37.3
0.178	17	1	+ 5.4	+ 5.5	+5.5	-36.3	-35.4	-35.3	+ 5.3	-33.9
0.236	24	1	+ 4.4	+ 4.4	+5.4	-32.9	-32.5	-31.3	+ 4.4	-30.9
0.323	32	1	+ 3.5	+ 3.4	+3.5	-28.9	-28.7	-28.3	+ 3.5	-27.6
0.420	41	1	+ 2.7	+ 2.8	+3.2	-26.0	-25.3	-25.3	+ 2.8	-24.7
0.558	54	1	+ 2.4	+ 2.3	+2.4	-22.5	-22.0	-22.3	+ 2.2	-21.6
0.748	72	1	+ 1.7	+ 1.7	+1.9	-20.0	-19.7	-18.3	+ 1.7	-18.5
1.02	96	1	+ 1.4	+ 1.4	+1.5	-16.9	-16.3	-16.3	+ 1.3	-15.6
1.26	126	1	+ 0.8	+ 1.2	+1.3	-15.0	-15.0	-14.3	+ 1.0	-14.0
1.76	174	1	+ 0.6	+ 0.8	+0.9	-12.4	-12.4	-12.3	+ 0.7	-11.5
2.36	228	1	+ 0.6	+ 0.6	+0.9	-11.0	-11.2	-10.3	+ 0.6	- 9.8
3.23	306	1	+ 0.5	+ 0.5	+0.5	- 9.4	-10.4	- 8.8	+ 0.4	- 8.4
4.18	403	1	+ 0.3	+ 0.4	+0.4	- 9.0	-13.5	- 8.3	+ 0.3	- 7.6
5.62	536	1	+ 0.3	+ 0.2	+0.3	- 7.3	- 9.4	- 7.8	+ 0.2	- 7.2
5.57*	536	1	+ 0.3	+ 0.2	+0.3	- 7.5	- 9.0	- 7.8	+ 0.2	- 7.2
7.31	720	1	+ 0.2	+ 0.1	+0.2	- 8.0	- 9.1	- 7.3	+ 0.2	- 7.2
9.88	707	1	+ 0.1	+ 0.0	+0.1	- 8.6	- 9.0	- 7.8	+ 0.1	- 7.7
12.7	929	1	+ 0.3	+ 0.1	+0.1	-10.5	-10.0	- 8.8	+ 0.0	- 8.6
17.8	1281	1	- 0.3	- 0.0	-0.1	-14.5	-11.6	-11.3	- 0.1	-10.3
23.4	1681	1	- 0.4	- 0.4	-0.1	-15.2	-14.5	-12.8	- 0.2	-12.4
32.1	1444	1	- 0.4	- 0.4	-0.3	-17.0	-18.0	-14.8	- 0.3	-15.7
41.4	2160	3/4	- 0.2	- 0.4	+0.2	-19.9	-19.0	-18.3	- 0.5	-19.3
55.8	2873	3/4	-	- 0.5	-0.4	-	-21.9	-24.3	- 0.9	-24.7

* Counterclockwise

Table 123-II

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 200, 3/16/62)

Power	11.5 Mw	Water Level	16 ft-0 in.
Pressure	590 psig	Boric Acid	5.0 gm/gal
Rods Nos. 1-8	29.1 in.	No. 9 Rod Worth	$5.5 \times 10^{-4} \delta k/\text{in.}$
Rod No. 9	23.0 in.		

			OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ
0.00757	1	3/4	+ 2.3	+ 2.1	-	+10.8	+10.6	-
0.0100	1	3/4	+ 1.8	+ 1.7	-	+ 5.7	+ 5.9	-
0.0131	1	3/4	+ 1.6	+ 1.4	-	+ 1.8	+ 1.7	-
0.0189	2	3/4	+ 1.2	+ 0.9	-	- 1.0	- 1.1	-
0.0233	3	3/4	+ 1.1	+ 0.8	-	- 1.0	- 0.8	-
0.0318	3	3/4	+ 0.9	+ 0.6	-	- 0.8	- 0.7	-
0.0413	4	3/4	+ 0.8	+ 0.5	-	+ 0.0	+ 0.0	-
0.0556	5	3/4	+ 0.7	+ 0.3	-	+ 0.5	+ 0.7	-
0.0739	7	3/4	+ 1.0	+ 0.7	-	+ 1.8	+ 2.6	-
0.0987	10	3/4	+ 1.3	+ 1.1	-	+ 1.4	+ 1.9	-
0.133	13	3/4	+ 1.9	+ 1.7	-	- 1.6	- 1.5	-
0.182	18	3/4	+ 2.2	+ 2.2	-	- 6.1	- 5.4	-
0.233	24	3/4	+ 2.2	+ 1.9	-	-10.8	-10.2	-
0.322	32	3/4	+ 1.8	+ 1.9	+2.4	-18.5	-20.3	-19.3
0.419	41	3/4	- 0.2	+ 0.5	-0.0	-29.2	-30.1	-24.3
0.557	54	3/4	- 2.8	- 2.7	-	-28.0	-28.1	-
0.740	72	3/4	- 5.4	- 4.6	-	- 7.3	- 6.4	-
1.00	96	3/4	- 3.6	- 2.7	-	+12.2	+12.5	-
1.31	126	3/4	- 2.2	- 2.0	-2.7	+15.2	+14.3	+12.8
1.82	174	3/4	- 0.8	- 0.7	-0.6	+15.5	+13.9	+ 5.8
2.36	228	3/4	- 0.1	+ 0.1	+0.8	+12.5	+11.2	+ 5.8
2.36	78	3/4	+ 0.3	- 1.3	+0.8	+11.9	+10.2	+ 6.3
3.16	306	3/4	+ 0.4	+ 0.7	+0.8	+ 9.0	+ 8.7	+ 4.8
4.13	202	3/4	+ 1.1	+ 1.3	+1.1	+ 4.0	+ 3.0	+ 3.8
5.55	268	3/4	+ 0.8	+ 1.0	+0.8	- 1.9	- 1.3	- 2.3
7.53	360	3/4	+ 0.5	+ 0.2	+0.7	- 5.6	- 6.4	- 6.3
9.88	480	3/4	+ 0.2	- 0.1	+0.3	- 7.0	- 6.1	- 6.8
13.2	630	3/4	+ 0.0	- 0.3	+0.1	- 8.0	- 5.1	- 7.3
18.3	870	3/4	- 0.2	- 0.3	-0.0	-11.3	-11.3	- 9.8
23.9	1141	3/4	- 0.4	- 0.5	-0.2	-10.4	-11.7	-10.3
32.1	1531	3/4	- 0.7	- 0.6	-0.2	-14.6	-15.1	-13.3
42.1	1450	3/4	- 0.5	-	-0.3	-15.6	-	-16.3
55.7	1929	3/4	- 0.6	- 0.4	-1.2	-20.3	-21.5	-21.3

Table 123-III

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 300, 3/19 and 20/62)

Power	20 Mw	Water Level	16 ft-0 in.
Pressure	600 psig	Boric Acid	4.0 gm/gal
Rods Nos. 1-8	32.2 in.	No. 9 Rod Worth	$5.9 \times 10^{-4} \delta k/\text{in.}$
Rod No. 9	23.0 in.		

			OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ
0.00765	1	1	- 1.3	- 1.4	-	+20.5	+20.7	-
0.00991	1	1	- 1.7	- 1.9	-	+12.4	+12.7	-
0.0133	1	1	- 2.1	- 2.3	-	+ 6.6	+ 6.9	-
0.0182	2	1	- 2.5	- 2.8	-	+ 4.5	+ 4.5	-
0.0235	3	1	- 3.1	- 3.1	-	+ 2.9	+ 2.4	-
0.0318	3	1	- 3.1	- 3.2	-	+ 3.5	+ 3.5	-
0.0419	4	1	- 3.1	- 3.3	-	+ 3.7	+ 3.0	-
0.0557	5	1	- 3.0	- 3.2	-	+ 5.9	+ 4.9	-
0.0756	7	1	- 2.6	- 2.9	-	+ 7.5	+ 7.5	-
0.0999	10	1	- 2.3	- 2.5	-	+ 8.1	+ 7.9	-
0.132	13	1	- 1.7	- 1.8	-	+ 8.6	+ 8.1	-
0.185	17	1	- 0.9	- 1.1	-	+ 3.6	+ 3.4	-
0.236*	24	1	- 0.5	- 0.7	-	- 1.6	- 1.4	-
0.239	24	1	- 0.4	- 0.5	-	- 1.0	- 1.1	-
0.322	32	1	- 0.3	- 0.4	-	-11.2	-11.9	-
0.421	41	1	- 1.0	- 1.1	-0.5	-24.0	-23.7	-29.3
0.549	54	1	- 3.4	- 3.6	-2.7	-33.7	-33.9	-34.3
0.748	72	1	- 8.1	- 8.2	-5.3	-22.2	-23.0	- 9.3
0.992	96	1	- 8.5	- 8.8	-6.5	+ 7.4	+ 6.1	+10.8
1.32	126	1	- 5.8	- 6.0	-4.8	+22.9	+22.6	+20.8
1.81	174	1	- 4.5	- 4.7	-3.4	+30.0	+30.1	+31.8
2.40	228	1	- 2.2	- 2.4	-3.4	+31.7	+31.3	+31.8
2.39	228	3/4	- 2.0	-	-3.4	+30.1	-	+33.8
3.17	306	3/4	- 0.4	- 0.5	-0.0	+29.8	+30.3	+22.8
4.20	202	3/4	+ 1.2	+ 1.2	+1.8	+23.3	+24.5	+22.3
5.59	268	3/4	+ 2.3	+ 2.3	+2.6	+10.3	+12.0	+10.8
7.49	360	3/4	+ 2.0	+ 1.8	+2.0	- 1.9	+ 0.1	- 1.3
9.96	480	3/4	+ 1.1	+ 0.9	+1.3	- 7.2	- 3.7	- 5.8
13.3	630	3/4	+ 0.7	+ 0.4	+0.6	- 8.4	- 6.0	- 7.3
18.4	870	3/4	- 0.3	- 0.3	-0.1	-11.3	- 8.4	- 9.8
23.5	1141	3/4	- 0.6	- 0.4	-0.2	-10.6	- 9.2	-10.3
31.9	1531	3/4	- 0.5	- 0.4	-0.3	-13.2	-13.5	-13.3
41.8	1450	3/4	- 0.6	-	-0.3	-15.9	-	-16.3
55.6	1929	3/4	- 0.7	- 0.6	-0.5	-20.3	-21.0	-20.3

* Counterclockwise

Table 123-IV

TRANSFER FUNCTION EXPERIMENTAL DATA

(Exp. No. 400, 3/21 and 22/62)

Power	20 Mw	Water Level	16 ft-0 in.
Pressure	300 psig	Boric Acid	4.0 gm/gal
Rods Nos. 1-8	32.2 in.	No. 9 Rod Worth	$5.7 \times 10^{-4} \delta k/in.$
Rod No. 9	23.0 in.		

ω	Cycles	Stroke, in.	OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
			db	db	db	ϕ	ϕ	ϕ
0.00752	1	1	- 2.1	- 2.5	-	+ 8.2	+ 7.3	-
0.00987	1	1	- 2.5	- 2.8	-	+ 5.5	+ 5.7	-
0.0130	1	1	- 2.8	- 3.1	-	+ 2.2	+ 2.9	-
0.0185	2	1	- 3.3	- 3.6	-	+ 1.4	+ 1.2	-
0.0235	3	1	- 3.5	- 3.9	-	+ 0.6	+ 0.9	-
0.0316	3	1	- 3.9	- 4.2	-	+ 2.3	+ 2.4	-
0.0415	4	1	- 3.8	- 4.2	-	+ 4.7	+ 5.0	-
0.0557	5	1	- 3.6	- 4.0	-	+ 7.7	+ 7.6	-
0.0759	7	1	- 3.0	- 3.3	-	+ 8.3	+ 9.4	-
0.100	10	1	- 2.3	- 2.7	-	+10.5	+11.0	-
0.131	13	1	- 1.5	- 1.8	-	+ 11.4	+ 11.5	-
0.183	18	1	- 0.6	- 0.9	-	+ 5.3	+ 5.6	-
0.233	24	1	- 0.1	- 0.4	-	- 1.5	- 2.1	-
0.239*	24	1	- 0.2	- 0.5	-	- 3.1	- 2.0	-
0.321	32	1	- 0.5	- 0.8	-	-12.8	-14.2	-
0.416	41	1-1/2	- 1.3	- 1.5	-	-24.0	-24.7	-
0.552	54	1-1/2	- 3.6	- 3.8	-	-32.3	-32.7	-
0.767	72	1-1/2	- 7.6	- 7.7	-	-26.7	-28.3	-
1.01	96	1-1/2	-10.3	-10.5	-	- 2.4	+ 0.9	-
1.31	126	1-1/2	- 7.8	- 8.1	-	+20.6	+19.8	-
1.85	174	1-1/2	- 5.2	- 5.7	-	+30.2	+30.1	-
2.33	228	1-1/2	- 4.3	- 4.5	-4.6	+36.1	+33.5	+43.8
3.19	306	1-1/2	- 1.1	- 1.4	-1.3	+36.5	+37.0	+30.8
3.18	306	1/2	- 1.0	- 1.5	-	+36.9	+34.5	-
4.12	202	1/2	+ 0.6	+ 0.1	-	+29.1	+28.1	-
5.56	268	1/2	+ 2.1	+ 1.7	-	+18.2	+18.4	-
7.57	360	1/2	+ 2.5	+ 2.1	+1.5	+ 2.5	+ 2.0	- 1.3
9.98	480	1/2	+ 1.5	+ 1.1	+0.8	- 6.2	- 3.7	- 7.3
13.2	630	1/2	+ 0.6	+ 0.2	-0.0	- 8.7	- 6.1	-11.3
18.3	870	1/2	+ 0.0	- 0.3	-0.2	-11.6	-11.8	-10.3
23.9	1141	1/2	- 0.1	- 0.4	-0.2	-10.6	-12.2	-10.8
32.0	1531	1/2	- 0.3	- 0.5	-0.4	-13.3	-14.7	-13.8
41.9	1450	1/2	- 0.7	- 0.6	-0.5	-14.5	-16.7	-15.8
56.2	1929	1/2	- 0.8	- 0.7	-0.4	-19.5	-20.0	-20.3

* Counterclockwise

Table 123-V

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 500, 3/23/62)

Power	10 Mw	Water Level	16 ft-0 in.
Pressure	300 psig	Boric Acid	4.0 gm/gal
Rods Nos. 1-8	27.3 in.	No. 9 Rod Worth	4.8×10^{-4} δ k/in.
Rod No. 9	23.0 in.		

			OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ
0.00745	1	3/4	+ 0.0	- 0.1	-	+ 4.8	+ 5.3	-
0.0101	1	3/4	- 0.3	- 0.4	-	+ 1.5	+ 2.7	-
0.0132	1	3/4	- 0.5	- 0.7	-	+ 0.1	+ 0.7	-
0.0182	2	3/4	- 0.6	- 0.8	-	- 1.5	- 0.9	-
0.0239	3	3/4	- 0.9	- 1.1	-	- 1.5	- 0.9	-
0.0317	3	3/4	- 1.2	- 1.3	-	- 0.2	+ 0.7	-
0.0416	4	3/4	- 1.2	- 1.3	-	+ 1.8	+ 2.5	-
0.0555	5	3/4	- 1.0	- 1.1	-	+ 4.0	+ 4.8	-
0.0764	7	3/4	- 0.5	- 0.6	-	+ 4.8	+ 5.7	-
0.101	7	3/4	+ 0.0	- 0.1	-	+ 5.2	+ 5.9	-
0.101	10	3/4	+ 0.1	+ 0.0	-	+ 5.3	+ 6.2	-
0.131	13	3/4	+ 0.6	+ 0.6	-	+ 3.5	+ 4.3	-
0.186*	18	3/4	+ 1.3	+ 1.3	-	- 2.5	- 2.0	-
0.185	18	3/4	+ 1.4	+ 1.1	-	- 2.8	- 3.0	-
0.238	24	3/4	+ 1.3	+ 1.2	-	- 8.6	- 8.4	-
0.322	32	3/4	+ 0.6	+ 0.5	-	-18.4	-18.2	-
0.415	41	1-1/2	- 0.7	- 0.7	-0.2	-23.8	-24.5	-19.3
0.558	54	1-1/2	- 2.9	- 3.0	-2.8	-26.3	-26.9	-29.3
0.736	72	1-1/2	- 5.4	- 5.5	-6.1	-19.1	-20.3	-21.3
1.00	96	1-1/2	- 6.5	- 6.6	-6.3	+ 5.1	+ 4.2	+15.8
1.32	126	1-1/2	- 4.4	- 4.5	-6.3	+20.7	+19.7	+23.8
1.84	174	3/4	- 2.4	- 2.6	-1.3	+25.6	+25.0	+28.8
2.37	228	3/4	- 0.5	- 0.7	-0.1	+22.4	+22.9	+22.3
3.19	306	3/4	+ 0.4	+ 0.3	+0.4	+16.9	+16.7	+19.8
4.16	202	3/4	+ 1.1	+ 1.0	+0.8	+12.3	+11.5	+11.8
5.57	268	3/4	+ 1.5	+ 1.4	+1.3	+ 3.9	+ 4.1	+ 5.8
7.57	360	3/4	+ 1.1	+ 1.0	+1.0	- 3.0	- 4.2	- 0.8
9.99	480	3/4	+ 0.6	+ 0.4	+0.4	- 6.3	- 4.2	- 3.8
13.3	630	3/4	+ 0.3	+ 0.1	+0.2	- 7.4	- 7.0	- 5.8
18.2	870	3/4	- 0.1	- 0.1	-0.1	-11.5	-11.3	- 9.8
23.8	1141	3/4	- 0.5	- 0.3	-0.2	-10.7	-12.6	- 9.8
31.9	1531	3/4	- 0.6	- 0.5	-0.1	-13.1	-14.6	-11.3
41.1	1450	3/4	- 0.4	- 0.5	+0.0	-15.7	-17.0	-14.3
56.0	1929	3/4	- 0.8	- 0.8	-0.1	-20.8	-22.0	-21.8

* Counterclockwise

Table 123-VI

TRANSFER FUNCTION EXPERIMENTAL DATA

(Exp. No. 600, 3/27 and 28/62)

Power	10 Mw	Water Level	16 ft-0 in.
Pressure	600 psig	Boric Acid	0.2 gm/gal
Rods Nos. 1-8	18.7 in.	No. 9 Rod Worth	$8.7 \times 10^{-4} \delta k/in.$
Rod No. 9	14.5 in.		

			OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ
0.00746	1	1/2	- 1.6	- 1.5	-	+16.2	+16.8	-
0.0100	1	1/2	- 1.4	- 1.2	-	+10.6	+11.0	-
0.0132	1	1/2	- 1.2	- 1.1	-	+ 8.1	+ 8.5	-
0.0186	2	1/2	- 1.3	- 1.2	-	+ 5.0	+ 5.6	-
0.0238	3	1/2	- 1.3	- 1.2	-	+ 3.3	+ 3.4	-
0.0320	3	1/2	- 1.3	- 1.1	-	+ 0.0	+ 0.1	-
0.0415	4	1/2	- 1.5	- 1.4	-	+ 0.4	+ 0.8	-
0.0561	5	1/2	- 1.2	- 1.1	-	- 0.6	- 0.2	-
0.0751	7	1/2	- 1.2	- 1.0	-	+ 0.5	+ 0.6	-
0.0996	10	1/2	- 0.9	- 0.7	-	- 1.5	- 1.2	-
0.133	13	1/2	- 0.9	- 0.7	-	- 4.4	- 3.9	-
0.185	18	1/2	- 0.5	- 0.2	-	-12.4	-12.2	-
0.243	24	1/2	- 1.1	- 1.3	-0.5	-20.0	-19.4	-21.3
0.317	32	1	- 2.7	- 2.9	-2.6	-26.0	-27.2	-29.3
0.417	41	1	- 5.1	- 5.4	-4.2	-25.3	-26.0	-21.3
0.555	54	1	- 6.3	- 6.7	-6.1	-12.5	-13.0	-11.3
0.740	72	1	- 6.9	- 7.3	-6.2	- 3.2	- 2.9	- 3.3
1.01	96	1	- 6.3	- 6.7	-6.3	+ 9.9	+10.0	+ 7.8
1.31	126	1	- 5.8	- 6.3	-5.7	+18.2	+18.5	+18.8
1.83	174	1	- 3.5	- 4.0	-3.3	+25.7	+26.1	+24.3
2.40	228	1	- 2.4	- 2.8	-2.1	+23.7	+23.9	+22.8
2.35	228	1/2	- 2.4	- 2.8	-1.1	+24.0	+24.5	+24.8
3.19	306	1/2	- 1.0	- 1.1	-1.0	+24.6	+25.7	+23.8
4.13	202	1/2	+ 0.5	+ 0.3	+0.2	+20.5	+20.3	+21.8
5.57	268	1/2	+ 1.4	+ 1.3	+1.2	+10.3	+11.3	+11.8
7.57	350	1/2	+ 1.3	+ 1.2	+1.6	- 0.3	+ 0.5	- 0.3
10.1	480	1/2	+ 0.7	+ 0.8	+0.9	- 6.2	- 5.8	- 4.8
13.2	630	1/2	+ 0.2	+ 0.1	+0.2	- 9.0	- 7.0	- 7.5
18.3	870	1/2	- 0.3	- 0.2	- 0.1	-11.6	-11.5	- 9.8
23.5	628	1/2	- 0.5	- 0.1	- 0.3	-11.8	-13.3	-10.3
23.5	1141	1/2	- 0.6	-	-0.4	-12.1	-	-10.3
31.6	1531	1/2	-	- 0.6	-	-	-15.3	-
32.2	1531	1/2	- 0.6	- 0.5	-0.4	-14.9	-15.1	-13.3
41.8	1450	1/2	- 0.6	- 0.6	-0.5	-18.9	-18.7	-16.5
55.4	1929	1/2	- 0.6	- 0.6	-0.5	-23.6	-23.8	-21.3

Table 123-VII

TRANSFER FUNCTION EXPERIMENTAL DATA

(Exp. No. 700, 3/29 and 30/62)

Power	10 Mw	Water Level	16 ft-0 in.
Pressure	300 psig	Boric Acid	0.1 gm/gal
Rods Nos. 1-8	17.8 in.	No. 9 Rod Worth	8.0×10^{-4} δ k/in.
Rod No. 9	14.5 in.		

			OUT-OF-CORE					
			Analog	Digital	Null	Analog	Digital	Null
ω	Cycles	Stroke, in.	db	db	db	ϕ	ϕ	ϕ
0.00763	1	1/2	- 3.0	- 3.2	-	+ 8.6	+ 8.6	-
0.0100	1	1/2	- 3.2	- 3.4	-	+ 6.1	+ 6.1	-
0.0130	1	1/2	- 3.0	- 3.2	-	+ 4.5	+ 4.5	-
0.0183	2	1/2	- 2.9	- 3.0	-	+ 2.5	+ 2.9	-
0.0236	3	1/2	- 3.1	- 3.4	-	+ 1.4	+ 1.8	-
0.0317	3	1/2	- 3.2	- 3.4	-	+ 0.4	+ 1.2	-
0.0413	4	1/2	- 3.2	- 3.5	-	+ 2.9	+ 3.3	-
0.0555	6	1/2	- 3.0	- 3.2	-	+ 2.1	+ 2.7	-
0.0748	7	1/2	- 2.6	- 2.9	-	+ 4.0	+ 5.0	-
0.0971	10	1/2	- 2.2	- 2.0	-	+ 3.6	+11.3	-
0.100	10	1/2	- 1.7	- 1.8	-	+ 2.1	+ 2.4	-
0.131	13	1/2	- 1.7	- 1.9	-	- 0.6	- 0.7	-
0.185	18	1/2	- 1.2	- 1.5	-	-10.3	-10.3	-
0.239	24	1/2	- 1.3	- 1.6	-	-15.2	-14.4	-
0.312	32	1	- 2.5	- 2.8	-3.2	-25.4	-25.6	-24.3
0.414	41	1	- 4.7	- 4.8	-4.9	-30.4	-29.8	-34.3
0.561	54	1	- 7.5	- 7.5	-7.5	-19.8	-20.5	-20.3
0.755	72	1	- 8.1	- 8.2	-	- 5.8	- 6.0	-
1.01	96	1	- 8.1	- 8.1	-	+ 7.9	+ 7.5	-
1.34	126	1	- 7.3	- 7.5	-	+20.9	+21.0	-
1.85	174	1	- 5.5	- 5.7	-5.1	+31.8	+31.1	+25.8
2.45	228	1	- 3.2	- 3.0	-3.6	+32.9	+33.1	+33.8
2.39	228	1/2	- 3.4	- 3.2	-3.2	+33.5	+33.3	+32.8
3.21	306	1/2	- 1.8	- 2.0	-1.7	+30.7	+29.9	+28.8
4.18	202	1/2	- 0.2	- 0.4	-0.7	+28.5	+28.2	+27.8
5.56	268	1/2	+ 1.3	+ 1.3	+1.2	+18.0	+18.8	+17.8
7.60	360	1/2	+ 1.9	+ 1.7	+1.8	+ 4.9	+ 5.3	+ 7.8
9.93	480	1/2	+ 1.4	+ 1.3	+1.2	- 4.3	- 2.4	- 1.3
13.2	630	1/2	+ 0.7	+ 0.5	+0.6	- 9.6	- 5.9	- 7.3
18.4	870	1/2	- 0.0	- 0.1	-0.1	-13.5	-12.4	-10.3
23.9	1141	1/2	- 0.5	- 0.4	-0.2	-13.6	-13.1	-11.3
32.1	1531	1/2	- 0.6	- 0.6	-0.6	-15.5	-15.5	-13.8
41.8	1450	1/2	- 0.6	- 0.6	-0.6	-19.3	-18.8	-16.3
55.7	1929	1/2	- 0.6	- 0.7	-0.7	-24.1	-23.9	-21.3

Table 123-VIII

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 800, 4/2 and 3/62)

Power	20 Mw	Water Level	16 ft-0 in.
Pressure	600 psig	Boric Acid	0.08 gm/gal
Rods Nos. 1-8	22.4 in.	No. 9 Rod Worth	$9.0 \times 10^{-4} \delta k/in.$
Rod No. 9	14.5 in.		

ω	Cycles	Stroke, in.	OUT-OF-CORE			
			Analog	Digital	Analog	Digital
			db	db	ϕ	ϕ
0.00746	1	1/2	- 1.1	- 1.3	+34.2	+34.6
0.0100	1	1/2	- 2.1	- 2.4	+18.1	+17.5
0.0131	1	1/2	- 2.0	- 2.3	+ 8.7	+ 9.3
0.0181	1	1/2	- 3.4	- 3.6	+ 3.9	+ 5.1
0.0236	1	1/2	- 2.6	- 2.9	- 7.5	- 6.9
0.0318	1	1/2	- 2.9	- 3.3	+ 6.5	+ 6.8
0.0416	2	1/2	- 3.5	- 3.9	+ 7.4	+ 7.6
0.0551	3	1/2	- 2.7	- 3.0	+ 6.1	+ 6.4
0.0752	4	1/2	- 3.0	- 3.3	+ 6.3	+ 7.0
0.101	6	1/2	- 2.7	- 3.1	+ 0.2	+ 1.8
0.101	24	1/2	- 2.4	-	+ 6.4	-
0.132	7	1/2	- 1.8	- 2.1	- 2.5	- 1.9
0.183	10	1/2	- 0.4	- 0.7	-10.4	-10.6
0.183	10	1/2	- 0.2	- 0.6	-10.4	- 9.3
0.233	13	1/2	- 1.8	-	-16.9	-
0.241	13	1/2	- 1.0	- 1.0	-26.5	-26.4
0.241	13	1/2	- 1.5	- 1.6	-24.1	-23.5
0.241	13	1/2	- 2.2	- 2.3	-16.9	-18.6
0.315	16	1/2	- 3.6	- 3.4	-35.9	-35.6
0.414	21	1/2	- 7.1	- 7.1	-40.1	-40.3
0.412	21	1	- 5.6	- 5.9	-40.2	-40.9
0.550	37	1	- 9.1	- 9.5	-25.5	-25.4
0.752	36	1	-10.0	-10.4	-12.8	-12.1
0.981	48	1	-10.7	-11.2	+10.8	+11.6
1.31	63	1	- 7.4	- 7.6	+17.6	+18.2
1.84	87	1	- 7.6	- 7.9	+29.3	+30.4
2.36	114	1	- 5.0	- 5.1	+35.2	+36.4
2.36	76	1	- 5.1	- 5.4	+34.4	+34.6
2.41	38	1	- 4.9	- 5.1	+34.7	+35.6
2.38	38	1/2	- 4.9	- 4.2	+39.0	+48.0
3.19	51	1/2	- 3.5	- 3.8	+35.6	+36.8
4.14	67	1/2	- 1.7	- 1.7	+36.1	+36.4
5.53	89	1/2	+ 0.7	+ 0.7	+28.1	+29.2
7.50	180	1/2	+ 2.1	-	+13.3	-
7.54	360	1/2	+ 2.1	-	+13.6	-
7.59	120	1/2	+ 2.0	+ 1.9	+12.6	+12.7
10.1	160	1/2	+ 1.9	+ 1.7	-11.2	- 0.5
13.2	210	1/2	+ 0.9	+ 0.7	- 8.0	- 5.2
18.4	290	1/2	- 0.2	- 0.1	-12.5	-11.9
23.6	380	1/2	- 0.5	- 0.4	-12.1	-13.0
31.9	510	1/2	- 0.5	- 0.5	-14.7	-15.5
41.2	670	1/2	- 0.6	- 0.5	-17.9	-19.5
55.4	890	1/2	- 0.7	- 0.7	-22.9	-23.3

Table 123-IX

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 900, 4/4 and 5/62)

Power	20 Mw	Water Level	16 ft-0 in.
Pressure	300 psig	Boric Acid	0.15 gm/gal
Rods Nos. 1-8	22.7 in.	No. 9 Rod Worth	8.9×10^{-4} $\delta k/in.$
Rod No. 9	14.5 in.		

			OUT-OF-CORE			
			Analog	Digital	Analog	Digital
(ω)	Cycles	Stroke, in.	db	db	ϕ	ϕ
0.00753	1	1	- 2.2	-	+18.1	-
0.0100	1	1	- 3.0	-	+ 9.0	-
0.0132	1	1	- 3.4	-	+ 3.0	-
0.0184	2	1	- 4.3	-	+ 3.8	-
0.0236	3	1	- 4.1	-	+ 2.5	-
0.0320	3	1	- 4.6	-	+ 8.5	-
0.0415	4	1	- 4.3	-	+ 8.9	-
0.0556	5	1	- 3.9	-	+ 3.8	-
0.0746	7	1	- 3.1	-	+10.3	-
0.0987	10	1	- 2.6	-	+11.4	-
0.130	14	1	- 1.4	- 2.0	+ 2.6	+ 2.9
0.172	18	1	- 0.4	- 0.8	- 8.8	+13.3
0.237	24	1	- 1.2	- 0.9	-25.3	+23.9
0.321	32	1	- 3.8	- 4.1	-38.7	-38.8
0.416	41	2	- 6.9	- 7.0	-40.0	-43.2
0.559	54	2	- 9.3	- 9.8	-32.0	-31.9
0.747	72	2	-10.5	-10.8	-18.9	-19.2
1.00	96	2	-13.0	-13.3	+ 2.8	+ 3.9
1.01	96	2	-12.9	-	+ 5.5	-
1.01	96	2	-13.0	-13.1	+ 0.6	0
1.31	126	2	-10.5	-10.7	+24.3	+23.9
1.83	174	2	- 8.5	- 8.7	+27.9	+28.0
2.35	228	2	- 8.0	- 8.2	+37.8	+37.7
3.25	306	1	- 4.7	- 4.9	+43.6	+43.5
4.17	202	1	- 2.8	- 3.0	+46.0	+45.6
4.16	202	1/2	- 3.0	- 3.2	+45.9	+45.1
5.59	268	1/2	- 0.0	- 0.1	+41.1	+41.0
7.56	360	1/2	+ 2.5	+ 2.3	+26.2	+26.3
10.0	480	1/2	+ 3.0	-	+ 5.3	-
9.99	320	1/2	+ 3.0	+ 2.8	+ 5.7	+ 8.6
13.2	630	-	+ 1.9	+ 1.6	- 7.8	- 3.8
18.3	870	-	+ 0.3	+ 0.3	-14.2	-13.6
23.5	1141	1/2	- 0.3	- 0.3	-13.6	-14.0
31.7	1531	1/2	- 0.5	- 0.5	-15.7	-15.9
41.6	1450	1/2	- 0.6	- 0.6	-18.3	-18.5
55.9	890	1/2	- 0.7	- 0.7	-23.2	-23.1

Table 123-X

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 1000, 7/5 and 6/62)

Power	40 Mw	Water Level	16 ft-0 in.
Pressure	595 psig	Boric Acid	0.15 gm/gal
Rods Nos. 1-8	28.5 in.	No. 9 Rod Worth	6.4×10^{-4} δ k/in.
Rod No. 9	21.5 in.		

			OUT-OF-CORE			
			Analog	Digital	Analog	Digital
ω	Cycles	Stroke, in.	db	db	ϕ	ϕ
0.0102	1	1	- 5.5	- 6.9	+26.4	+28.6
0.0134	1	1	- 5.1	- 6.9	+13.1	+15.1
0.0186	2	1	- 4.8	- 7.5	+15.8	+11.4
0.0244	3	1	- 4.7	- 6.6	+12.6	+12.9
0.0327	3	1	- 4.0	- 4.2	+16.4	+16.1
0.0422	4	1	- 2.1	- 2.8	+ 8.2	+ 8.1
0.0560	5	1	- 3.3	- 4.3	+ 3.7	+ 7.7
0.0765	7	1	- 2.4	- 3.3	+ 2.4	+ 4.0
0.0766	7	1	- 2.1	-	+ 3.4	-
0.101	10	1	- 3.6	- 4.6	- 6.2	- 5.4
0.133	13	1	- 2.8	- 3.7	- 5.1	- 5.5
0.184	18	1	- 4.1	- 4.9	-26.4	-26.3
0.238	24	2	- 5.2	- 6.0	-25.7	-26.2
0.322	32	2	- 7.9	- 8.5	-17.4	-16.3
0.419	41	2	- 7.7	- 8.4	-17.7	-17.7
0.556	54	2	- 8.6	- 9.4	-21.5	-21.5
0.764	72	2	-11.8	-12.6	- 9.6	- 8.7
1.02	96	2	-11.1	-11.8	+ 9.2	+10.2
1.33	126	2	-10.0	-10.8	+13.4	+14.5
1.85	174	2	-10.6	-11.2	+26.3	+27.2
2.41	228	2	- 9.0	- 9.6	+36.2	+36.8
3.25	218	2	- 6.8	- 7.3	+48.3	+48.7
4.20	202	1	- 4.4	- 5.0	+52.8	+53.1
5.68	268	1	- 1.2	- 1.8	+52.1	+52.2
7.46	360	1/2	+ 2.3	+ 1.7	+41.8	+42.9
9.26	414	1/2	+ 4.3	+ 3.3	+24.7	+23.2
10.2	480	1/2	+ 4.2	+ 3.5	+12.6	+13.9
11.8	555	1/2	+ 4.1	+ 2.8	+ 1.2	- 4.9
13.3	630	1/2	+ 3.1	+ 2.3	- 6.1	- 4.9
18.6	870	1/2	+ 1.2	+ 0.5	-13.9	-12.6
24.1	1141	1/2	+ 0.6	-	-14.2	-
24.4	1141	1/2	+ 0.9	-	-14.5	-
24.4*	1141	1/2	+ 0.8	- 0.1	-12.8	-13.6
32.2	1531	1/2	+ 0.2	- 0.5	-17.7	-16.1
42.0	1450	1/2	- 0.6	- 0.6	-20.1	-18.7
55.3	1929	1/2	- 0.9	- 0.6	-23.6	-22.9

* Counterclockwise

Table 123-XI

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 1100, 9/5 and 6/62)

Power
Pressure
Rods Nos. 1-8

40 Mw
592 psig
48.0 in.

Rod No. 9
Water Level
Boric Acid
No. 9 Rod Worth

30.0 in.
16 ft-0 in.
2.73 gm/gal
 5.7×10^{-4} δ k/in.

ω	Cyc- les	Stroke, in.	OUT-OF-CORE						IN-CORE EPITHERMAL				IN-CORE THERMAL			
			Ana- log	Digi- tal	Null	Ana- log	Digi- tal	Null	Ana- log	Digi- tal	Ana- log	Digi- tal	Ana- log	Digi- tal	Ana- log	Digi- tal
			db	db	db	ϕ	ϕ	ϕ	db	db	ϕ	ϕ	db	db	ϕ	ϕ
0.0132	1	1-1/2	- 5.0	- 5.3	-	+28.1	+28.6	-	- 7.0	- 7.8	+18.5	+16.1	- 6.5	- 7.3	+31.1	+31.9
0.0183	2	1-1/2	- 4.5	- 4.7	-	+29.2	+30.2	-	- 5.6	- 5.9	+38.7	+38.5	- 5.9	- 5.7	+32.5	+33.6
0.0233	3	1-1/2	- 4.2	- 4.4	-	+25.3	+26.2	-	- 3.7	- 3.6	+30.2	+30.2	- 5.4	- 4.9	+20.0	+28.2
0.0318	3	1-1/2	- 3.3	- 3.5	-	+26.0	+26.8	-	- 3.9	- 3.7	+30.3	+29.4	- 3.9	- 3.6	+28.1	+28.6
0.0416	4	1-1/2	- 1.9	-	-	+20.6	-	-	- 0.9	-	+25.4	-	- 2.3	-	+21.8	-
0.0557	5	1-1/2	- 3.3	- 3.6	-	+13.2	+13.8	-	- 2.4	- 3.0	+18.9	+18.5	- 3.2	- 3.9	+14.0	+14.1
0.0747	7	1-1/2	- 2.2	- 2.5	-	+17.9	+18.8	-	- 1.7	- 2.2	+20.7	+20.2	- 2.1	- 2.8	+19.3	+20.1
0.0998	10	1-1/2	- 0.3	- 0.6	-	+12.4	+12.9	-	- 0.5	- 0.5	+12.2	+11.9	+ 0.7	- 1.0	+10.2	+10.9
0.131	13	1-1/2	- 0.5	- 0.8	-	- 4.2	- 3.7	-	- 0.8	- 0.9	- 3.6	- 3.9	- 1.3	- 1.4	- 5.2	- 5.2
0.194	18	1-1/2	+ 0.1	- 0.2	-	-12.4	-11.8	-	+ 0.6	- 0.1	-12.6	-12.2	- 0.3	- 1.0	-13.8	-13.9
0.235	24	1-1/2	- 0.6	- 1.1	-	-29.3	-29.8	-	- 0.2	- 1.2	-29.0	-29.1	- 1.3	- 2.0	-28.5	-28.8
0.318	32	1-1/2	- 2.8	- 3.2	-	-38.1	-37.4	-	- 2.6	- 3.7	-36.5	-35.9	- 3.4	- 4.2	-35.3	-34.6
0.417	41	1-1/2	- 5.0	- 5.5	-	-38.0	-38.4	-	- 5.3	- 6.1	-33.6	-34.5	- 5.8	- 6.7	-32.6	-32.6
0.557	58	3	- 7.9	- 8.3	-	-30.8	-30.7	-	- 8.0	- 8.0	-27.6	-22.8	- 7.3	- 8.6	-23.6	-26.1
0.744	72	3	-10.4	-10.9	-	-21.6	-20.8	-	-10.0	-10.6	-17.2	-17.0	- 9.9	-10.4	-14.5	-14.4
0.986	96	3	-10.4	-10.8	-	+ 6.7	+ 7.6	-	- 9.6	-10.1	+ 6.7	+ 5.0	- 9.3	-10.2	+ 6.4	+ 5.7
1.31	126	3	- 8.4	- 8.7	-	+15.7	+16.3	-	- 7.8	- 8.4	+15.2	+14.1	- 8.0	- 8.6	+15.2	+15.6
1.83	174	3	- 8.1	- 8.4	-	+25.5	+26.3	-	- 7.3	- 8.1	+25.1	+25.2	- 7.3	- 8.4	+23.9	+24.0
2.39	228	3	- 6.7	- 6.8	-	+35.8	+34.9	-	- 8.0	- 6.7	+35.6	+33.7	- 7.9	- 6.8	+36.0	+32.9
2.36	228	1-1/2	- 6.9	- 7.1	-	+34.4	+34.1	-	- 7.0	- 7.1	+32.9	+32.1	- 7.3	- 7.3	+34.5	+31.1
3.19	306	1-1/2	- 4.1	- 4.3	-	+43.5	+43.1	-	- 3.5	- 4.4	+42.9	+42.3	- 4.5	- 4.7	+45.5	+42.9
4.16	202	1-1/2	- 1.9	- 1.8	-	+44.7	+43.0	-	- 1.5	- 2.7	+45.9	+42.3	- 1.6	- 2.2	+39.7	+43.5
5.58	268	1/2	+ 1.3	+ 1.2	-	+35.0	+34.3	-	+ 1.3	+ 1.3	+36.5	+34.7	+ 1.2	+ 1.2	+38.5	+34.7
7.46	360	1/2	+ 2.9	+ 2.8	+2.9	+12.8	+13.3	+10.0	+ 3.1	+ 3.1	+13.8	+12.6	+ 2.7	+ 2.8	+15.5	+13.5
10.0	480	1/2	+ 1.9	+ 1.8	+1.6	- 5.1	- 4.1	0.0	+ 2.6	+ 2.1	- 3.8	- 4.2	+ 2.3	+ 1.9	- 3.7	- 3.8
13.2	630	1/2	+ 0.8	+ 0.6	+0.5	-10.8	- 9.4	-11.0	+ 1.0	+ 0.7	- 9.8	-10.3	+ 1.0	+ 0.6	- 9.6	- 9.4
18.2	870	1/2	- 0.2	- 0.2	-0.3	-12.5	-10.9	-10.0	+ 0.2	+ 0.0	-11.8	-12.4	+ 0.3	+ 0.1	-11.7	-12.5
23.5	1141	1/2	- 0.4	- 0.3	-0.3	-12.5	-11.9	-12.0	- 0.1	- 0.3	-12.6	-13.8	- 0.1	- 0.4	-13.1	-13.4
31.5	1531	1/2	- 0.5	- 0.4	-0.4	-14.9	-14.9	-15.0	- 0.5	- 0.5	-16.0	-14.2	- 0.2	- 0.4	-16.7	-16.3
41.1	1450	1/2	- 0.6	- 0.6	-0.5	-17.0	-17.1	-18.0	- 0.8	- 0.6	-19.0	-20.1	- 0.9	- 0.6	-19.4	-19.8
55.1	1929	1/2	- 0.7	- 0.7	-0.8	-21.1	-21.0	-22.0	- 0.4	- 0.6	-25.0	-22.5	- 0.6	- 0.8	-24.8	-24.6

Table 123-XII

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 1200, 9/10/62)

Power
Pressure
Rods Nos. 1-8

60 Mw
600 psig
48.0 in.

Rod No. 9
Water Level
Boric Acid
No. 9 Rod Worth

30.0 in.
16 ft-0 in.
0.46 gm/gal
5.4 x 10⁻⁴ δk/in.

ω	Cycles	Stroke, in.	OUT-OF-CORE						IN-CORE EPITHERMAL				IN-CORE THERMAL			
			Ana- log	Digi- tal	Null	Ana- log	Digi- tal	Null	Ana- log	Digi- tal	Ana- log	Digi- tal	Ana- log	Digi- tal	Ana- log	Digi- tal
			db	db	db	φ	φ	φ	db	db	φ	φ	db	db	φ	φ
0.0101	1	1-1/2	-11.8	-11.8	-	+35.7	+40.7	-	-	-	-	-	-13.8	-14.4	+30.2	+35.8
0.0132	1	1-1/2	-8.3	-8.2	-	+37.5	+38.4	-	-	-9.5	-	+31.2	-9.6	-9.7	+33.1	+33.6
0.0184	2	1-1/2	-8.4	-8.2	-	+44.6	+45.8	-	-	-7.7	-	+45.2	-10.0	-9.8	+37.9	+39.8
0.0235	3	1-1/2	-7.4	-7.3	-	+40.8	+40.6	-	-	-8.6	-	+42.8	-6.7	-6.6	-	+40.9
0.0319	3	1-1/2	-5.1	-4.9	-	+32.7	+33.2	-	-4.6	-4.4	+38.6	+40.9	-3.5	-3.6	+32.9	+34.0
0.0417	4	1-1/2	-5.1	-4.6	-	+28.7	+27.2	-	-4.3	-5.0	+33.2	+34.7	-4.6	-4.3	+28.4	+28.3
0.0558	5	1-1/2	-5.2	-4.8	-	+38.1	+37.2	-	-5.2	-5.1	+43.8	+44.1	-4.9	-4.6	+38.3	+38.6
0.0743	7	1-1/2	-2.3	-2.1	-	+14.0	+14.2	-	-2.7	-2.7	+18.6	+20.5	-2.5	-2.6	+14.2	+15.2
0.0997	10	1-1/2	-1.6	-1.5	-	+12.5	+14.0	-	-1.6	-1.5	+14.0	+16.1	-1.9	-1.9	+13.6	+15.3
0.132	13	1-1/2	-0.5	-0.3	-	-6.5	-5.4	-	-0.6	-0.6	-6.5	-3.7	-1.0	-1.1	-7.1	-6.0
0.184	17	1-1/2	-2.8	-2.7	-	-24.1	-22.7	-	-2.7	-2.7	-25.9	-23.5	-3.9	-3.9	-23.7	-22.8
0.236	24	1-1/2	-1.8	-2.0	-	-42.4	-40.3	-	-1.8	-2.3	-41.9	-38.2	-2.5	-2.8	-39.9	-38.0
0.325	32	1-1/2	-4.6	-4.8	-	-37.9	-36.5	-	-4.6	-5.1	-38.0	-35.0	-5.2	-5.6	-34.4	-33.4
0.414	41	3	-8.6	-8.6	-	-47.7	-46.5	-	-	-8.8	-	-34.8	-9.4	-9.5	-40.0	-39.1
0.556	54	3	-11.3	-11.6	-	-30.2	-28.9	-	-	-11.9	-	-23.9	-11.0	-11.2	-21.0	-20.7
0.756	72	3	-11.5	-12.0	-	-24.3	-23.0	-	-11.4	-11.4	-22.3	-18.5	-10.9	-11.4	-16.6	-14.5
1.01	96	3	-15.1	-15.1	-	-13.4	-11.1	-	-14.4	-9.6	-13.8	-21.5	-13.8	-13.6	-6.9	-7.7
1.32	126	3	-12.2	-12.1	-	+17.4	+19.6	-	-12.1	-12.0	+13.9	+15.2	-11.4	-11.3	+14.7	+16.9
1.84	174	3	-12.2	-11.9	-	+24.5	+20.2	-	-11.9	-11.6	+21.2	+24.2	-11.2	-11.3	+21.9	+22.3
2.41	228	3	-10.6	-10.2	-	+33.8	+34.4	-	-10.5	-10.2	+29.4	+31.7	-10.1	-10.1	+29.4	+31.2
3.19	306	1-1/2	-9.6	-9.3	-	+46.9	+48.0	-	-10.7	-9.2	+38.5	+45.2	-9.5	-9.2	+41.9	+44.8
4.15	202	1-1/2	-6.9	-6.7	-	+57.5	+58.3	-	-	-8.9	-	+65.5	-8.2	-7.2	+51.6	+56.2
5.58	268	1-1/2	-3.1	-2.9	-	+65.7	+66.3	-	-3.3	-3.1	+66.0	+66.3	-3.5	-3.3	+65.3	+66.4
7.54	360	1/2	+1.9	+2.1	-	+54.2	+54.9	-	+1.5	+1.9	+54.0	+55.0	+2.1	+2.2	+54.0	+54.7
10.0	480	1/2	+5.5	+5.6	+5.6	+13.2	+15.3	-4.0	+5.4	+5.5	+12.2	+15.1	+5.6	+5.7	+14.0	+15.3
13.2	630	1/2	+2.5	+2.7	+2.6	-13.1	-10.5	-11.0	+2.7	+2.5	-15.3	-11.1	+2.6	+2.4	-13.5	-11.4
18.1	870	1/2	+0.4	+0.4	+1.0	-16.3	-13.2	-12.0	+0.4	+0.4	-19.6	-14.3	+0.6	+0.5	-17.0	-14.4
23.8	1141	1/2	-0.2	-0.1	+0.3	-16.3	-13.4	-12.0	-0.2	-0.2	-19.7	-14.8	-0.2	-0.2	-16.9	-14.1
31.9	1531	1/2	-0.5	-0.4	-0.3	-18.6	-15.8	-15.0	-0.6	-0.4	-21.2	-18.7	-0.3	-0.3	-19.3	-17.2
41.6	1450	1/2	-0.7	-0.7	-0.6	-20.8	-18.1	-18.0	-0.5	-0.6	-25.5	-20.7	-0.7	-0.7	-22.5	-21.0
55.3	1929	1/2	-0.7	-0.7	-0.8	-24.2	-21.5	-25.0	-0.6	-0.7	-31.1	-25.3	-0.7	-0.7	-27.2	-25.0

Table 123-XIII

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 1300, 10/16/62)

Power	67 Mw	Rod No. 9	30.0 in.
Pressure	595 psig	Water Level	16 ft-0 in.
Rods Nos. 1-8	48.0 in.	Boric Acid	2.35 gm/gal
		No. 9 Rod Worth	4.2×10^{-4} δ k/in.

			OUT-OF-CORE		IN-CORE EPITHERMAL		IN-CORE THERMAL			
			Analog		Analog		Analog Digital		Analog Digital	
ω	Cycles	Stroke, in.	db	ϕ	db	ϕ	db	db	ϕ	ϕ
5.72	268	1/2	-1.7	+68.2	-2.6	+67.0	-3.0	-2.7	+70.4	+70.0
6.54	268	1/2	+0.6	+69.1	-0.3	+67.3	-0.6	-0.3	+70.1	+69.2
7.47	360	1/2	+3.0	+63.7	+1.9	+61.5	+2.5	+2.6	+64.1	+63.3
8.63	360	1/2	+6.1	+47.7	+5.7	+46.8	+5.7	+5.8	+48.2	+48.5
9.97	480	1/2	+7.1	+19.7	+6.8	+17.9	+7.0	+7.1	+19.4	+19.7
11.4	480	1/2	+5.4	-4.5	+5.0	- 5.4	+5.2	+5.2	- 5.5	- 4.0
13.1	630	1/2	+3.2	-12.9	+3.2	-13.3	+2.9	+3.0	-14.4	-12.0
18.4	870	1/2	+0.5	-14.6	+0.5	-15.7	+0.3	+0.5	-17.6	-14.4
23.6	1141	1/2	-0.1	-13.5	-0.1	-14.7	-0.3	-0.2	-16.9	-14.8
31.6	1531	1/2	-0.5	-15.4	-0.5	-16.9	-0.5	-0.6	-19.7	-17.0
41.1	1450	1/2	-0.7	-17.9	-0.7	-19.5	-0.6	-0.6	-21.4	-18.6
54.7	1929	1/2	-0.6	-21.3	-0.6	-22.5	-0.6	-0.6	-25.9	-23.0

Table 123-XIV

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 1400, 10/18/62)

Power	70.7 Mw	Rod No. 9	30.0 in.
Pressure	590 psig	Water Level	16 ft-0 in.
Rods Nos. 1-8	40.6 in.	Boric Acid	1.1 gm/gal
		No. 9 Rod Worth	$4.1 \times 10^{-4} \delta k/in.$
		Carryover	9.1%

			OUT-OF-CORE		IN-CORE EPITHERMAL		IN-CORE THERMAL			
			Analog		Analog		Analog	Digital	Analog	Digital
ω	Cycles	Stroke, in.	db	ϕ	db	ϕ	db	db	ϕ	ϕ
0.554	54	4-1/2	-14.8	-24.2	-14.2	-17.1	-13.5	-13.9	- 9.5	-11.9
0.719	72	4-1/2	-12.5	-15.8	-12.0	-11.3	-11.7	-11.9	- 6.0	- 8.1
0.963	96	4-1/2	-13.9	-20.4	-13.0	-13.4	-12.8	-12.7	- 6.0	- 7.9
1.33	126	4-1/2	-14.7	+18.7	-13.4	+17.7	-12.8	-13.2	+17.5	+14.7
1.69	162	2-1/4	-12.4	+24.6	-11.8	+25.2	-11.3	-11.8	+25.3	+24.2
1.82	174	4-1/2	-11.9	+20.4	-11.1	+20.6	-11.3	-11.6	+21.4	+20.7
2.34	228	4-1/2	-12.1	+31.2	-11.8	+28.7	-11.0	-11.0	+28.5	+27.4
3.21	306	4-1/2	-11.7	+39.3	-12.2	+35.4	-10.8	-10.9	+35.7	+35.6
4.15	202	2-1/4	-10.7	+52.1	-10.2	+51.5	- 9.6	-10.1	+49.9	+49.7
5.51	268	2-1/4	- 6.3	+71.9	- 6.1	+71.9	- 5.9	- 6.2	+70.6	+70.3
6.59	310	3/4	- 3.5	+74.2	- 2.6	+77.1	- 2.6	- 3.1	+72.7	+74.9
7.60	360	3/4	- 0.7	+71.0	- 0.2	+73.3	- 0.0	- 0.5	+70.8	+69.3
8.71	410	3/8	+ 2.8	+64.2	+ 3.5	+67.4	+ 3.6	+ 3.0	+64.8	+66.3
9.98	480	3/8	+ 6.7	+45.4	+ 7.1	+48.2	+ 7.2	+ 6.9	+48.3	+47.7
11.5	500	3/8	+ 7.8	+ 8.8	+ 8.0	+11.3	+ 8.0	+ 7.7	+12.1	+11.5
13.2	630	3/8	+ 5.3	-12.5	+ 5.4	- 9.9	+ 5.2	+ 5.1	- 7.7	- 8.7
13.2	630	3/4	+ 5.5	-12.2	+ 5.6	-10.6	+ 5.2	+ 5.2	- 9.8	- 9.7
18.4	870	3/4	+ 0.9	-19.8	+ 0.9	-16.8	+ 0.8	+ 0.8	-16.0	-16.3
23.3	1141	3/4	- 0.1	-18.3	- 0.1	-14.8	- 0.1	- 0.1	-13.2	-14.5
31.6	1531	3/4	- 0.4	-19.0	- 0.5	-15.5	- 0.5	- 0.4	-14.2	-15.7
41.3	1450	3/4	- 0.6	-21.2	- 0.5	-17.5	- 0.5	- 0.6	-16.8	-18.2
53.3	1929	3/4	- 0.6	-24.2	- 0.7	-20.9	- 0.7	- 0.6	-20.6	-21.5

Table 123-XV

TRANSFER FUNCTION EXPERIMENTAL DATA
(Exp. No. 150, 4/6/62)

Power	20 kw	Water Level	16 ft-0 in.
Pressure	600 psig	Boric Acid	0.15 gm/gal
		No. 9 Rod Worth	$7.7 \times 10^{-4} \delta k/\text{in.}$

			OUT-OF-CORE			
			Analog	Digital	Analog	Digital
ω	Cycles	Stroke, in.	db	db	ϕ	ϕ
10.2	480	1/2	-0.3	-	- 7.7	-
13.3	630	1/2	+0.0	-	- 8.2	-
18.5	870	1/2	-0.3	+0.1	-11.5	-11.0
24.2	1141	1/2	-0.6	-0.6	-12.2	-12.5
31.8	1531	1/2	-0.4	-0.5	-14.4	-15.2
41.2	1450	1/2	-0.4	-0.8	-17.9	-17.8
55.9	1929	1/2	-0.4	-0.3	-23.2	-22.0

The absolute gain of the reactor is not measured directly. The block diagram illustrating reactor behavior at power is shown below.

123-SK-1.

The power transfer function is

$$P = \frac{G}{1 + GH} = \frac{\Delta n/n_o}{\delta k/\beta}$$

where: G - zero power transfer function

H - feedback transfer function

P - power transfer function.

If the reactor is excited with a sinusoidal reactivity input, $\delta k/\beta \sin \omega t$, then G, H, P, and $\Delta n/n_o$ are all functions of frequency. $\Delta n/n_o$ is measured directly as a percentage, but $\delta k/\beta$ is unknown. The data is normalized by taking advantage of the fact that as $\omega \rightarrow \infty$, $H \rightarrow 0$ and $P \rightarrow G$. The amplitude data at the three highest frequency points $\omega = 32, 42$, and 56 rad/sec are adjusted to give a best fit to the zero power transfer function at these frequencies. The rod worth is then calculated from

$$\frac{\delta k}{\text{in.}} = \frac{\Delta n}{n_o} \times \frac{1}{(\text{normalized gain})} \times \frac{1}{(\text{stroke})} \times \beta$$

Three techniques were used to measure the reactor transfer function. The first technique was based on null analysis which is described in Reference 1. The second technique employed an analog computer to do a Fourier analysis by computing the following integrals:

$$\begin{aligned} \frac{1}{KT} \int_0^{KT} \left[n_o + \Delta n \sin(\omega t + \phi) + N(t) \right] \sin(\omega t) dt &= \frac{\Delta n \cos \phi}{2} \\ &+ \frac{1}{KT} \int_0^{KT} N(t) \sin(\omega t) dt \\ \frac{1}{KT} \int_0^{KT} \left[n_o + \Delta n \sin(\omega t + \phi) + N(t) \right] \cos(\omega t) dt &= \frac{\Delta n \sin \phi}{2} \\ &+ \frac{1}{KT} \int_0^{KT} N(t) \cos(\omega t) dt \end{aligned}$$

- K - number of cycles
- T - time for one cycle
- n_0 - average neutron flux
- Δn - peak value of sinusoidal component of neutron flux
- ω - angular frequency
- t - time
- ϕ - phase shift of flux signal
- N(t) - noise component of flux signal

The integration shown above assumed an integral number of cycles. KT is the total integrating time. The presence of a noise signal, N(t), gives rise to an error term. The effect of this term can be reduced by selecting the total time to give a desired accuracy. If the noise error terms are assumed to be negligible, then the amplitude of the transfer function is given by

$$\Delta n = 2 \sqrt{\left(\frac{\Delta n \cos \phi}{2}\right)^2 + \left(\frac{\Delta n \sin \phi}{2}\right)^2}$$

The phase shift is given by

$$\phi = \tan^{-1} \frac{1/2 \Delta n \sin \phi}{1/2 \Delta n \cos \phi}$$

The third technique employed a digital recorder and digital computer program to carryout the Fourier analysis.

The effect of chamber location on the reactor transfer function was investigated and is reported in Table 123-XI. The out-of-core ionization chamber was located in an instrument thimble in the reactor shield. The in-core ionization chambers were small enough to be placed inside a fuel element. The epithermal chamber was covered with cadmium.

The RMS (root mean square) value of the reactor noise was computed using the analog computer and digital recorder in combination with a digital computer. The results of these measurements are reported in Table 123-XVI.

Table 123-XVI

ROOT MEAN SQUARE (RMS)

Pressure, psig	Power, Mw	Water Level, ft-in.	Boric Acid Concentration, gm/gal	Feedwater Ring	Rods Nos. 1-8, in.	Rod No. 9, in.	Analog, %	Digital, %
300	10.0	16-0	4.0	Lower	27.3	23.0	0.97	0.91
300	10.0	16-0	0.1	Upper	17.8	14.5	1.06	1.01
300	20.0	16-0	4.0	Lower	32.2	23.0	1.72	1.89
300	20.0	16-0	0.2	Upper	22.7	14.5	3.20	3.18
600	11.5	16-0	5.0	Lower	29.1	23.0	0.81	0.80
600	10.1	16-0	0.2	Upper	18.7	14.5	0.90	1.23
	10.0	16-0	0.2	Lower	18.4	14.5	0.77	0.78
600	13.4	16-0	0.2	Upper		14.5	1.31	1.38
600	20.1	16-0	3.9	Lower	30.0	23.0	0.97	1.01
600	20.0	16-0	0.1	Upper	22.4	14.5	1.88	1.96
600	40.0	16-0	0.2	Upper	29.2	21.5	3.27	3.20
600	40.0	16-0	2.7	Upper	48.0	30.0	4.28	4.01
600	60.0	16-0	0.5	Upper	48.0	30.0	5.64	4.96
600	70.7	16-0	1.1	Upper	40.6	30.0	6.26	7.43
600	71.0	16-0	1.0	Upper	34.7	39.8	4.72	4.55
600	79.3	16-0	1.1	Upper	35.2	39.8	5.20	5.15
*600	74.8	16-0	1.7	Upper	37.0	30.0	4.86	4.93
*600	75.8	16-0	1.7	Upper	34.6	34.6	4.11	4.04
*600	76.8	16-0	1.2	Upper	35.0	35.0	4.19	
*600	72.0	16-0	1.2	Upper	34.0	40.0	3.66	3.45
*600	82.0	16-0	1.2	Upper	36.7	36.7	6.88	5.78
*600	90.7	16-0	1.2	Upper	37.7	37.7	8.47	6.57
*600	100.0	15-9	0.1	Upper	48.0	48.0	6.01	6.77
*600	60.0	16-0	0	Upper	30.0	40.0	2.94	2.65
*600	56.0	15-3	0	Upper	32.0	40.0	1.38	1.35
*600	60.7	15-3	0	Upper	34.0	40.0	1.56	1.47
*600	78.1	15-3	0	Upper	48.0	48.0	2.65	2.59
*600	74.7	16-0	0	Upper	34.0	40.0	3.51	3.50
*600	82.0	14-3	?	Upper	48.0	48.0		2.21
*600	84.6	15-3	?	Upper	48.0	48.0		3.35

* Automatic pressure control.

REFERENCE

- (1) J. A. DeShong, Jr., "Power Transfer Functions of the EBWR Obtained Using a Sinusoidal Reactivity Driving Function," ANL-5798 (January, 1958).

Section IV. CONTROL AND CONTROL RODS

<u>Title</u>	<u>Page</u>
Investigation of Irradiated EBWR Control Rods (Test Report No. 121A, dated November 15, 1960, by T. L. Kettles)	289
Investigation of Irradiated EBWR Control Rods (Test Report No. 121B, dated February 28, 1961, by T. L. Kettles)	294
Investigation of Irradiated EBWR Control Rods (Test Report No. 121C, dated March 21, 1961, by T. L. Kettles)	307
Radiochemical Analysis of Original EBWR Control Rod and Follower Samples (Test Report No. 121D, dated September 8, 1961, by R. J. Armani)	322
Boron Burnup in Original EBWR Control Rod No. 4 (Test Report No. 121E, dated September 21, 1961, by R. J. Armani)	325
Radiochemical Analysis of Crud Removed from the Bottom of EBWR Control Rods No. 4 and No. 9 (Test Report No. 112, dated November 30, 1960, by R. J. Armani)	326
Disposition of Original EBWR Control Rods (Test Report No. 121F, dated October 27, 1961, by T. L. Kettles)	327
Partial Investigation of Irradiated EBWR Control Rods (Test Report No. 121G, dated January 28, 1963, by T. L. Kettles and N. Balai)	329
Effect of U ²³⁵ Containing Stainless Steel Followers on EBWR Reactivity and Control Rod Worth (Test Report No. 100C, dated May 11, 1960, by K. Almenas)	339
Calibration of Control Rod No. 9 (Test Report No. 100E, dated June 14, 1961, by K. Almenas)	345
EBWR Control Rod Replacement (Test Report No. 119, dated January 8, 1962, by V. M. Kolba, E. A. Wimunc, E. L. Martinec, and J. F. Matousek)	353
A Preliminary Calibration of the New System of Control Rods in the Cold EBWR (Test Report No. 100G, dated January 22, 1962, by H. P. Iskenderian)	363
Inspection of the EBWR Control Rod (Test Report No. 119A, dated July 6, 1962, by T. L. Kettles)	367
Calibration of Control Rod System in EBWR - Empirical Determination of β eff (Test Report No. 100I, dated January 31, 1963, by H.P. Iskenderian)	368

INVESTIGATION OF IRRADIATED EBWR CONTROL RODS

(Test Report No. 121A, dated

November 15, 1960, by T. L. Kettles)

INITIAL PREPARATION OF CONTROL RODS FOR INVESTIGATION

The No. 4 boron and No. 9 hafnium rods were each cut into thirty pieces approximately 15 cm long, and three pieces of varying lengths were cut from the No. 1 hafnium rod using a reciprocating saw located in the EBWR fuel storage pit. The saw, as shown in Fig. 121A-1, was suspended under 2 m of water which eliminated radiation hazards during the sawing operation. Prior to sawing, each control rod was identified at each 6-in. interval on each blade surface. A symbol identification (two digits and one letter)

Fig. 121A-1. Rig for Cutting Irradiated EBWR Control Rods Assembled in Fuel Storage Pit

Fig. 121A-2. Radioactivity Measurement of EBWR Irradiated Control Rods

water indicates mean energies to be less than 0.15 Mev for Zircaloy-2 and 0.06 Mev for hafnium. Specific energy spectrums will be determined later from laboratory size material samples removed in the caves.

The No. 9 hafnium rod, which is at the center of the nine EBWR control rod configuration, shows up to 17% variation in the radioactivity of its four quadrants, based on the measurement of a significant number of pieces. The No. 4 boron rod, which is a corner rod, shows a variation of 80%, but this figure is based on measurement of only one piece to date.

The relatively high radioactivity of the bottom pieces (No. 30, Fig. 121A-2) of No. 4 and No. 9 rods was found caused by an accumulation of crud in the pocket formed by the intersection of the four angles of which the rods are made. This crud probably originated in the core region and silted through the slight openings between adjacent angle faces to the center hollow stem where it settled to the bottom. An analysis now being made shows the crud included the nuclides Co^{58} , Co^{60} , and Cs^{137} .

HAFNIUM-ZIRCALOY-2 TRANSITION

This transition for the No. 9 rod was found to have parted when the rod was removed from the reactor. Whether the failure happened during operation or upon removal is not known but visual examination of the pieces (Nos. 8 and 9, Fig. 121A-2), 0.9 m under water, revealed the failure developed in either two or three stages, with final parting occurring near or at the end of reactor operation.

The failure is shown in Fig. 121A-3, the hafnium piece being the one with the vee-shaped notch at its top. The right side-lap on the hafnium piece shows failed spotwelds covered with a uniform corrosion product, light brown in color under water, but the left side-lap shows not only freshly sheared spotwelds but a bright metallic map-shaped area outlining the tight fitting portion of the lap joint. A similar area appears on one other blade, revealing the history of failure; initially two blades failed either separately or simultaneously, then the remaining two blades failed simultaneously.

The No. 9 rod has been oscillated, to obtain power transfer functions, so a logical question is whether the high dynamic forces so imposed on the transition had caused its failure? Since the oscillation work had been done during two periods over a year apart, this fact would explain the separate failure stages. The hafnium-Zircaloy-2 transition of the No. 1 rod (piece No. 9, Fig. 121A-2), which had not been oscillated, was cut out for comparison and is shown in Fig. 121A-4. The hafnium portion is marked 40C in the upper right hand corner.

There is no sign of distress in any part of this transition piece; the slight parting of the Zircaloy-2 portion in the lower left corner was torn up

Fig. 121A-3. Hafnium-Zircaloy-2
Transition, No. 9 EBWR
Control Rod Showing
Spot Weld Failures

Fig. 121A-4. Hafnium-Zircaloy-2
Transition, No. 1
EBWR Control Rod

by the saw blade during cutting. It is concluded that failure of the hafnium-Zircaloy-2 transition of No. 9 control rod was caused by the forces imposed on it during oscillation.

2% BORON-STAINLESS STEEL-ZIRCALOY-2 TRANSITION

An alarming fact determined during this investigation of EBWR control rods is the condition of the subject transition as revealed in Fig. 121A-5. The countersunk heads at one end of over 90% of the rivets had popped off. The rivets remained in their holes with their opposite heads intact, but the failures allowed the 0.16-cm thick Zircaloy-2 side sheets to open up causing a wedging action in the core shroud guides.

The extreme corners of this transition were abraded by this wedging action each time the control rod was raised or lowered. A typical wear area so caused can be clearly seen in the figure as a small dark triangle at the upper left corner of the left-size Zircaloy-2 sheet. Fortunately, lateral forces developed by the wedging action during control rod scram tended to close the wedge.

Fig. 121A-5. 2% Boron-Stainless Steel-Zircaloy-2 Transition, No. 4 EBWR Control Rod Showing Rivet Failures

Detailed examination of the transition, to define the cause(s) of rivet failure, had to be done in the caves. Meanwhile, two rivets from the top of control rod No. 9 (piece No. 1, Fig. 121A-2), which had failed in a similar manner, were removed and were also examined. Microanalysis indicated a rather distinct line of demarcation between the work hardened rivet heads and the softer rivet body at about the line of failure.

GENERAL INVESTIGATION PROCEDURE

Twenty-one of the 63 pieces of control rods now available for examination measure 200 or less mr/hr at 5 cm in air, and investigation of these pieces has begun in building D-331. The remaining 42 pieces are to

be investigated in building D-301 caves. As representative data are obtained, it will be possible to determine which pieces need only cursory examination and which require detailed analysis.

Mechanical tests will include measurement of tensile strength, hardness, bending limitations, and microanalysis to determine the effect of radiation on material properties. Chemical analysis will include measurement of boron burnout. Radioanalysis will determine the major long-lived nuclides in the control rod materials and in the associated crud.

INVESTIGATION OF IRRADIATED EBWR CONTROL RODS

(Test Report No. 121B, dated February 28, 1961, by T. L. Kettles)

The first report of this series⁽¹⁾ was based on information obtained incidental to cutting control rods No. 4 and No. 9, into lengths convenient for detailed investigation in building D-301 caves. A more specific examination has now been completed and the information obtained regarding rivets and crud deposition is included in this report. A future report will concern the hafnium and 2% boron-stainless steel poison sections.

PHYSICAL ARRANGEMENT FOR EXAMINATIONS

In Fig. 121B-1 the EBWR fuel storage pit is shown prepared for control rod examination. The aluminum tube and angle rack can support the longest (5-1/2 ft) control rod section available, and it can be vertically adjusted to provide a water shield up to 7 ft deep. The handling tool employs the off-center toggle principle for positive clamping action which provides sufficient gripping power to manipulate even the long rod sections into and out of the rack.

RIVETS

Introduction

The brittle condition of rivets connecting the 2% boron-stainless steel poison section of No. 4 control rod to its Zircaloy-2 follower, and the similar condition of two rivets connecting the retrieving tip of No. 9 rod to its hafnium poison section, was previously described.⁽¹⁾ Investigation of rivets from the control rods proper has continued and has been supplemented by investigation of unused rivets for comparison.

Investigation of Control Rod Rivets

1. Micro-examination of Whole Rivet from Piece No. 1 of Control Rod No. 9.

Fig. 121B-1. Tube and Angle Rack in EBWR Fuel Storage Pit Prepared for Control Rod Examination

A small piece of control rod cross section, encompassing a whole rivet, was hack-sawed from the retrieving end of No. 9 control rod. The piece was ground and polished diametrically through the rivet's longitudinal axis for micro-examination. Two cracks were revealed by the

examination, extending from the intersection of rivet head and shank along a boundary between severely crushed crystals in the head and less severely crushed crystals in the shank.

Both cracks are of a massive nature that mask the type of failure, but crystals at their edges reveal predominantly intergranular separations. There is little indication of carbide precipitation at grain boundaries and no branching cracks of any consequence. Inconel side plates, part of the retrieving end structure, show No. 6 austenitic grain size, extensive carbide precipitation in grain boundaries, but no evidence of cracking.

2. Examination of Riveted Transitions

The conclusion from (1) above, that a rivet might look intact but be in a failed condition, indicated the need for a comprehensive investigation of all control rod riveted transitions. Completion of the rack for underwater use, in the EBWR fuel storage pit, made such an investigation possible.

The original EBWR control rods include three types of riveted transitions: (a) No. 1, connecting the retrieving tips to hafnium poison sections (rods Nos. 1, 3, 5, 7, and 9); (b) No. 9, connecting the 2% boron-stainless steel poison sections to Zircaloy-2 followers (rods Nos. 2, 4, 6, and 8); and, (c) No. 19, connecting the 3-in. Zircaloy-2 followers to 3-in. Type 304 stainless steel followers (rods Nos. 1 through 9). The transition numbers when divided by 2 provide the approximate location (in feet) of the transition from the retrieving tip.

Fig. 121B-2. Riveted Transition Wedge Test On Original EBWR Control Rods (Transition No. 19; Rods [Nos. 2, 4, 6, and 8] Shown)

The No. 9 transitions were examined visually, through a 3-ft water depth, and the number of missing rivet heads counted. The No. 1 and No. 19 transitions were investigated by means of a "wedge test."

The wedge test, as made through a 3-ft water depth, is shown in Fig. 121B-2. A long handled tool, with chisel end, was driven between transition blades, and the nature of resultant rivet failure noted. Three types of failures occurred: (a) "very brittle," the rivet heads popped upon initial wedge penetration; (b) "brittle," the rivet heads popped upon substantial wedge penetration; and (c) "ductile," the rivet heads started to pull through the blades upon substantial wedge penetration.

For all tests the 1/16-in. thick Inconel (transition No. 1) and the 1/16-in. thick Zircaloy-2 and the 1/16-in. thick Type 304 stainless steel (transition No. 19) bent open ductilely. Test locations were selected at random, and an average of two tests were made for each transition. The results are summarized in Table 121B-I.

Table 121B-I

EXAMINATION OF RIVETED TRANSITIONS

Rod No.	<u>Transition No. 9</u> % Rivet Heads Missing	<u>Transition No. 1</u> Type Failure	<u>Transition No. 19</u> Type Failure
1	—	Very Brittle, Brittle, & Ductile	Brittle
2	15	—	Brittle
3	—	Ductile	Brittle
4	94*	—	Ductile
5	—	Ductile	Very Brittle
6	8	—	Brittle & Ductile
7	—	Brittle	Very Brittle
8	0	—	Ductile
9	—	Very Brittle	Brittle

* This high percentage of missing rivet heads is not necessarily significant because No. 4 control rod was more severely handled, having been clamped in a hydraulic press for sawing into 6-in. long pieces. (1)

3. Examination of Rivet Head Fracture Surfaces

Twenty rivet heads collected during the wedge test were examined at up to 96-power magnification. The appearances of their fracture surfaces are schematically represented in Fig. 121B-3 and correlated with the manner in which they had failed.

"VERY BRITTLE" FRACTURE

NATURE - SHALLOW CUP.

"BRITTLE" FRACTURE

NATURE - SHALLOW CUP, USUALLY ASSOCIATED WITH A NARROW LEDGE AS SHOWN. THE LEDGE SOMETIMES APPEARS AS CODE 2.

"DUCTILE" FRACTURE

NATURE - FLAT, SOMETIMES ASSOCIATED WITH A LEDGE. SOMETIMES THE CODE 1 AREA IS ISOLATED AS AN ISLAND.

NO RIVET HEAD REMOVED BY THE WEDGE TEST REVEALED A PURE CODE 1 APPEARANCE.

CODE:

- 1 BRIGHT SILVER GRANULAR APPEARANCE. LOOKS LIKE NEW METAL.
- 2 BLACK COARSE GRANULAR APPEARANCE WITH SILVER POOLS.
- 3 BLACK COARSE GRANULAR APPEARANCE WITH SILVER POOLS AND WITH A GOLD TINT OF VARYING DENSITY ON THE SILVER POOLS. SOME INTERMITTENT POOLS ARE BLUE.
- 4 RED CRUD DEPOSIT.

SCALE: 1/8"

Fig. 121B-3. EBWR Control Rod Rivet Head Fractures

All fractures, "very brittle," "brittle," and "ductile" reveal metal deterioration beginning at the intersection of rivet head and shank. The bright silver areas represent sound metal and their boundaries of black granular areas represent the extent of deterioration. The extent of gold discoloration on the latter areas is believed to show the extent of crack propagation that allowed entrance of reactor water to produce incipient corrosion.

A red color on a fracture area reveals a location of crud deposition. When red color could not be traced up the tapered rivet head sides to the fracture edge, no red color was seen on the fracture area.

4. Investigation to Detect Presence of Chlorides on Rivet Head Fracture Surfaces

The mechanism most commonly reported as cause for rapid failure of austenitic stainless steels is chloride stress corrosion. It seemed logical, therefore, to investigate the rivet head fracture surfaces for the presence of chlorides. The most abundant isotope of chlorine is Cl^{35} (75%). A neutron gamma reaction will produce Cl^{36} which decays by emission of beta rays, with energies up to 0.71 Mev, and has a half life of 3.1×10^5 years.

The fracture surfaces of fourteen rivet heads were etched and the sample was separated by chemical carrier techniques. The separated compound was counted and found to measure an average of 28 counts/min. Absorption of beta rays in aluminum was plotted and showed maximum beta ray energy to be about 0.7 Mev but the rate of decay proved to be too great to be attributed to chlorine.

Investigation of Unused Rivets

1. Samples

A bag of 1/8-in. diameter x 3/8-in. long countersunk head rivets identified with the original EBWR hafnium control rods' shop order number was discovered in Central Shops' stock. These rivets are being investigated on the assumption they are representative of rivets used for control rod fabrication.

2. Chemical Analysis

A wet chemical analysis was made of eighteen rivets, randomly selected, to determine weight-percent constituency of certain elements of interest. Results are as follows:

w/o					
C	Ni	Cr	P	S	Se
0.08	11.45	19.55	0.024	0.012	<0.01

The amounts of C, Ni, Cr, P, and S are within limits required by AISI specifications for Type 304 stainless steel with the exception of nickel which is slightly high (11.45 versus 11.00). The amounts of P, S, and Se individually, or in combination, are not considered large enough to be responsible for control rod rivet embrittlement.

3. Mechanical Properties

Twenty-one sample rivets were measured for hardness and revealed a variation from Rb 39 to Rb 110. Obviously, more than one batch of rivets was contained in the bag found in stock. However, the rivets could be separated into seven major groups (some odd ones left over) by characteristic "fingerprints," made by heading and clipping machinery, and by coloring indicative of their final heat treatment.

For investigative purposes, the rivets were classified by hardness of their shank ends into two groups: (a) "soft," hardness Rb 77 or less; and (b) "hard," hardness Rb 105 or more. Table 121B-II lists the hardness measurements made and includes some rivets measured in a riveted condition. The table reveals an important fact; the heads of the size rivets in question approach a limit of hardness when riveted, regardless of whether the rivet was initially "soft" or "hard."

Table 121B-II

RIVET HARDNESS

Rivet Group	Hardness (Rb) as Received		Hardness (Rb) after Riveting	
	Shank End	Manufactured Head	Riveted Head	Manufactured Head
Soft ↓	--	39	---	---
	60	41	112	101
	60	70	---	---
	61	65	---	---
	61	68	---	---
	64	71	---	---
	65	60	---	---
	65	70	112	101
	65	74	---	---
	67	70	112	100
	70	75	---	---
	77	65	112	105
	--	--	110	102
	--	--	110	103
	--	--	108	100
Hard ↓	105	91	---	---
	106	107	---	---
	107	93	---	---
	108	89	---	---
	108	98	---	---
	109	69	---	---
	109	74	113	111
	109	108	---	---
	110	83	---	---
	---	---	111	110
	---	---	113	111

Qualitative type tensile tests were made on riveted rivets. A rivet was riveted through two blocks, placed one atop the other, then the blocks were pulled apart causing the rivet to fail in tension. Initially "soft" and "hard" rivets both failed ductilely in their shanks with no cracking at their heads.

4. Carbide Precipitation

Microscopic examination of both "soft" and "hard" rivets did not reveal massive carbide precipitation in grain boundaries.

5. Stress Corrosion

Riveted rivets developed cracks at the intersection of head and shank when immersed for a period of time in a boiling solution of magnesium chloride.

Discussion

1. Introduction

The mechanisms most suspect, as the cause of control rod rivet failures, are stress concentration, carbide precipitation, radiation damage, and stress corrosion. In order to evaluate them, each is considered separately below as having been the sole possible cause.

2. Stress Concentration

For - Control rod rivets failed invariably in the region between head and shank where the grain structure changed character and orientation.

Against - The heads of sample rivets developed full strength of their shanks in tensile tests.

3. Carbide Precipitation

For - Micro-examination of a failed rivet showed mainly intergranular type grain separations.

Against - Micro-examination of a failed rivet and sample rivets did not reveal massive carbide precipitation in grain boundaries.

Control rod rivets failed rapidly. Some rivets failed within one year after EBWR began operation.

4. Radiation Damage

For - Visual inspection revealed most missing rivet heads on those transitions that received highest nuclear radiation.

Against - Gross rivet failures also occurred on transitions that received relatively low nuclear radiation.

Gross rivet failures occurred adjacent to "ductile" rivet failures on those transitions that received relatively low nuclear radiation.

The geometry of the transitions showing the most missing rivet heads is conducive to a greater retention of foreign matter (see No. 5 below).

5. Stress Corrosion

For - Rivets were highly stressed.

Rivets attained a temperature of at least 488°F during reactor operation.

Control rod sandwich construction is conducive to the retention of foreign matter between the blades. This fact was discovered during the use of liquid dye penetrant to reveal cracks in 2% boron-stainless steel poison sections.

Control rod rivets failed rapidly. Some failed within one year after EBWR began operation.

Sample rivets cracked when boiled in an aqueous solution of magnesium chloride.

Chlorides are common constituents of cutting and cleaning compounds used during shop fabrication.

Water dissociation by nuclear phenomenon provides a source of oxygen.

Against - Micro-examination of a failed rivet showed mainly intergranular type grain separations and no branch type cracking; however, cracks of this description have been induced by stress corrosion in austenitic and semi-austenitic stainless steels and are reported in the literature.^(2,3)

The presence of chlorides and other contaminants in EBWR reactor water are closely controlled.

Conclusions

The facts developed to date are not sufficient to explain the cause of EBWR control rod rivet failures. Further information is required and is being obtained.

Rivet specimens have been removed from No. 4 control rod transitions of high radioactivity (piece No. 9 from control rod No. 4, 400 γ /hr @ 1 in.) and low radioactivity (piece No. 19 from control rod No. 4, 6 γ /hr @ 1 in.). They are being ground and polished for microstructure comparison. It is hoped the results will provide some idea of relative degree of radiation damage.

It is anticipated that a second test, being prepared, will reveal the comparative effects of initial rivet condition: carbide precipitation, chloride stress corrosion, and, by inference, radiation damage. Two riveted Type 304 stainless steel coupons simulating control rod sandwich construction are being fabricated. Each coupon will be assembled with "hard" and "soft" sample rivets; and, the "hard" sample rivets will be solution annealed for measurement of least degree of carbide precipitation.

One coupon will be degreased in a solution containing trichlorethylene. The second coupon will not be degreased. Both coupons will be autoclaved in water at 500°F and removed periodically for examination.

CRUD DEPOSITION

Control Rod Appearance as Viewed through a 3-Ft Water Depth

1. Hafnium Poison Sections — Rods Nos. 1, 3, 5, and 7

Blue-grey background with a pink tinted overlay. The pink color increases slightly in density from the retrieving end down to the transition with Zircaloy-2.

Hafnium Poison Sections — Rod No. 9

Same as above except the pink tint is denser and changes to a brown color about one foot above the transition with Zircaloy-2.

2. 2% Boron-Stainless Steel Poison Sections — Rods Nos. 2, 4, and 6

Similar to the hafnium poison section (rods Nos. 1, 3, 5, and 7), except for intermittent deposits of red-brown corrosion products at the blade edges and in the crevices between blades.

2% Boron-Stainless Steel Poison Sections — Rod No. 8

Same as above, except the pink tint is denser and changes to a brown color at the transition with Zircaloy-2.

3. Zircaloy-2 Followers (10 in. and 3 in.) — Rods Nos. 1-7

Grey background with a pink tinted overlay. The pink color increases slightly in density from the poison section transition down to a position along the 3-in. section where the color changes to brown. Spotwelds appear as black circles.

Zircaloy-2 Followers (10 in. and 3 in.) — Rod No. 8

Grey background with a brown overlay changing to darker brown at the transition to the 3-in. section. Spotwelds appear as solid grey circles.

Zircaloy-2 Followers (10 in. and 3 in.) — Rod No. 9

Dark brown all over. Spotwelds cannot be distinguished.

4. Type 304 Stainless Steel Followers — All Rods

Brown overlay from the Zircaloy-2 transition ending abruptly over a metallic brown tarnish film. The tarnish film extends down another 3 in. (avg.) and can also be seen underlying the brown overlay where the brown has rubbed thin against the control rod guides. The rods are a dull silver from the tarnish film to the latching end except for rods Nos. 1 and 3 which are bright and shiny.

The distances from the latching end to the start of tarnish and brown overlay were measured (see Table 121B-III).

Table 121B-III
FOLLOWER EXAMINATION

Control Rod	Start of Tarnish, in.	Start of Brown, in.	Difference, in.
1	33½	37½	4
2	30	- -	-
3	35½	38	2½
4	33	37	4
5	37	39½	2½
6	31½	33	1½
7	32	35	3
8	34	36	2
9	- -	38	-
Avg	33	36	3

Discussion

The blue-grey and grey background appearances of the hafnium, 2% boron-stainless steel, and Zircaloy-2 are protective oxide film. The pink tint indicates a thin crud deposit which changes color to brown, then dark brown as its thickness increases.

Control rods Nos. 1 through 8 reveal an increase in crud thickness from their retrieving ends to their 3-in. Zircaloy-2 sections. A similar gradation is observed on the hafnium section of No. 9 rod, but deposition on its 10-in. Zircaloy-2 section becomes so dense that no further gradation can be discerned. Neither can the amount of crud on the stainless steel followers be compared with that on adjacent 3-in. Zircaloy-2, due to the relatively shiny background of the stainless steel.

Crud deposition is greatest on the No. 9 (center) rod, intermediate on the No. 8 (southwest corner) rod, and relatively light on all other rods. Photographs of the 10-in. Zircaloy-2 portions of the Nos. 4 and 9 rods (Figs. 121B-4 and 5 respectively) reveal the extremes of deposition. Note the clarity of the spotwelds on No. 4 rod. The fact that No. 4 (boron) rod is thicker, 1/4 in. versus 1/8 in., is not significant since the spotwelds on all hafnium rods other than No. 9 can be seen as easily.

Fig. 121B-4. Zircaloy-2 Section (10 in.), No. 4 EBWR Control Rod

Fig. 121B-5. Zircaloy-2 Section (10 in.) No. 9 EBWR Control Rod

During reactor operation, totaling 8800 Mwd, no control rod was withdrawn more than 36 in. for any significant length of time, except boron corner rods Nos. 2, 4, 6, and 8 which were banked at 47 in. for the first 1000 Mwd; however, the integrated average position for these rods was also below 36 in. Figure 121B-6 relates the elevations at which tarnish and crud begin on the stainless steel followers to the reactor vessel and fixed components for 36-in. rod withdrawal. Figure 121B-7 shows the tarnish boundary on No. 2 rod; the crud boundary does not appear distinctly on the photograph.

Fig. 121B-6. Relative Location of Control Rod and Fixed Reactor Components for 36-in. Control Rod Withdrawal (4-ft Core)

Fig. 121B-7. Type 304 Stainless Steel Tail Section. No. 2 EBWR Control Rod

The tarnish and crud boundaries are within the control rod thimbles for full rod withdrawal and well within the thimbles for 36-in. withdrawal. The thimble environment is low in nuclear flux, restricted in circulation, and of lower than saturated water temperature due to the influx of control rod drive seal cooling water.

Conclusions

The heavy crud deposit on No. 9 rod correlates with the high nuclear flux in the center core region. Whether an intermediate flux existed in the southwest core quadrant, to account for the intermediate crud deposit on No. 8 rod, is not known. Whether nuclear flux enhances crud deposition, by direct phenomenon or secondary heating effects, is not revealed by these examinations.

The goodly amount of crud on the stainless steel followers in a region of low nuclear flux indicates that there is a mechanism for crud deposition on vertical surfaces in the EBWR reactor vessel not dependent on nuclear phenomenon. Stainless steel follower crud seems peculiarly related to its tarnish boundary; none below, all beginning 1-1/2 in. to 4 in. above, where control rod thimble ambient temperature would be slightly higher.

Development of tarnish is dependent, among other things, on temperature and time at temperature. Crud deposition may be similarly dependent, with a threshold temperature something above that for tarnish, but it is probably most dependent on tarnish (incipient corrosion) to gain a foothold on these vertical stainless steel surfaces.

REFERENCES

- (1) EBWR Test Report No. 121A.
- (2) E. E. Denhard, Jr., "Effect of Composition and Heat Treatment on the Stress Corrosion Cracking of Austenitic Stainless Steels," Corrosion, Vol. 16 (July, 1960), p. 259t.
- (3) E. H. Phelps and A. W. Loginow, "Stress Corrosion of Steels for Aircraft and Missiles," Corrosion, Vol. 16 (July, 1960), p. 325t.

INVESTIGATION OF IRRADIATED EBWR CONTROL RODS

(Test Report No. 121C, dated March 21, 1961, by T. L. Kettles)

The condition of riveted transitions and the extent of crud deposition as revealed by visual examination of control rods in the EBWR fuel storage pit is described in Reference 1. During that examination, information was also obtained regarding condition of materials and relative radioactivity.

TYPE 304 STAINLESS STEEL

The Type 304 stainless steel section of each control rod is a 3-in. cruciform, 68-1/2 in. long, that terminates with a forged and machined notched connector, 3 in. long. The sections incorporated into hafnium control rods consist of four 1/16-in. x 1-1/2-in. x 1-1/2-in. angles spotwelded together, and those incorporated into the boron control rods include, in addition, 1/8-in. filler pieces.

The upper portions of all Type 304 stainless steel sections were observed to be tarnished. The lower portions, were dull silver in appearance caused by a thin light milky film except for rods Nos. 1 and 3 which were bright and shiny.

The spotwelds employed to assemble the angles and filler plates (where used) and the connector butt weld appeared sound. The material acted in a ductile manner when sawed⁽²⁾ and when its transition to Zircaloy-2 was given the "wedge test."⁽¹⁾

There were minor abrasion marks on the angle surfaces where they had rubbed along the guide structure within the thimbles. There were minor wear marks on the connector-notch faces caused by the action of mating control rod drive, Type 309 stainless steel, connectors during normal operation and/or scram. There were minor deposits of red corrosion products in the connector-notch corners; the source of these products has not been determined.

ZIRCALOY-2

The Zircaloy-2 section of each control rod is a 10-in. cruciform, 39 in. long (mean avg.), spotwelded to a 3-in. cruciform, 22 in. long. The sections incorporated into hafnium control rods consist of four 1/16-in. thick angles spotwelded together, and those incorporated into boron control rods include, in addition, 1/8-in. thick filler pieces.

The material was covered with a tightly adherent oxide film, grey in color. No white corrosion products were observed. The spotwelds employed to assemble the angles and filler plates (where used) appeared sound. The material acted in a ductile manner when sawed⁽²⁾ and when its transition with Type 304 stainless steel was given the "wedge test."⁽¹⁾ There are minor abrasion marks on the angle surfaces where they had rubbed along the guide plates of the core shroud structure.

Failure of the spotwelds between the Zircaloy-2 and the hafnium sections of No. 9 rod was attributed to the excessive dynamic forces imposed by the oscillation tests which were performed to obtain power transfer functions.⁽²⁾ The other hafnium control rods (Nos. 1, 3, 5, and 7) showed no sign of distress at this transition.

HAFNIUM

The hafnium sections of control rods Nos. 1, 3, 5, 7, and 9 are 10-in. cruciforms, 43-1/2 in. long. They consist of four 1/16-in. x 5-in. x 5-in. angles spotwelded together.

Examination

The material was covered with a tightly adherent oxide film, blue grey in color. No white corrosion products were observed. The spotwelds appeared sound. There were minor abrasion marks on the angle surfaces where they had rubbed along the guide plates of the core shroud structure.

Three cracks were discovered; one each in rods Nos. 3, 5, and 7. The crack in rod No. 7, which is typical, is shown in Fig. 121C-1. The cracks occurred in the 1/2-in. radius, 90-degree angle bend, and extend from the Zircaloy-2 transition towards the retrieving tip (a distance ranging from 11 in. on rod No. 5 to 14-1/2 in. on rod No. 3.) The crack in rod No. 3 is about 1/32 in. open at its widest point and the ones in rods Nos. 5 and 7 about 1/8 in.

Fig. 121C-1. Hafnium Section, No. 7 EBWR Control Rod

The hafnium was observed to be brittle when control rod No. 9 and a portion of control rod No. 1 were sawed into short pieces.⁽²⁾ At times the air powered reciprocating saw snagged the hafnium sheet angle cruciform and stalled the air motor. On two occasions, however, the remaining portion of the piece being cut snapped off when the saw snagged.

Discussion

The original hafnium sheet material was contaminated when it was converted from iodide crystal bar form. The sheet could not be bent cold, more than 15 degrees, without cracking so it had to be bent hot. These facts

are discussed in Reference 3. Material of such low ductility would retain high stresses in the angle bends where the cracks occurred.

All control rods were proof tested when originally installed in the reactor vessel. Among other things they were scrambled a number of times when their temperature was not over 70°F. This type of test imposed high shock forces on the control rods when the hafnium material was most amenable to notch propagation.

The crack in one of the control rods (thought to be No. 5) existed when the rod was installed in the reactor. Persons who had seen that crack, advise it was about 1/8-in. maximum width and about a foot long. This description matches the present appearance so it is apparent the crack did not propagate more than a few inches, if at all, during the period it was operated in the reactor.

Control rod No. 9 was so severely punished by oscillation tests that its spotwelded hafnium to Zircaloy-2 transition failed. Control rod No. 9 was located at the core center in the region of highest nuclear flux. However, this rod was not one of those in which cracks were discovered.

Conclusions

The cracks in the hafnium sections of EBWR control rods are attributed to extreme residual stresses developed when poor quality sheet material was bent through 90 degrees. The hafnium, although brittle, resisted notch propagation under EBWR conditions. Nuclear radiation does not appear to have seriously affected hafnium mechanical properties.

Future work in building D-301 caves will provide quantitative data concerning the bending properties of the irradiated hafnium. This information is expected to qualify the degree of radiation damage.

2% BORON-STAINLESS STEEL

The 2% boron-(Type 304) stainless steel sections of control rods Nos. 2, 4, 6, and 8 are 10-in. cruciforms, 46 in. long. They consist of four 1/8-in. x 5-in. x 5-in. angles spotwelded together.

Examination

The material was covered with a tightly adherent oxide film, blue-grey in color. There were minor abrasion marks on the angle surfaces where they had rubbed along the guide plates of the core shroud structure. Some minor red-brown corrosion products occurred along the angle edges and in the crevices between angles. They are attributed to the corrosion of

carbon steel particulate matter imbedded in the 2% boron-stainless steel material when fabricated and/or handled.

Cracks were observed at 60% of the spotwelds used to assemble the angles into cruciform configuration. The cracks were generally arc-shaped, at an average radius of 1/4 in. from the spotweld centers, and subtended through angles from 160 to 200 degrees. Some cracks in the outer rows of spotwelds, were only slightly curved and extended to the edge of the angles. No cracks were observed in the 1/2-in. radius, 90-degree angle bends, except at the extreme top, adjacent to the weld, that connects the retrieving tip. All cracks observed occurred in the vicinity of a weld.

Figure 121C-2 shows the top of control rod No. 4 (piece No. 1, control rod No. 4, Fig. 121A-2, Reference 2), which reveals the different types of cracks. In the figure, the cracks have been marked with arrows to distinguish them from surface scratches. The bottom two cracks, and the crack on the left, are typical arc-shaped cracks. The two cracks marked with double arrows start to be arc-shaped but extend to the angle top edge; they are also typical of those that occur along the angle side edges (not shown in this figure). As shown, the top center crack is the type observed in the angle bend only at the retrieving tip weld.

Fig. 121C-2. Retrieving End, No. 4 EBWR Control Rod

Figure 121C-3 shows the 2% boron-stainless steel section of control rod No. 2 as viewed through a 3-ft water depth. Its appearance is typical of all boron poison sections although the photograph does not reveal every crack. However, many of the arc-type and some of the edge-type can be distinguished. The black tail-like marks which extend from arc-shaped cracks

and some of the black transverse lines at angle edges are surface marks produced by local disturbances of circulating reactor water.

Fig. 121C-3. 2% Boron-Stainless Steel Section, No. 2 EBWR Control Rod

Fig. 121C-4. Schematic of Clamping Arrangement Used for Sawing Irradiated EBWR Control Rods

a more precise set of measurements was obtained when the pieces were examined in the fuel storage pit.

A 5-r ionization type meter (Juno) was used to measure gamma activity through 12-in. and 6-in. water shielding; and, a 500-r ionization type meter (Jordan) was used to measure activity in air at 1 in. All four cruciform quadrants were measured in air.

The measurements listed in Table 121C-I were obtained eighteen months after reactor shutdown. The piece numbers listed in the table are identified on Fig. 121A-2 of Reference 2. All data is in r/hr.

Table 121C-I

REACTIVITY MEASUREMENTS 18 MONTHS AFTER REACTOR SHUTDOWN

Piece No.	Control Rod No. 4			Control Rod No. 9		
	Water, 12 in.	Water, 6 in.	Air, 1 in.	Water, 12 in.	Water, 6 in.	Air, 1 in.
1	—	—	0.038	—	—	0.135
2	0.009	0.011	0.045/0.050	0.002	0.003	0.025/0.030
3	0.010	0.013	0.100/0.125	0.003	0.004	0.040
4	0.012	0.019	0.400	0.003	0.005	0.080
5	0.025	0.060	3.0/4.0	0.004	0.012	0.400/0.425
6	0.085	0.350	15/18	0.010	0.048	3.0
7	0.300	1.1	50/60	0.050	0.230	11/12
8	0.900	3.8	100/110	0.200	1.0	35/40
9	2.8	7.0	400*	0.100	0.480	30
10	0.140	0.500	30/35	0.210	0.900	50
11	0.195	0.900	50/55	0.270	1.3	65
12	0.240	0.950	50/60	0.280	1.2	70
13	0.245	1.0	50/60	0.230	1.0	50
14	0.200	0.900	45/50	0.180	0.800	40
15	0.125	0.450	28/35	0.110	0.500	30
16	0.050	0.250	12	0.040	0.210	9.0
17	0.017	0.080	5.0	0.012	0.060	3.0
18	0.031	0.130	5.0	0.004	0.016	0.375
19	0.028	0.140	6.0	0.010	0.055	2.5

* Only one quadrant was measured due to the high activity.

Radioactivities of control rod quadrants are measures of relative nuclear core flux opposite the quadrants. Pieces Nos. 10 through 13 were within the active core during all periods of EBWR operation. Their activity measurements indicate nuclear flux was uniform at the core center and varied up to 17% around control rod No. 4, in the northeast corner.

The average radioactivity measurements in air of the 10-in. cruciform pieces were plotted on Fig. 121C-5. The graph reveals that activity along the Zircaloy-2 sections of control rods Nos. 4 and 9 is comparable in

The cracks were observed to have no tendency to propagate, when control rod No. 4 was sawed into pieces in the EBWR fuel storage pit water, at a temperature not exceeding 70°F. The rod was held rigid, for sawing, by two opposing hydraulic rams as shown schematically in Fig. 121C-4. A load of about four tons was required to bend the angles slightly, as necessary, to seat them properly in the vise jaw corners. It would be difficult to devise a better method to test the tendency toward crack propagation than was obtained incidental to the sawing operation.

Discussion

The 2% boron-stainless steel sheets used to fabricate the original EBWR control rods were produced from vacuum melted ingots and sheared to size. They were bent cold into angular form over a 16-gage aluminum sheet laid on the female die to prevent cracking. The angles were not preheated when spotwelded together and the assembled cruciforms were not stress relief annealed.

Two 2% boron-stainless steel cruciforms were made to develop fabrication procedures. One has never been used, except to develop techniques for sawing irradiated control rods, and after five years, at room temperatures, it shows no sign of cracking.

The second cruciform was incorporated into a control rod which was then installed in an autoclave for the purpose of testing control rod drives. It was operated many hours in water at 600 psig and 488°F and was scrambled thousands of times before removal in February, 1961, for examination. Cracks were observed at spotwelds in the longitudinal rows nearest the angle bend. They were arc-shaped, but at an average radius of 3/16 in. from spotweld centers versus 1/4 in. for similar cracks in the irradiated control rods.

Conclusions

The cracked appearance of the 2% boron-stainless steel sections of the original EBWR control rods belies their strength and resistance to crack propagation. The cracks developed in order to relieve extreme stresses at welds, and may have been initiated by unequal thermal expansion effects.

The cracks and adjacent material structure will be examined in building D-301 caves. Material bending properties will also be compiled to provide a measure of radiation damage.

RADIOACTIVITY

The radioactivity of each piece of control rods Nos. 4 and 9 was measured when sawed and listed on Fig. 121A-2 of Reference 2. The measurements were obtained incidental to the sawing operation, therefore,

magnitude and distribution. The abrupt increase, at the Zircaloy-2 to 2% boron-stainless steel transition of No. 4 control rod, from less than 30 r/hr to 400 r/hr, is caused by a change of radionuclides. The radioactivity at the top 2 ft of hafnium and 2% boron-stainless steel sections is low because those portions of the poison sections were relatively far from the active core during all periods of EBWR operation.

The radionuclides responsible for long lived activity of control rod materials will be determined by future work in building D-301 caves.

Fig. 121C-5. Radioactivity of Original EBWR Control Rods (Nos. 4 and 9) after 18-Month Decay

RADIONUCLIDES

Crud

Radiochemical analysis of crud that had deposited at the bottom of control rod No. 9 during reactor operations ⁽²⁾ has been completed. The major activity after a sixteen-month decay is from the radionuclide Co^{60} (92%) with only minor contributions from fission products (less than 0.2%). The mechanism primarily responsible for the Co^{60} activity was, most probably, a neutron-proton reaction with Ni^{60} .

The crud sample was representative of material that had circulated through the core region during reactor operation. The results of the analysis, although not indicative of short lived radionuclides, are listed in Table 121C-II.

Table 121C-II

RADIOCHEMICAL ANALYSIS DATA

Isotope	d/sec/gm 11/17/60	% Total Activity 11/17/60	d/sec/gm 7/15/59
Mn ⁵⁴	3.60×10^4	0.925	1.07×10^5
Ni ⁶³	8.33×10^3	0.214	8.42×10^3
Co ⁵⁸	2.41×10^5	6.19	2.35×10^7
Co ⁶⁰	3.59×10^6	92.24	4.28×10^6
Zn ⁶⁵	1.08×10^4	0.277	4.18×10^4
Cs ¹³⁷	0.28×10^2	0.0007	0.28×10^2
Ce ¹⁴⁴	0.65×10^2	0.0017	1.30×10^2
*Sb	1.85×10^3	0.0474	---
*Sr	0.19×10^2	0.0005	---
*Zr	3.03×10^3	0.0778	---
*Te	1.15×10^3	0.0295	---
Cr ⁵¹	0	0	---
Fe ⁵⁹	0	0	---
Cd ¹¹³	0	0	---
Cd ¹¹⁵	0	0	---
Ba ¹⁴⁰ -La ¹⁴⁰	0	0	---

*Specific isotopes of Sb, Sr, Zr, and Te were not determined owing to their low activity.

The connector end of control rod No. 9 measured 40 mr/hr [hard (H)] and hard and soft (H and S) @ 2 in. before the crud, used as the radio-analysis sample, was removed from its hollow stem. It measured 4 mr/hr (H) and (H and S) @ 2 in. after the crud had been removed. The crud measured 75 mr/hr (H) and 125 mr/hr (H and S) @ 2 in. The connector ends of all control rods were measured to provide a relative indication of the crud amounts deposited and the data listed in Table 121C-III.

The measurements indicated that there was a large amount of crud deposited at the bottom of control rod No. 7 but little or none at the bottom of control rod No. 6. It is postulated that the sandwich construction of control rod No. 6 may have been fabricated tighter than all others, sufficient to prevent entrance of particulate matter.

Table 121C-III

ENTRAPPED CRUD ACTIVITY

<u>Control Rod No.</u>	<u>Radioactivity, mr/hr (H) @ 2 in.</u>
1	55
2	85
3	50
4	150*
5	60
6	5
7	125
8	45
9	40

* The hollow stem of this connector contained many saw chips in addition to crud.

Antimony

The antimony detected by crud radioanalysis is believed to be a fission product. However, since an antimony-beryllium source was used in EBWR the possibility of antimony leakage from the source has to be considered.

The antimony portion of the source was encapsulated in a stainless steel cylinder and lowered into the beryllium portion by means of a stainless steel wire connected to the top of the cylinder. The wire was then clipped off adjacent to the cylinder when EBWR went into operation and the remaining piece of wire was removed after EBWR was shut down in June, 1959.

The piece of wire would probably have antimony on its surface if the cylinder had leaked so it was qualitatively analyzed by radiochemical techniques. No radioantimony was detected.

Inconel

A piece of Inconel gusset removed from the retrieving tip end of control rod No. 9 was analyzed by radiochemical techniques. The purpose was to compare radionuclides in the crud sample with those deposited on the surface of a core component, and to determine whether the latter radionuclides had also been induced in the Inconel material proper. The results are tabulated below with crud data added for comparison.

Table 121C-IV

COMPARISON OF RADIONUCLIDES IN ENTRAPPED AND SURFACE CRUD

<u>Isotope</u>	<u>Crud</u>	<u>Inconel Surface</u>	<u>Inconel Material</u>
Mn ⁵⁴	Yes	No	No
Ni ⁶³	Yes	Trace	Trace
Co ⁵⁸	Yes	Yes	Yes
Fe ⁵⁹	No	Yes	Yes
Co ⁶⁰	Yes	Yes	Yes
Zn ⁶⁵	Yes	No	—
Cs ¹³⁷	Yes	No	—
Ce ¹⁴⁴	Yes	No	—
Sb	Yes	No	—
Sr	Yes	No	—
Zr	Yes	No	—
Te	Yes	No	—

There is no apparent reason why the radionuclide Fe⁵⁹ was detected on the surface of the Inconel gusset but not in the crud sample.

Zircaloy-2; Type 304 Stainless Steel

The original EBWR control rods were sawed into 5-ft lengths to facilitate removal from the reactor vessel. One cut was made through the 10-in. Zircaloy-2 cruciform and the other through the 3-in. type 304 stainless steel cruciform. Some of the saw chips dropped down the hollow control rod stems to the connector ends.

Those chips in the connector end of control rod No. 4 were removed and analyzed by radiochemical techniques for zirconium, tin, and cobalt. The radionuclides Zr⁹⁵, Sn¹¹³, Co⁵⁸, and Co⁶⁰ were detected.

CAVE WORK

Investigations are in progress in building D-301 caves to develop methods for machining specimens from the 6-in. long pieces of control rods Nos. 4 and 9 and to develop specimen test procedures.

PHOTOGRAPHS

Figures 121C-6 through 13 have been included as supplementary information.

Fig. 121C-6. Retrieving End, No. 9 EBWR Control Rod

Fig. 121C-7. Hafnium-Zircaloy-2 Transition, No. 1 EBWR Control Rod

Fig. 121C-8. Positive Action Handling Tool, View at Clamping End

Fig. 121C-9. Tube and Angle Rack Assembled in EBWR Fuel Storage Pit for Control Rod Examination

Fig. 121C-10. Hafnium Section, No. 9 EBWR Control Rod

Fig. 121C-11. 2% Boron-Stainless Steel Section, No. 4 EBWR Control Rod

Fig. 121C-12. Zircaloy-2 Sections (3 in.), Nos. 4 and 9 EBWR Control Rods

Fig. 121C-13. Portion of 2% Boron-Stainless Steel Section of EBWR Type Control Rod Used to Test Control Rod Drives

REFERENCES

- (1) EBWR Test Report No. 121B.
- (2) EBWR Test Report No. 121A.
- (3) "The EBWR - Experimental Boiling Water Reactor," ANL-5607 (May, 1957).

RADIOCHEMICAL ANALYSIS OF ORIGINAL EBWR CONTROL ROD AND FOLLOWER SAMPLES

(Test Report No. 121D, dated September 8, 1961, by R. J. Armani)

Eight samples taken from EBWR control rods Nos. 4 and 9 were submitted for radiochemical analysis. The analysis was made for activation products present in the samples, and quantitative chemical and counting techniques were made to determine the amount of activity of each isotope present. These samples were taken from control rods which were removed from EBWR after shutdown in July, 1959. Since about a two year decay has occurred, no effort was made to find any isotope whose half life was less than ~ 40 days. Table 121D-I identifies the sample and its location on the control rod, and Table 121D-II contains the results of the radiochemical analysis.

Table 121D-I

SAMPLE LOCATION

Control Rod No.	Material	Radial* Location	Identification†
4	Borated 304 SST	Near Center	3 9 C
4	Borated 304 SST	Near Extremity	3 9 C
4	Zircaloy-2	Near Center	5 11 E
4	Zircaloy-2	Near Extremity	5 11 E
9	Zircaloy-2	Near Center	6 1 C
9	Zircaloy-2	Near Extremity	6 1 C
9	Hafnium	Near Center	3 9 C
9	Hafnium	Near Extremity	3 9 C

* From center of control rod

† See Reference 1.

The presence of Sb^{125} is observed in the Zircaloy and hafnium samples. This could be a fission product, but it was not found to be in the borated stainless steel samples. Zircaloy-2 contains approximately 1.5% Sn, so it is possible that the Sb^{125} activity in the Zircaloy-2 samples is due to neutron capture in Sn^{124} and decaying to Sb^{125} by the reaction Sn^{124}

$(n,\gamma) \text{Sn}^{125} \xrightarrow[10 \text{ min, 10 days}]{\beta^-} \text{Sb}^{125}$. If trace amounts of Sn are present, this could also explain the small amount of Sb^{125} activity in the hafnium samples.

In the case of the hafnium samples, a large amount of hafnium activity was seen. The gamma spectrum showed the presence of Hf^{175} and Hf^{181} , but it was not possible to determine the relative amounts of each isotope.

In the past, large amounts of Co^{58} were found in EBWR crud and samples taken from inside the pressure vessel. Here, no Co^{58} could be seen in the gamma spectrum of the cobalt separation. This may be due to the long decay time involved since the reactor was shut down in July, 1959 and the samples analyzed in July, 1961.

REFERENCE

- (1) EBWR Test Report No. 121A.

Table 121D-II

RADIOCHEMICAL ANALYSIS (AID/SG - July 25, 1961)

Sample	Weight, gm	Sb ¹²⁵	As	Co ⁵⁸	Co ⁶⁰	Ni ⁶³	Cr ⁵¹	Mn ⁵⁴	Fe ⁵⁹	Zn ⁶⁵	Sn ¹¹³	Zr ⁹⁵	Hf ^{175,181}
Boron-Stainless Near Center*	0.8962	A	A	A	1.03×10^6	1.70×10^3	A	2.07×10^6	A	A	--	--	--
Boron-Stainless Near Extremity*	0.4698	A	A	A	1.44×10^6	4.07×10^3	A	2.20×10^6	A	A	--	--	--
Zr-2 No. 4 Near Center*	0.7232	2.10×10^7	A	A	5.67×10^5	1.20×10^3	A	--	A	-	4.30×10^5	2.27×10^5	--
Zr-2 No. 4 Near Extremity*	0.7681	2.07×10^7	A	A	6.65×10^5	6.55×10^2	A	--	A	-	3.78×10^5	1.98×10^5	--
Zr-2 No. 9 Near Center*	0.5057	1.57×10^7	A	A	1.17×10^6	1.85×10^3	A	--	A	-	5.25×10^5	3.45×10^5	--
Zr-2 No. 9 Near Extremity*	0.6014	3.52×10^7	A	A	7.37×10^5	1.29×10^3	A	--	A	-	4.13×10^5	3.20×10^5	--
Hafnium No. 9 Near Center*	9.5656	2.93×10^4	A	A	4.60×10^3	13.4	A	--	A	-	--	--	YES
Hafnium No. 9 Near Extremity*	6.4000	4.23×10^4	A	A	1.60×10^4	22.3	A	--		-	--	--	YES

A - Isotope is absent.

-- - No analysis made for this isotope.

YES - Activity is present, but unable to separate isotopes.

* - Radial location from center of control rod.

BORON BURNUP IN ORIGINAL EBWR CONTROL ROD NO. 4

(Test Report No. 121E, dated September 21, 1961, by R. J. Armani)

After 8800 Mwd of operation, the four boron-stainless steel and five hafnium control rods were removed from EBWR and subjected to various chemical and physical tests. These rods had all been operated in a bank, and Fig. 121E-1 shows the relative positions of the rods to each other. Rods Nos. 1, 3, 5, 6, and 9 are hafnium, and the remainder are boron-stainless steel.

Fig. 121E-1. Relative Control Rod Positions in the Core of EBWR and Typical Radial Locations of Samples

Three specimens were cut from the original control rod No. 4 and sent to the Chemistry Division for analysis in order to determine the amount of boron burnup that had taken place (Table 121E-I). Two of the specimens (41C near center and 41C near extremity) were taken from positions on the control rod which were in the core most of the time while the reactor was at power. The third specimen (010e near center) spent no time in the core when the reactor was at power.

Table 121E-I

SAMPLE ANALYSIS

Sample	Radial* Location	Identification†	% Boron	Li ppm
1	Near Center	4 1 c	2.07	77
2	Near Extremity	4 1 c	2.02	102
3	Near Center	0 10 e	2.05	1

* From center of control rod (Fig. 121E-1)

† See Reference 1

The results of the analysis are given in Table 121E-I. The table also contains the results of a lithium analysis. A lithium atom is formed when a boron atom interacts with a neutron in the reaction $B^{10}(n, \alpha)Li^7$, so the lithium analysis was performed to verify the results of the boron analysis.

The boron content of the control rod before irradiation is known to be 1.9-2.09%. The analysis after irradiation is shown in Table 121E-I to be a little more than 2% in all three specimens. The lithium analysis shows that only very small amounts of lithium are present. The indications are that little burnup has taken place in these specimens.

The following is a brief description of the method used to perform the boron and lithium analysis. Approximately one gram of the sample is weighed and dissolve in perchloric acid. The solution is filtered and placed in an instrument called a Dyna-Cath, which removes all the constituents of the steel by an amalgamation process and leaves the boron and lithium behind as a clear solution. The pH is adjusted to 6.7, and the solution is divided into two parts. A boron analysis is performed on one part by titration with sodium hydroxide, and a lithium analysis is performed on the other by the use of a Flame Photometer. The analysts feel they achieve an accuracy of 1% or better on the boron analysis and 2-5% on the lithium analysis.

REFERENCE

- (1) EBWR Test Report No. 121A.

RADIOCHEMICAL ANALYSIS OF CRUD REMOVED FROM THE BOTTOM OF EBWR CONTROL RODS NO. 4 AND NO. 9

(Test Report No. 112, dated November 30, 1960, by R. J. Armani)

A crud sample was submitted for radiochemical analysis. The crud was removed from inside the bottom of EBWR control rods No. 4 and No. 9.

The sample was put into solution and an analysis was made for isotopes that might be present, due to neutron activation, and for some of the fission products with appreciable half lives. Table 112-I lists the activities which were determined and the per cent of the total activity for each present on November 17, 1960.

Table 112-I

CRUD SAMPLE ANALYSIS

Isotope	d/s/gm on 11/17/60	% of total activity 11/17/60	d/s/gm 7/15/59
Sb	1.85×10^3	0.0474	--
Mn ⁵⁴	3.60×10^4	0.925	1.07×10^5
Ni ⁶³	8.33×10^3	0.214	8.42×10^3
Cr ⁵¹	0	0	0
Co ⁵⁸	2.41×10^5	6.19	2.35×10^7
Co ⁶⁰	3.59×10^6	92.24	4.28×10^6
Fe ⁵⁹	0	0	0
Zn ⁶⁵	1.08×10^4	0.277	4.18×10^4
Cd ¹¹³	0	0	0
Cd ¹¹⁵	0	0	0
Cs ¹³⁷	28.4	7.3×10^{-4}	28.4
Ba ¹⁴⁰ -La ¹⁴⁰	0	0	0
Sr	19.6	5.3×10^{-4}	--
Ce ¹⁴⁴	65.1	1.67×10^{-3}	1.30×10^2
Zr	3.03×10^3	0.0778	--
Te	1.15×10^3	0.0295	--

Where it was possible, specific isotopes were identified. From Table 112-I, it is seen that it was not possible to separate the isotopes of Sb, Sr, Zr, and Te within themselves. The activities separated were very low, and it was not feasible to follow the long half lives. Since EBWR was shut down in July, 1959, decay corrections were made on the isotopes specifically identified and these are also found in Table 112-I. Unfortunately, a long time has elapsed since shutdown, and no short lived isotopes were detected. Therefore, it is not possible to determine all the activity that was present in July, 1959.

Table 112-I indicates that the major portion of the activity is due to activation of the crud and a very small portion due to fission products which were carried along with the crud. The major isotope of the relatively long lived elements seems to be Co⁵⁸.

DISPOSITION OF ORIGINAL EBWR CONTROL RODS

(Test Report No. 121F, dated October 27, 1961, by T. L. Kettles)

The 5-ft long sections of control rods Nos. 1, 2, 3, 5, 6, 7, and 8 were removed from the EBWR storage pit and transferred to Vault No. 2 of the outdoor ANL Radioactive Storage Facility. The storage rack containing 6-in. long sections of control rods Nos. 4 and 9(1) will remain in the EBWR storage pit.

PROCEDURE

The 5-ft long tail sections were transferred without special provisions, since maximum radioactivity measured 50 mr/hr(H) @ 2 in a 30-ft length of 1/16-in. diameter (7 x 19) stainless steel aircraft cable was clamped to one end of each intermediate and top section for handling. A brass tag was affixed to the loose end of each cable for identification.

Each section was lifted from the EBWR storage pit and laid on a stretcher placed on the operating floor. The section was then carried quickly to a Dempster-Dumpster located immediately outside the freight door. At the Radioactive Storage Facility the sections were carried by crane from the dumpster and lowered into Vault No. 2.

RADIOACTIVITY AND IDENTIFICATION

The radioactivity measurements and the identification placed on the brass tags are tabulated in Table 121F-I. Identification letters used in the table are "Hf" - hafnium; "B" - 2 wt-% boron-stainless steel, and "ZR" - Zircaloy-2.

Table 121F-I
ACTIVITIES OF CONTROL ROD SECTIONS

Control Rod Number	Section	Radioactivity, r/hr (H) @ 10 ft	Identification
1	Top ↓ Intermediate ↓	0.005	1 Hf
2		0.900	2 B
3		0.040	3 Hf
5		0.075	5 Hf
6		0.900	6 B
7		0.045	7 Hf
8		1.000	8 B
1		0.115	1 Zr
2	Intermediate ↓	0.195	2 Zr
3		0.100	3 Zr
5		0.125	5 Zr
6		0.160	6 Zr
7		0.120	7 Zr
8		0.220	8 Zr

HAZARDS

A tenacious mud on the surface of the control rod sections was the only loose (relatively) radioactive matter encountered; therefore, heavy shielded containers were dispensed with during transfer by taking full advantage of time and distance factors. It took five seconds to move each section by stretcher to the dumpster during which time maximum body level radiation was 6 r/hr(H). It took ten minutes to haul the dumpster to the storage area

during which time maximum radiation level in the truck cab was 85 mr/hr(H). Three trips were made carrying sections 1 Hf, 3 Hf, 5 Hf, 7 Hf, 2 Zr, 6 Zr, 8 Zr; sections 2 B, 1 Zr, 3 Zr, 7 Zr; and sections 6 B, 8 B, and 5 Zr.

The job was accomplished in one working day. The health physics representative absorbed a dose of 155 mr while all other workers absorbed 100 mr or less.

REFERENCE

- (1) EBWR Test Report No. 121A.

PARTIAL INVESTIGATION OF IRRADIATED EBWR CONTROL RODS

(Test Report No. 121G, dated January 28, 1963, by T. L. Kettles and N. Balai)

INTRODUCTION

Reference 1 described the methods used to cut the No. 4 boron (corner) and No. 9 hafnium (center) control rods (that had been used throughout the original EBWR operational period of 8800 Mwd) into 6-in. long pieces. References 2 and 3 described the physical condition of these rods as determined by examination of the separate pieces. This report presents data on the mechanical properties of the irradiated 2 w/o boron-stainless steel, hafnium, and Zircaloy-2 materials from the two control rods examined.

Figure 121G-1, originally issued with Reference 3 to show the variations in radioactivity along No. 4 and No. 9 rods, also identifies the pieces selected for preparation as physical test specimens. Pieces Nos. 2 and 8 from control rod No. 4 provided material specimens of 2 w/o boron-stainless steel that have the same reactor environmental exposure except for the amount of neutron radiation each absorbed. Pieces Nos. 2 and 7 from control rod No. 9 provided hafnium material for a similar study. In both cases, the relative changes in mechanical properties between the opposite ends of the same rod indicated the extent of radiation damage sustained.

Zircaloy-2 followers from both No. 4 and No. 9 control rods absorbed relatively uniform doses of radiation along their lengths. (However, the follower from No. 9 control rod had about twice the specific activity of No. 4 rod follower.) The most radioactive piece (No. 12) from each rod was selected for comparison.

SPECIMENS AND TEST PROCEDURES

Bend and hardness tests were adopted for evaluating changes in mechanical properties. A number of sheet tensile test specimen coupons

Fig. 121G-1. Radioactivity of Original EBWR Control Rods (Nos. 4 and 9) after 18-Month Decay (Location of Test Specimens Shown)

were prepared and tested but results were unsatisfactory; the specimens invariably broke outside the gage length. Longitudinal strips, 3/8-in. wide x 6-in. long, and transverse strips, 3/8-in. wide x 4-in. long, were cut from areas between spotwelds to provide flaw-free specimens. After their edges had been stoned, the strips were bent in a bend test jig shown in Fig. 121G-2. The extensometer recorded over the full 1-1/4-in. travel (110-degree bend) range and provided a trace of load versus deflection. The diameter of the loading pin and the support pins is 1/2-in. and the span between the support pins is 2-1/2 in.

The operation of the test jig using an unirradiated 2 w/o boron-stainless steel specimen is shown in Fig. 121G-3. Hafnium and Zircaloy-2 specimens that withstood the maximum 110-degree bend were transferred to a vise where the bend angle was increased to 180 degrees.

Material hardness was measured on the bend specimen ends and along the length of adjacent scrap pieces. The Rockwell A scale was selected to measure the hardness of all materials.

Gamma activity of the cruciform pieces and of each specimen were measured in air at 1 in. with a conventional "cutie-pie" area survey meter.

SUMMARY OF TEST RESULTS

2 w/o Boron-Stainless Steel

Cruciform piece No. 2 activity, the topmost poison section, was 50 mr hr; specimens cut from this section were called "cold." Cruciform

Fig. 121G-2. Bend Test Jig

Fig. 121G-3. 2 w/o Boron-Stainless Steel Specimen Undergoing Bend Test

piece No. 8 activity, the poison section bottom was 110 r/hr; specimens cut from this section were called "hot." A total of 24 cold and 26 hot specimens were bent at a load rate of 200 lb/min. Average values and scatter band for longitudinal and transverse specimens are plotted on Fig. 121G-4. Test data, with estimates of the proportional limit, for longitudinal and transverse specimens are given in Tables 121G-I and II respectively.

Fig. 121G-4. Static Bend Traces (2 w/o Boron Stainless Steel)
Load Rate - 200-lb Minimum

Figure 121G-4 (and Tables 121G-I and II) reveals large changes in the mechanical properties of the lower poison sections (1) the ultimate strength increased 20%, (2) the proportional limit (P.L.) doubled, (3) the ductility decreased by 60 , and (4) the modulus of elasticity apparently increased from 28×10^6 psi by about 15%. The anisotropic ratio (5:4) between longitudinal and transverse properties was maintained by the irradiation process.

Two hundred and five measurements on cold material and 193 on hot material revealed a moderate hardening of material. The change in the average hardness was from 59 to 67 on the Rockwell A scale.

Table 121G-I

2 w/o BORON-STAINLESS STEEL BEND TEST DATA FOR
LONGITUDINAL SPECIMENS (1/8 in. x 3/8 in. x 6 in.)

Cold Specimens			Hot Specimens		
Ultimate Load (lb)	P.L.* (psi)	Defl.† (in.)	Ultimate Load (lb)	P.L.* (psi)	Defl.† (in.)
354	71,000	0.67	455	155,000	0.23
322	59,000	0.74	471	147,000	0.31
361	77,000	0.67	357	150,000	0.11
367	78,000	0.65	454	157,000	0.28
383	79,000	0.60	463	153,000	0.27
299	66,000	0.55	431	159,000	0.25
312	66,000	0.68	434	143,000	0.28
292	67,000	0.53	408	136,000	0.30
-	--	--	382	143,000	0.16
-	--	--	406	147,000	0.23
-	--	--	412	151,000	0.23
Avg. 343	70,000	0.66	425	149,000	0.24

* Proportional limit

† Deflection

Table 121G-II

2 w/o BORON-STAINLESS STEEL BEND TEST DATA FOR
TRANSVERSE SPECIMENS (1/8 in. x 3/8 in. x 4 in.)

Cold Specimens			Hot Specimens		
Ultimate Load (lb)	P.L.* (psi)	Defl.† (in.)	Ultimate Load (lb)	P.L.* (psi)	Defl.† (in.)
298	53,000	0.64	404	111,000	0.44
319	49,000	0.74	373	128,000	0.24
312	51,000	0.70	395	132,000	0.31
298	45,000	0.66	322	98,000	0.39
284	52,000	0.66	375	105,000	0.31
354	61,000	0.83	350	127,000	0.19
352	56,000	0.78	342	106,000	0.31
350	67,000	0.77	350	119,000	0.26
324	65,000	0.81	345	103,000	0.37
318	57,000	0.76	348	109,000	0.30
309	58,000	0.80	413	140,000	0.21
303	55,000	0.77	423	128,000	0.38
309	59,000	0.67	351	132,000	0.18
320	58,000	0.89	382	113,000	0.40
300	57,000	0.84	355	119,000	0.31
304	58,000	0.86	-	--	--
Avg. 316	56,000	0.76	369	117,000	0.31

* Proportional limit

† Deflection

The area under the load-deflection trace for the hot material is less than for the cold material which indicates that the total energy to fracture was reduced. The most significant loss was in the plastic region, (to the right of the proportional limit ordinate) which is loss of ductility. This loss of ductility, accompanied by minor hardening, had no effect on the performance of either the rods or the reactor under the original severe performance requirements, i.e., reactor pressure-accelerated power scrams as well as cold reactor gravity rod-drops.

Furthermore, these tests confirm the rod sectioning observations, which revealed that the highly irradiated 2 w/o boron-stainless steel material was resistant to brittle fracture. As discussed in Reference 3 the arc-shaped cracks were found partially surrounding many of the spotwelds employed to assemble the angles into cruciform configurations. These cracks are sharp notches, yet, none of the cracks had propagated outside the spotweld heat-affected zones, although the control rods had been scrammed numerous times in cold (80°F) and hot (488°F) reactor operations.

Hafnium

Cruciform piece No. 2 activity, the topmost poison section, was 30 mr/hr; cruciform piece No. 8 activity, the poison section bottom, was 12 r/hr.

A total of 38 cold and 36 hot specimens were bent at a load rate of 200 lb/min. Only ultimate loads and maximum bend angles are tabulated in Tables 121G-III and IV; the proportional limit, from the load-deflection traces, could not be obtained from these 1/16-in. thick specimens.

The data of Tables 121G-III and IV revealed minor changes in strength and large losses of ductility. The ultimate strength of hot specimens increased by 10%. Hot longitudinal specimens fractured at an average deflection of 0.45-in. (about 130-degree included angle of bend) and hot transverse specimens fractured at an average deflection of 0.85-in. (about 90-degree included angle of bend). Average hardness, Rockwell A scale, of cold material (203 readings) was 60, and, 65 (220 readings) for hot material.

In the absence of load-deflection traces, no estimates of the significance of loss in ductility were possible. Hot specimen deflection data revealed that the hafnium was more anisotropic than the boron-stainless steel control rod poison material.

Table 121G-III

HAFNIUM BEND TEST DATA FOR LONGITUDINAL SPECIMENS
(1/16 in. x 3/8 in. x 6 in.)

Cold Specimens		Hot Specimens	
Ultimate (lb)	Deflection (Degrees)	Ultimate (lb)	Deflection (in.)
72	180	97	0.57
77	180	98	0.53
73	180	96	0.52
50	180	99	0.57
70	180	97	0.55
68	180	89	0.62
80	180	87	0.42
70	180	94	0.48
63	180	94	0.50
91	180	92	0.52
87	180	88	0.53
86	180	89	0.44
90	180	84	0.41
84	180	87	0.36
85	180	81	0.37
59	180	79	0.40
86	180	75	0.31
69	180	82	0.32
72	180	77	0.31
58	180	84	0.35
--	--	88	0.31
Avg.	75	88	0.45

Table 121G-IV

HAFNIUM BEND TEST DATA FOR TRANSVERSE SPECIMENS
(1/16 in. x 3/8 in. x 4 in.)

Cold Specimens		Hot Specimens	
Ultimate (lb)	Deflection (Degrees)	Ultimate (lb)	Deflection (in. or degrees)
55	180	72	0.82
58	180	77	0.75
53	180	74	1.1
54	180	73	*165 deg
51	180	74	1.2
55	180	64	0.68
56	180	68	0.94
58	180	66	0.77
66	180	70	0.57
70	180	68	1.2
68	180	65	0.44
69	180	61	0.84
69	180	69	*140 deg
71	180	63	0.46
69	180	68	0.72
69	180	65	0.58
67	180	70	0.78
58	180	--	--
64	180	69	0.85

*Considered as 1.2-in. deflection to obtain average.

Zircaloy-2

Cruciform piece No. 12 activity, from the center of No. 4 control rod follower section (1/4-in. thick cross section), was 60 r/hr. Cruciform piece No. 12 activity, from the center of No. 9 control rod follower (1/8-in. thick cross section), was 70 r/hr.

A total of 26 No. 4 and 36 No. 9 specimens were bent at a load rate of 200 lb/min. Only ultimate loads, maximum bend angles, hardness, and activity levels were measured. Breaking loads and activity levels are presented in Tables 121G-V and VI. The proportional limit could not be obtained for these 1/16-in. thick specimens.

Table 121G-V

ZIRCALOY-2 BEND TEST DATA FOR LONGITUDINAL SPECIMENS (1/16 in. x 3/8 in. x 6 in.)

Rod No. 4 Specimens		Rod No. 9 Specimens	
Ultimate (lb)	Activity* (Roentgens)	Ultimate (lb)	Activity* (Roentgens)
38	0.10	97	0.37
39	0.14	91	0.33
48	0.13	87	0.35
62	0.20	100	0.37
57	0.17	85	0.33
74	0.20	88	0.33
50	0.10	89	0.33
52	0.11	93	0.33
69	0.12	91	0.33
56	0.20	94	0.33
40	0.14	99	0.33
--	---	93	0.36
--	---	101	0.37
--	---	101	0.37
--	---	99	0.33
--	---	93	0.37
--	---	90	0.33
--	---	92	0.37
Avg. 53	0.15	94	0.35

*Activity levels reduced proportionately to 4-in. length for comparison with transverse specimens.

Table 121G-VI

ZIRCALOY-2 BEND TEST DATA FOR TRANSVERSE SPECIMENS
(1/16 in. x 3/8 in. x 4 in.)

Rod No. 4 Specimens		Rod No. 9 Specimens	
Ultimate (lb)	Activity (Roentgens)	Ultimate (lb)	Activity (Roentgens)
57	0.21	70	0.40
58	0.15	70	0.40
30	0.14	96	0.35
51	0.21	93	0.40
51	0.21	95	0.35
51	0.19	94	0.35
68	0.21	100	0.40
65	0.22	88	0.35
40	0.14	94	0.40
42	0.20	86	0.35
34	0.13	94	0.40
61	0.15	94	0.35
64	0.16	100	0.40
69	0.20	103	0.35
60	0.15	98	0.40
--	---	96	0.40
--	---	87	0.35
--	---	109	0.40
Avg. 53	0.18	93	0.38

The ultimate strength of material from the No. 9 rod, the most highly irradiated rod, was 75% greater than the material from the No. 4 rod, in both the longitudinal and transverse directions. All specimens could be bent 180 degrees without fracture so no comparison of ductility could be made. Hardness averaged 60 Rockwell A for material from both the No. 9 rod (206 readings) and the No. 4 rod (206 readings).

The large difference in strength without differences in hardness are not explainable, in the absence of material and fabrication histories for the Zircaloy-2 material used in the construction of the follower sections of the control rods. The tests, however, reveal qualitatively that ductility of Zircaloy-2 was retained after long exposure in the reactor environment.

DISCUSSION

The bend test proved satisfactory for evaluating changes in mechanical properties of the 1/8-in. thick 2 w/o boron-stainless steel. It was not satisfactory for the more ductile and thinner, 1/16-in. thick, hafnium and Zircaloy-2 specimens; the proportional limit could not be defined for either material and no comparison of ductility changes could be made for Zircaloy-2 at the two irradiation levels.

These deficiencies can be overcome, in future testing, by modifying the bend test jig design as follows:

1. Attachment of the loading bar to the universal testing machine head to eliminate the friction effects which masked the proportional limit in the 1/16-in. thick specimens.
2. Decrease the span between the support bars to assure fracture in the bend jig.

Control rod materials and configuration functioned satisfactorily under the severe operational requirements of the reactor facility which included demonstrations of reactor stability at power levels exceeding 60 Mw.

REFERENCES

- (1) EBWR Test Report No. 121A.
- (2) EBWR Test Report No. 121B.
- (3) EBWR Test Report No. 121C.

EFFECT OF U^{235} CONTAINING STAINLESS STEEL FOLLOWERS ON EBWR REACTIVITY AND CONTROL ROD WORTH

(Test Report No. 100C, dated May 11, 1960, by K. Almenas)

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 hafnium and a boron-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%.
2. The integrated worth of the center control rod follower is 0.29%. Extrapolating this to the 9-control rod follower bank gives about 2%.

3. The integrated worth of a Hf rod in the center is 2.02%.
4. The integrated worth of a boron-stainless steel rod in the center is 2.07%.

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 1/4 in. thick which have lapping edges bent at an angle and seam welded to form a cross 1/4 in. thick with 10-in. wide blades. Each plate contains 50 gm 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% ρ since for the predominating thin enriched (ET) elements the addition of control rod followers results in the given k^∞ :

$$k^\infty(ET) = 1.174$$

$$k^\infty(ET + \text{control followers}) = 1.147.$$

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% ρ .

CONTROL ROD WORTH

A 2-dimension PDQ calculation simulating these conditions was performed giving a theoretical reactivity change of 1.42%. 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 in. 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 (hafnium and boron-stainless steel), 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 follow is 0.29% ρ over the whole 48 in. of rod travel and 0.192% 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:

$$(0.29) \left(\frac{1.4}{0.192} \right) = 2.11\% \rho$$

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.

The basis of comparison is the operating reactivity of core load 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.

Fig. 100C-1. Differential Worth of Center Rod (Based on $\beta = 6.27 \times 10^{-3}$)

This amounts to an excess reactivity of:

$$\begin{aligned}\text{reactivity in Xe} &= (1.18)(0.87) = 1.027\% \rho \\ \text{reactivity in voids} &= (0.66)(0.87) = \underline{0.574}\end{aligned}$$

$$\text{Total} = 1.60\% \rho$$

where the 0.87 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 Mwd burnup and 467°F. Conditions of the critical were:

$$\begin{aligned}\text{all rods at 21.52 in.} &= (1.3)(0.87)(2.82) = 3.19\% \rho \\ \text{reactivity in period} &= (0.16)(0.83)(0.87) = 0.115\% \rho \\ \text{reactivity in Xe} &= \underline{0.024\% \rho}\end{aligned}$$

$$\text{Total} = 3.329\% \rho$$

where

1.3 is the rod temperature coefficient
2.82 is obtained from the rod calculation curve
0.83 is the Pu factor.

This reactivity must be corrected for the 355-Mwd burnup and the 21°F temperature difference from operating temperature.

$$\text{Burnup correction} = \left[-0.36\% \frac{\% \rho}{1000 \text{ Mwd}} \right] 355 = -0.128\% \rho .$$

$$\text{Temp. correction} = \left[-0.0105 \frac{\% \rho}{^\circ \text{F}} \right] 21^\circ \text{F} = -0.22\% \rho .$$

Therefore the total corrected reactivity equals 2.98% ρ , giving an operating follower worth of $2.98 - 1.60 = 1.38\% \rho$.

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:

$$\begin{aligned}\text{reactivity in Xe} &= (2.1)(0.87) = 1.827 \\ \text{reactivity in voids} &= (2)(0.87) = 1.74 \\ \text{reactivity in No. 9 rod} &= 0.31 \text{ (from Fig. 100C-1)}\end{aligned}$$

$$\text{Total} = \underline{3.877\% \rho}$$

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% ρ for a rod position of 26 to 32 in. dial.

HAFNIUM VS BORON-STAINLESS STEEL ROD

A relatively "clean" experimental comparison of the two types of control rods was obtained. Experimental conditions are shown in Table 100C-I.

Previous to the experiment the rod bank indicated a reactivity of 3.24% in the reactor. Since H_3BO_3 is worth 1.23%/gm/gal, the total worth of the central rods can be obtained as follows:

Worth of rod = 3.24% - worth of H_3BO_3 - reactivity in period

Worth Hf rod = 3.24% - 1.064 - 0.154 = 2.02%

Worth boron-stainless rod = 3.24 - 1.042 - 0.123 = 2.075%.

Table 100C-I

EXPERIMENTAL CONDITIONS

Hafnium Rod	Boron-Stainless Steel Rod
Critical conditions: No. 9 rod (Hf) all in Rods Nos. 1-8 all at 42.5 in. Period conditions: No. 9 rod all in Rods Nos. 1-8 at 48 in. (all out) Reactor on a 24-sec period. Poison conc. 0.865 gm H_3BO_3 /gal in both cases	Critical conditions: No. 9 rod (boron-stainless steel) all in Rods Nos. 1-8 at 43.4 in. Period conditions: No. 9 rod @ 4 in. Rods Nos. 1-8 all out Reactor on a 35-sec period Poison conc. 0.847 gm H_3BO_3 /gal in both cases

The slight experimental edge of the boron-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 boron-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 k_{∞} of 0.9902

and 0.9967 was obtained for the boron-stainless steel and hafnium rod containing cells respectively. This gives a factor of about 1.0065 to the boron-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 Fig. 100C-2. The data was obtained with two different control rods in the center, and as can be seen that points for the boron-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 in. 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. Differential and Integral Rod Worth for 9-Rod Bank (Based on $\beta = 6.27 \times 10^{-3}$)

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 hafnium and a boron-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%.
2. The integrated worth of the center control rod follower is 0.29%. Extrapolating this to the 9-control rod follower bank gives about 2%.
3. The integrated worth of a hafnium rod in the center is 2.02%.
4. The integrated worth of a boron-stainless steel rod in the center is 2.07%.

CALIBRATION OF CONTROL ROD NO. 9

(Test Report No. 100E, dated June 14, 1961, by K. Almenas)

INTRODUCTION

It has been the practice in the past to use the 9-control rod bank as a reactivity standard, however, for some purposes the precision possible with this method is insufficient. To provide, therefore, in effect, a "fine reactivity control," the central No. 9 control rod was calibrated over a distance of 10 in. of rod travel.

SUMMARY

The differential control rod worth is given in Fig. 100E-1. The precision of the measurements are discussed in detail and the maximum possible percentage error is found to be $\pm 8.3\%$. (Some unknown uncertainties, i.e.,

Fig. 100E-1. Differential Worth of Central Control Rod [Rods (Nos. 1-8) banked at 21.1 in.]

linearity of Keithley recorder in the range 10^{-10} to 10^{-9} are not included in this estimate.) Incidental to the experiment the worth of H_3BO_3 dissolved in the reactor water was found to be $1.59 \pm 0.32 \text{ } \rho / (\text{gm/gal})$.

EXPERIMENTAL PROCEDURE

The worth of an individual control rod is dependent on the position of the remaining eight. This calibration was performed with rods Nos. 1 through 8 at 21.1 in. The positive period method was used, with different rod positions being made possible by changes in the H_3BO_3 concentration. To minimize the effect of the source on the reactivity measurements, the power at which the just critical rod positions were taken was always identical ($\sim 10 \text{ kw}$ or 5×10^{-9} on the linear flux indicator). The data obtained and their interpretation are given in Table 100E-I.

CALCULATIONS

Reactivity measurements of reactors having strong stationary sources are performed under two power limitations, namely at low powers the source contributes an appreciable and indefinitely known fraction of the total flux, and the power cannot be raised too high if temperature effects are to be avoided. The limitations become narrower if, in order to allow ample time for transient decay, long periods are used. A way to minimize the source error is to take the static and dynamic reactivity measurements at the same power level so that the source affects both of them to an equal extent. This assumption is correct if the spatial flux distribution is identical in both cases. If, however, gross rod movements are made so that the flux distribution is distorted, identical instrument readings do not necessarily correspond to identical power levels. In the experiments reported here the Nos. 1-8 rods were at all times at 21.1 in., only the No. 9 rod (not shadowing the current chamber from the source directly) being moved.

SOURCE STRENGTH CALCULATION

Since the experiments were performed in a short span of time (three days as compared to the half-life of the Sb-Be source of 61 days), and since no high fluxes, which could reactivate the source, were reached at that time, it is possible to evaluate the source strength for the duration of the experiment.

Table 100E-1

SUMMARY OF CRITICALS PERFORMED DURING NO. 9 ROD CALIBRATION

Test	Remarks	1 H_3BO_3 Concentration, gm/gal	2 Consecutive Periods	3 P_1 Power at Start of Period	4 P_2 Power at End of Period	5 $\ln \frac{P_2}{P_1}$	6 ΔT , sec	7 Asymptotic Period, sec	8 Reactivity of Period, $\rho \times 10^5$	9 Average Central Rod Position, in.	10 Differential Rod Movement, in.	11 $\Delta \rho \times 10^5$	12 $\Delta \rho / \text{in.} \times 10^5$
A	Nos. 1-8 rods brought out to 21.1 in. No. 9 rod at 19 in. Power increasing slowly. Time. 14.41, May 22, 1961.	1.52 ± 0.03		44.1	51.9		150	980	7.92	20	2.1	85.8	42.9
	No. 9 rod dropped and raised after 10-min delay to bank position.			0.3	0.8155	1	36.1						
	After periods, No. 9 rod dropped and H_3BO_3 removed.		53.7	0.9	2.446	1	47.9						
			55.6	3	8.155	1	53.7						
			56.1	9	24.46	1	55.3						
B			57.2	18	48.93	1	56.9			13.1 13.35	1.8 2.3	99.03 119.9	55.02 52.13
			56.8	30	81.55	1	56.6	56.8	93.7				
	Nos. 1-8 rods unmoved. No. 9 rod brought out to 14 in.	1.26 ± 0.06	56.2	1	2.718	1	50.7						
			53.1	3	8.155	1	51.9						
			55.5	9	24.46	1	54.8	55.5 $\pm$ 1	95.0				
	No. 9 rod moved to 12.2 in.		-1573	61.6	60		45	-1875	-4.03				
	No. 9 rod dropped and raised after 10-min delay to 14.5 in. After test all rods dropped. H_3BO_3 diluted.		43	1	2.718	1	39.7						
			39.7	3	8.155	1	38.7						
			41.1	9	24.46	1	40.7						
			41.1	30	81.55	1	41.0						
C			40.1	50	135.9	1	41.2	40.8 $\pm$ 0.3	115.9				
	Nos. 1-8 rods brought out to 21.1 in. No. 9 rod at 10 in. reactor going up on period.	1.09 ± 0.02	130	0.9	2.446	1	89.0			9.54 9.87	0.92 1.62	46.12 77.9	50.13 48.09
			121	3	8.155	1	111.9						
			124	9	24.46	1	120.5	123.4 $\pm$ 2	51.1				
	No. 9 rod brought to 9.08 in. increase shown measured 160 sec after rod stop.		151.4	51	55		105	1530	4.98				
	No. 9 rod dropped and brought out after 10-min delay to 10.7 in. After test all rods dropped. H_3BO_3 added.		65.2	1	2.718	1	57.1						
			66.7	3	8.155	1	63.9						
			67.1	9	24.46	1	66.1						
			67.7	30	81.55	1	67.4	67.5	82.9				
D	Rods Nos. 1-8 brought out to 21.1 in. No. 9 rod after power rise lowered to 11.70 in. Power increasing slowly.	1.22 ± 0.03	3880	51.9	53.4		90	3500	2.16	12.85	2.3	124.9	54.30
	No. 9 rod dropped and raised after 10-min wait to 14 in.		35.1	1	2.718	1	32.9						
			35.6	3	8.155	1	34.8						
			35.8	9	24.46	1	35.5						
				30	81.55	1	34.6	35.6	127.1				

In the asymptotic period range

$$\frac{dn(t)}{dt} = \frac{n(t)}{T} + S \quad , \quad (1)$$

where:

T = asymptotic period

S = source strength (n/sec).

If dn/dt versus $n(t)$ is plotted, a straight line with a slope $1/t$ and an intercept S will result. Figure 100E-2 shows two such lines obtained after the central rod was dropped from 12.2 and 9.08 in. The units used are readings of the linear flux recorder (Keithley) on the 10^{-10} scale.

Fig. 100E-2. Derivative of Negative Periods Versus $n(t)$

At $dn/dt = 0$ we have

$$S = - \frac{n(t)}{T} = - \frac{n(t) k_{ex}}{\ell^*} = \frac{n(t) (1 - k_{eff})}{\ell^*} \quad . \quad (2)$$

The values of S obtained are:

$S = 0.0028 \times 10^{-10}/\text{sec}$ for Test B.

$S = 0.0031 \times 10^{-10}/\text{sec}$ for Test C.

In order to evaluate S from positive periods, very long periods must be available, since sufficient time must be allowed for the transients to die out, for short periods this carries the power too high above the source level. Test C had a reasonably long period which was evaluated at high power (assuming a source of $0.003 \times 10^{-10}/\text{sec}$) and using the relationship:

$$e^{(t_2-t_1)/T} = \frac{n(t_2) + S T}{n(t_1) + S T} \quad (3)$$

This relationship with T as a known parameter was then used to obtain S at low powers. Here

$$T = 123.4 \text{ sec}$$

and the calculated

$$S = 0.0035 \times 10^{-10}/\text{sec.}$$

The value of S used for the evaluation of the experiments was $0.003 \times 10^{-10}/\text{sec}$ (arithmetic average). The influence of a 50% error in this source value on a standard reactivity measurement was estimated and found to be $\pm 0.09\%$.

PERIOD EVALUATION

The asymptotic period was evaluated using Eq. (3) (an iterative root finding routine was coded for the LGP-30 computer for this purpose). Several overlapping readings in the higher power regions were taken and the results averaged in order to minimize reading and instrument noise errors. The final asymptotic period is given in Column 7, Table 100E-I.

The reactivity corresponding to this period was obtained from graphed solutions of the six-delay group inhour equation, the weighted delayed neutron fraction used being 6.55×10^{-3} (smaller than the U^{235} delayed fraction because of the Pu^{239} present.) Reactivity is given in Column 8, Table 100E-I.

ACCURACY OF METHOD

Since the objective of the calibration was to establish an accurate reactivity standard, it is important to evaluate the maximum possible errors inherent in the method. Test A was taken as a typical representative and the cumulative known error possibilities break down as follows:

Rod Positioning Error	$\sim \pm 2.0\%$
Error Due to Remaining Transients	$\sim \pm 1.8\%$
Temperature Effects	$\sim -0.6\%$

Source Errors	$\sim \pm 0.9\%$
Noise (Instrument and Neutrons)	$\sim \pm 1.2\%$
Time Reading Error	$\sim \pm 1.9\%$
Reactivity Reading Error	$\sim \pm 0.5\%$

$\pm 8.3\%$ of the total differential
worth at 20 in.

Not included in this total are some unknown instrument errors which are mentioned in the subsequent discussion.

Rod Repositioning Error

For the present differential measurements only one rod was involved. The position indicating dial could be read easily to within an accuracy of ± 0.02 . Since the total differential movement for Test C was 2 in., this contributed a maximum possible error of 2%.

NOTE

This assumes that the instrument dial position precisely indicated the control rod position, the extent of the non-linearity between indicated and actual positions was not known.

Error Due to Remaining Transients

The short period was determined 200 sec after rod movement stopped. According to Reference 1, the presence of transients after such a waiting time contributed less than 1% error in the asymptotic period or about $\pm 0.06 \times 10^{-5} \rho$ in the resulting reactivity. The long period was measured 5 min after control rod movement and a possible error of $\sim 10\%$ in the period and $\pm 0.7 \times 10^{-5} \rho$ existed. The total possible error in the difference of the two reactivities then was $\pm 1.5 \times 10^{-5} \rho$ or $\pm 1.8\%$.

Temperature Effects

The experiments were performed at a maximum power of ~ 10 kw; this was about three times the decay heat at that time. Since an equilibrium between subcooling and decay heat was established, the additional nuclear heat could raise the reactor water temperature $\sim 0.06^\circ\text{C}$ in an hour. The maximum time at 10 kw during any experiment was 10 min. The reactivity involved in the temperature rise of 0.1°C was $\sim -0.5 \times 10^{-5}$. (Temperature coefficient with control rods at ~ 20 in. was $\sim 5 \times 10^{-5} \rho/^\circ\text{C}$.) Thus the maximum percentage error was -0.6% .

Source Errors

Assuming an outside error of $\pm 50\%$ in the calculated source strength, an error of ± 0.3 sec corresponding to $\pm 0.4 \times 10^{-5} \rho$ was introduced in the short period and ± 50 sec corresponding to $\pm 0.35 \times 10^{-5} \rho$ into the long period. Total maximum error of the difference then was $\pm 0.75 \times 10^{-5} \rho$ or $\pm 0.9\%$.

Noise Effects

Neutron and instrument noises combined to make the power trace somewhat indefinite. In the experimental ranges the maximum-to-average noise peaks were about 3% of the total power reading. However, since peaks are never read but an approximate average (by eye) is estimated, the power reading error should not exceed $\pm 1\%$. This corresponds to a reactivity error of $\pm 0.08 \times 10^{-5} \rho$ for the short and $\pm 0.02 \times 10^{-5} \rho$ for the long period. Cumulative total for the difference was $1 \times 10^{-5} \rho$ or $\pm 1.2\%$.

Time Reading Errors

The time during which the power rise occurred in this experiment was read from the flux chart. A precision of at least ± 1 sec was possible in this method, introducing an error of $\pm 1.2 \times 10^{-5} \rho$ for the short period and an entirely negligible error for the long one. Total per cent error then was $\pm 1.4\%$.

NOTE

Two additional comments apply to error sources, "Noise Effects" and "Time Reading Errors." First, the random reading errors were minimized by taking several overlapping period readings and using the averaged result. Second, the estimated errors are errors of reading and graphed noise only and do not include possible inaccuracy in the recording instrument. That is, chart travel was assumed to be precisely constant with time and the flux chamber was assumed to be linear over power rises by factors of 2 to 3.

Reactivity Reading Errors

The reactivity corresponding to an asymptotic period was read off a graph. The accuracy of a reading was about $\pm 0.5\%$ of the total. The error applicable to the difference was then about $\pm 0.5\%$ since the relative error of the long period reactivity was insignificant.

NOTE

The accuracy mentioned here is the reading accuracy and not the absolute accuracy of the reactivity in question. The absolute accuracy was probably significantly less since the total effective delayed neutron fraction of

this isotope mixture is difficult to calculate (i.e., the spatial distribution of Pu^{239} in calculating beta is ignored). This absolute error would not contribute any scatter to the points of Fig. 100E-1, being equally applicable to all of them; it does become important, however, when comparisons to theoretical calculations are to be made.

Total Reactivity Error

As seen, the total cumulative error possible from the listed causes amounts to $\pm 8.3\%$. The scatter of points on Fig. 100E-1 seems to be of that order of magnitude ($\sim 5\%$). The probability that such a large majority of the listed errors were additive is small; therefore, it is quite possible that the mentioned unknown instrument errors (rod position, chart travel, flux linearity) were appreciable, maybe even predominant.

Error in H_3BO_3 Worth

The titrometric H_3BO_3 concentration determination had a precision of $\sim \pm 3\%$ (± 0.2 ml out of a total of 6-8 ml). The difference in H_3BO_3 concentration over the calibrated rod travel was $1.52 \pm 0.03 - 1.09 \pm 0.02 = 0.44 \pm 0.05$ or $\sim 12\%$. The possible error in the integrated worth of the No. 9 rod should not have exceeded half of the possible error of individual measurements, the total possible known error in H_3BO_3 worth was then $\sim 16\%$ giving a worth of $1.59 \pm 0.25\% \rho / (\text{gm/gal})$. This, however, does not include possible errors in the calculated (6.55×10^{-3}) effective delayed neutron fraction. If that number should be off $\sim 5\%$, the worth of H_3BO_3 becomes $1.59 \pm 0.32\% \rho / (\text{gm/gal})$, which is still appreciably higher, even at its lowest possible value, than the previously used calculated values of 1 to $1.16\% \rho / (\text{gm/gal})$.

REFERENCE

- (1) B. J. Toppel, "Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements," Nuc. Science and Eng. 5, pp. 88-89 (February, 1959).

EBWR CONTROL ROD REPLACEMENT

(Test Report No. 119, dated January 8, 1962,
by V. M. Kolba, E. A. Wimunc, E. L. Martinec, and J. F. Matousek)

The purpose of this test was to replace the fueled follower control rods with Zircaloy-2 follower control rods.

Two fission counters were installed in the new spike fuel element storage rack. The old Be target was placed in the spike rack and Sb source I-b was installed in the target. This installation permitted monitoring the rack while loading spikes.

The sequence of unloading the reactor is shown in Table 119-I. A critical was run first to establish a base value. The reactor was critical with all rods banked at 16.05 in. The rods were then banked at 6 in. and the first 8 moves, removing 8 central ET fuel elements, were made. All fuel elements were transferred from the core to the storage pit via the coffin, and lowered directly into the storage pit. The upper limit stop on the control rod drives was then changed to 60 in. The reactor was then critical with the 9 rods banked at 26 in. A base count of the spike storage counters was taken and then Moves 9-12, removing 4 spike elements, were made. The spikes were first placed in location C-14 or the spike rack. This was the point of transfer of the spike and was selected because of restrictions on coffin movement over the storage pit. A one minute count was taken at this point. Counts and moves in the storage pit are noted in Table 119-II. The spike was then moved from location C-14 using the rigid tool and the overhead crane to the final location in the storage rack. After being placed in the final location, 2 one minute counts were taken. After Move 12 the reactor was critical with rods Nos. 1, 2, 7, and 8 @ 40 in. and rods Nos. 3, 4, 5, 6, and 9 at 39 in. Moves 13-16 were then made and after that the reactor was subcritical with all rods out to 60 in. The rods were then banked to 24 in. and Moves 17-29 were completed.

Table 119-I

FUEL UNLOADING

Date	Time	Move	Fuel Element Number	From - Location	To - Location
12/11/61	Critical - all rods @ 16.05 in.				
	14:25	1	ET-5	C.I. 129-370	S.P. F-1
	14:56	2	ET-54	129-159	F-2
	15:17	3	ET-46	129-264	F-3
	15:39	4	ET-18	129-53	F-4
12/12/61	15:59	5	ET-17	109-148	I-5
	09:29	6	ET-52	109-359	K-6
	09:56	7	ET-33	109-42	K-7
	10:24	8	ET-55	109-254	K-8
12/13/61	Critical - all rods @ 26 in.				
	11:00	9	S-5	C.I. 47-308	S.P. C-26
	12:50	10	S-9	63-337	B-25
	13:42	11	S-1	63-312	B-26
	14:06	12	S-2	63-101	A-25
12/14/61	Critical - rods Nos. 1, 2, 7, and 8 @ 40 in. rod Nos. 3, 4, 5, 6, and 9 @ 39 in.				
	15:25	13	S-10	C.I. 72-360	S.P. A-24
	15:51	14	S-16	72-149	C-24
	16:13	15	S-8	72-43	A-26
	16:32	16	S-15	72-254	B-24
12/15/61	Subcritical with all rods @ 60 in.				
	09:57	17	S-7	C.I. 72-414	S.P. A-22
	13:04	18	S-28	72-202	B-22
	13:27	19	S-11	72-96	C-23
	13:52	20	S-12	72-308	C-23
	14:15	21	S-14	77-64	B-23
	14:42	22	S-13	77-276	C-22
	15:28	23	S-27	77-170	B-21
	15:55	24	S-6	77-382	A-21
	16:21	25	S-18	77-402	B-20
	08:56	26	S-4	77-85	C-20
	09:17	27	S-3	77-297	C-27
	09:47	28	S-17	77-170	C.I. 77-191
	10:02	29	S-17	77-191	S.P. C-21

Table 119-II

SPIKE STORAGE IN NEW STORAGE RACK

Date	Move	Counts/min		Location			Spike	Spike Location	Strip Orientation N↑
		Counter 1	Counter 2	Counter 1	Counter 2	Source			
12/13/61	1	225	68	C-27	B-27	C-25			
		249	66						
	2	213	54				S-5	C-14	
	3	29,173	12,129				S-5	C-26	
		29,049	12,072				S-5	C-26	
	4	37,270	12,903				S-5	C-26	
		37,332	13,145				S-5	C-26	
	5	9	9				S-5	C-26	
	6	30,627	12,493				S-9	C-14	
	7	34,494	15,470				S-9	B-25	
		34,538	15,284				S-9	B-25	
	8	34,272	15,263				S-1	C-14	
	9	41,667	29,474				S-1	B-26	
		41,495	29,978				S-1	B-26	
	10	41,712	30,314				S-2	C-14	
	11	42,189	30,976				S-2	A-25	
		42,033	31,228				S-2	A-25	
		41,998	30,829				S-2	A-25	
	12	41,757	30,869				S-10	C-14	
	13	42,825	30,023				S-10	A-24	
		42,607	29,882				S-10	A-24	
	14	42,934	26,879				S-16	C-14	
	15	43,865	28,106				S-16	C-24	
		43,462	28,115				S-16	C-24	
	16	43,573	27,732				S-8	C-14	
	17	41,746	33,586				S-8	A-26	
		41,807	33,280				S-8	A-26	
	18	43,392	35,062				S-15	C-14	
12/14/61		39,379	34,885				S-15	C-14	
	19	37,402	35,902				S-15	B-24	
		37,882	36,373				S-15	B-24	
	20	37,557	35,368				S-7	C-14	
	21	37,694	35,543				S-7	A-22	
12/14/61		37,543	35,390				S-7	A-22	
	22	26	32,660	A-20			S-7	A-22	
		32	33,138				S-7	A-22	
	23	25	32,839				S-28	C-14	
	24	48	32,307				S-28	B-22	
		32	32,527				S-28	B-22	
	25	32	32,325				S-11	C-14	
	26	39	29,620				S-11	C-23	
		41	29,256				S-11	C-23	
	27	39	29,107				S-12	C-14	
							S-12	C-14	
	28	49	29,071				S-12	A-23	

Table 119-II (Cont'd.)

Date	Move	Counts/min		Location			Spike	Spike Location	Strip Orientation N↑
		Counter 1	Counter 2	Counter 1	Counter 2	Source			
12/14/61	28	43	28,924	A-20	B-27	C-25	S-12	A-23	
	29	63	29,144				S-14	C-14	
	30	93	29,395				S-14	B-23	
		92	29,405				S-14	B-23	
	31	82	29,123				S-13	C-14	
	32	95	30,326				S-13	C-22	==
		96	32,171				S-13	C-22	
			32,040				S-13	C-22	
	33	109	31,334				S-27	C-14	
	34	156	29,479				S-27	B-21	==
		154	29,164				S-27	B-21	
	35	151	28,867				S-6	C-14	
	36	279	29,350				S-6	A-21	==
		257	28,968				S-6	A-21	
	37	277	28,923				S-18	C-14	
	38	329	28,974				S-18	B-20	==
		333	34,736*				S-18	B-20	
		311	34,941*				S-18	B-20	
		321	34,923*				S-18	B-20	
	12/15/61	316	34,422*				S-18	B-20	
12/15/61		297	28,168				S-18	B-20	
	39	323	29,059				S-4	C-14	
	40	297	28,797				S-4	C-20	
		327	28,975				S-4	C-20	
12/15/61	41	302	29,025				S-3	C-14	
	42	288	36,839				S-3	C-27	
		282	36,408				S-3	C-27	
	43	277	36,547				S-17	C-14	
	44	422	28,979				S-17	C-21	
		429	29,123				S-17	C-21	
		432	29,159				S-17	C-21	
Physics Experiment									
12/15/61	A	0	0	A-20	B-27	Removed			
	B	Jam	Jam			A-19			
	C	283	3			A-18			
		336	3			A-18			
		304	4			A-18			
	D	5(10)	7			A-17			
		8/1 min	3/1 min			A-17			
		13/2 min	5/2 min			A-17			
		19/3 min	8/3 min			A-17			
	E	2,402	1			A-17	S-17	C-21	
		2,467	3			A-17	S-17	C-21	
	F	430	29,150			C-25	S-17	C-21	==

* Tied tool to chamber tube.

Monitoring of the storage pit during loading of the spikes revealed no gross change in counting rate once a nominal level had been established. The condition of the core after fuel element removal is shown in the attached fuel status board, Fig. 119-1. The condition of the storage pit after loading the spikes is shown in the attached storage rack fuel status board, Fig. 119-2.

Fig. 119-1. Core - Fuel Status Board after Fuel Element Removal (Core Load No. 59a, 12/15/61)

After the fuel elements were removed, the changing of the control rods was begun. Eight control rods were banked at 24 in. and the rod to be changed was raised to 37 in. The tack weld on the handling tip was then broken remotely. Some difficulty was experienced in breaking some of the tack welds, however, all were successfully broken. The handling tip was then unscrewed and removed from the reactor. The activity of the handling tip was ~ 10 mr. The handling tool for the 10-in. cross, located from a winch over the coffin, was attached to the cross and the rod was raised to 60 in. The 10-in. cross was then freed from the locating pins and raised in 6-in. increments for the first 2 ft, until clear of the core, then raised into the coffin. Counts were noted in the control room while removing the fueled portion from the active region of the core. In several instances, freeing an assembly was difficult and accomplished by driving the rod down $\sim 1/2$ in. with the drive unit. This permitted the utilization of the torque limiter on the

control rod drive to establish a limited maximum force on the assembly. All units were freed successfully by either the hand crank or the latter method. The 10-in. cross was then taken to the storage pit, lowered directly into the storage pit from the coffin, and placed in the storage rack as shown in the attached EBWR storage rack fuel status board, Fig. 119-2.

The handling tip was then rethreaded on the 1/2-in. bar and, after appropriate tool changes, the 3-in. cross and 1/2-in. bar were lowered to the "all in" position. The 3-in. cross was then unlatched from the drive mechanism; and the 1/2-in. bar and 3-in. cross were then raised slowly using the winch on the overhead crane. In several instances some difficulty was experienced with an interference of the 10 in. transition zone and the grid plate, so in this stage of removal the unit was raised by manually pulling on the lifting cable until this area of "hang up" was cleared. The unit was then raised out of the reactor and transferred in air to the storage pit. Activity was ~ 200 mr at 10 ft during this transfer operation. The 3-in. cross-1/2-in. bar assembly was lowered into the storage pit and placed in the old fuel element storage rack. These units were later transferred to the control rod storage rack and placed together with the respective control rod poison and fuel section.

The new rod was then loaded into the reactor. On entering the active portion of the core, with the Zircaloy-2 follower section, the rod was lowered in 6-in. increments while counts were noted on the plant counters. After seating, the rod was latched and tested for attachment. Six rod drops were run on each new rod, after installation, to check the drop time. All rod drops were free and within the prescribed time limits. All rods were changed by the above method. Table 119-III lists the sequence of removal of the old rods and the poison plate numbers of the new replacement rods and a completed rod which was not loaded. This last rod was to replace the No. 9 rod after the rod oscillation tests.

Table 119-III
CONTROL ROD CHANGE

Date	Time	Old Rod No.	New Rod Poison Numbers				Time New Rod In	CS Markings	
			1	2	3	4		Tape	Green No.
12/18/61	13:10	2	1	35	36	39	15:46	5	-
12/19/61	09:05	1	5	11	26	33	14:45	9	6
↓ 12/20/61	16:03	3	4	8	17	25	17:20	25	7
	10:06	4	7	23	31	34	11:17	6 or 9	4
	13:20	5	10	24	32	42	15:35	1	3
↓ 12/21/61	15:20	6	13	19	37	40	16:35	3	X
	13:20	7	3	22	27	38	14:30	7	5
	14:35	8	12	14	15	16	15:49	8	-
	Rod not loaded Distorted heat treated cross		2	21	28	29		-	2
↓			9	18	30	41			

Boron plates 6 and 20 cracked in forming. These plates were not used for poison crosses in control rods.

After all the rods were changed the fuel was reloaded in accord with the moves in Table 119-IV. The control rods were banked at 24 in. during loading of the eight ET fuel elements and the first three spike fuel elements.

Table 119-IV

FUEL LOADING

Date	Time	Move	Fuel Element No.	From - Location		To - Location		Strip Orientation N ↑
12/26/61	13:35	1	ET-5	S.P.	F-1	C.I.	129-370	
	14:12	2	ET-54		F-2		129-159	
	14:42	3	ET-46		F-3		129-264	
	15:04	4	ET-18		F-4		129-53	
	15:28	5	ET-17		I-5		109-148	
	15:51	6	ET-33		K-7		109-42	
	16:15	7	ET-52		K-6		109-359	
	16:35	8	ET-55		K-8		109-254	
12/27/60			Subcritical - all rods out to 48 in.					
	10:37	9	S-6	S.P.	A-21	C.I.	77-402	
	11:12	10	S-18		B-20		77-191	
	11:32	11	S-4		C-20		77-85	
	15:25	12	S-3		C-27		77-297	
	15:42	13	S-14		B-23		77-64	
	16:02	14	S-13		C-22		77-276	
	16:26	15	S-17		C-21		77-170	
12/28/61			Critical - all rods @ 36 in.					
	09:31	16	S-27	S.P.	B-21	C.I.	77-382	
	09:52	17	S-7		A-22		72-414	
	10:14	18	S-28		B-22		72-202	
12/28/61	10:48	19	S-11		C-23		72-96	
			Critical - all rods @ 24.4 in.					
	13:18	20	S-12	S.P.	A-23	C.I.	72-308	
	13:46	21	S-10	S.P.	A-24	C.I.	72-360	
			Critical - all rods @ 22 in.					
	15:01	22	S-16	S.P.	C-24	C.I.	72-149	
	15:20	23	S-8		A-26		72-43	
	15:42	24	S-15		B-24		72-254	
12/29/61			Critical - all rods @ 17 in.					
	09:10	25	S-1	S.P.	B-26	C.I.	63-312	
	13:35	26	S-2	S.P.	A-25	C.I.	63-101	
			Critical - Rods Nos. 1-8 @ 16 in. and rod No. 9 @ 17 in.					
1/2/62	16:35	27	S-5	S.P.	C-26	C.I.	47-308	
			Critical - Rods Nos. 1-8 @ 15.8 in. and rod No. 9 @ 15.6 in.					
	09:49	28	S-9	S.P.	B-25	C.I.	63-337	
			Critical - Rods Nos. 1-8 @ 14.5 in. and rod No. 9 @ 15 in.					

The ET elements were loaded into the coffin directly from the storage pit location, transferred to the reactor, and loaded into the reactor in 6-in. increments, when in the active portion of the core; the plant counters were noted while lowering the elements. After loading the eight central ET elements, Moves 1-8, the reactor was subcritical with all rods banked at 48 in.

The spike elements were first moved to the common transfer point (C-14), and then loaded directly from the storage rack into the coffin; they were then transferred to the reactor and lowered in 6-in. increments, when in the active portion of the core, while counts were noted on the plant counters. Reciprocals of the counting rate were plotted after loading each spike. This data is shown in Table 119-V and plotted in Fig. 119-3.

Table 119-V
LOADING OF SPIKES

Date	Fuel Element No.	Plant Counter 1		Plant Counter 2		Rod Bank Position, in.	
		counts/ 5 min	$\frac{1}{\text{counts/min}}$	counts/ 5 min	$\frac{1}{\text{counts/min}}$		
12/27/61	ET-5 thru ET-55 loaded	225	0.0222	305	0.0164	24	
	S-6	440	0.0114	556	0.009	24	
	S-18	653	0.00765	828	0.00605	24	
	S-4	686	0.0073	923	0.0054	24	
		381	0.0132	478	0.0105	20	
	S-3	378/386	0.0132	519/523	0.00965	20	
	S-14	482	0.0104	614	0.00814	20	
	S-13	543	0.0092	695	0.0072	20	
	S-17	765	0.00652	1142	0.00437	20	
		Critical - 62 sec period, all rods @ 36 in.					
12/28/61		414	0.0121	579	0.00866	16	
	S-27	659	0.0076	940	0.0053	16	
	S-7	1211	0.00413	1846	0.00271	16	
		741	0.00675	1179	0.00425	14	
	S-28	1479	0.00338	1870	0.00268	14	
	S-11	1591	0.00315	2253	0.00222	14	
		Critical - all rods @ 24.4 in.					
		1003	0.005	1448	0.00345	12	
	S-12	1136	0.0044	1544	0.00324	12	
	S-10	1875	0.00267	2390	0.0021	12	
		Critical - all rods @ 22 in.					
		Subcritical - with Rods Nos. 1, 2, 3, 4, 6, 7, 8, and 9 @ 12 in. and rod No. 5 @ 48 in.					
		S-16	2586	0.00194	4027	0.00125	12
		S-8	3307	0.00152	5257	0.00095	12
		S-15	4522	0.00111	6396	0.000782	12
	Critical - all rods @ 17 in.						
	Subcritical - with any one rod (Nos. 1-9) raised to 48 in. with the remaining 8 rods at 12 in.						
12/29/61	S-1	5099	0.00098	7082	0.000701	12	
	S-2	5799	0.000863	8631	0.00058	12	
		Critical - all rods @ 16 in.					
	S-5	6729	0.000743	10,132	0.000493	12	
	Critical - Rods Nos. 1-8 @ 15.8 in. and rod No. 9 @ 15.6 in.						
1/2/62	S-9	12,732	0.000393	18,428	0.000271	12	
	Critical - Rods Nos. 1-8 @ 14.5 in. and rod No. 9 @ 15 in.						

Reactor was subcritical with any rod raised to 48 in. with remaining 8 rods at 12 in.

Fig. 119-3

Loading of Spikes Into Reactor

$$\left(\frac{1}{\text{counts/min}} \text{ Versus } \begin{array}{c} \text{Number of Spikes} \end{array} \right)$$

rod change is 16.05 in. - 14.5 in. = ~ 1.5 in. of rod bank equivalent to ~ 0.3% ρ .

The reactor is subcritical with any rod raised to 48 in. and the remaining rods banked at 12 in. Number 5 rod was withdrawn to 60 in. and the linear flux started dropping off while raising the rod from 48 in. to 60 in. All limits were set to 48 in.

The core is now reloaded to the No. 59a core load, Reference 1.

REFERENCE

- (1) EBWR Test Report No. 110N.

After Move 11, three spikes having been loaded, the reactor rod bank position was changed to 20 in. After Move 15, seven spikes loaded, the reactor was critical with all rods banked at 36 in. The rod bank position for loading was changed to 16 in. after Move 15 and to 14 in. after Move 17. After loading eleven spikes, Move 19, the reactor was critical with all rods banked at 24.4 in. The rod bank position for loading was then changed to 12 in. for the remainder of the loading of the spikes. After loading 13 spikes, Move 21, the reactor was critical with all rods banked at 22 in. After addition of 16 spikes, Move 24, the reactor was critical with all rods banked at 17 in. After addition of 18 spikes, Move 26, the reactor was critical with rods Nos. 1-8 banked at 16 in. and rod no. 19 at 17 in. With 19 spikes loaded, Move 27, the reactor was critical with rods Nos. 1-8 banked at 15.8 in. and rod No. 9 at 15.6 in. With all 20 spikes loaded, Move 28, the reactor was critical with rods Nos. 1-8 banked at 14.5 in. and rod No. 9 at 15 in. The net gain of the

A PRELIMINARY CALIBRATION OF THE NEW SYSTEM OF CONTROL RODS IN THE COLD EBWR

(Test Report No. 100G, dated January 22, 1962, by H. P. Iskenderian)

INTRODUCTION

Recently the eight outer control rods with stainless steel fueled followers, used in the EBWR, were replaced by similar borated steel rods but with Zircaloy followers. It was estimated that this change would result in a gain of about 1.8% in bank worth of the rods, when fully inserted. It was of interest, therefore, to obtain a calibration of the new system of rods, and to compare results with those of the old rods, which had been calibrated in a core of similar fuel element configuration. It was not planned to obtain a calibration of highest accuracy, but it was expected that the increase in the worth of the new rods would be noted.

METHOD OF CALIBRATION

The method of bank calibration of the rods consisted in adding known amounts of H_3BO_3 (Table 100G-I), and after mixing (blowing air) for about half an hour, obtaining a bank criticality. The rod removal worths $\delta\rho$ were then evaluated (Table 100G-II) from

$$\delta\rho = \frac{\bar{\Sigma}_a^B}{\bar{\Sigma}_a^B + \bar{\Sigma}_a} \quad (1)$$

where

$\bar{\Sigma}_a^B$ was calculated for each addition of H_3BO_3 (Table 100G-I), and

$\bar{\Sigma}_a$, the reactor absorption cross section, was determined as explained below.

Table 100G-I

CONCENTRATION OF BORIC ACID IN REACTOR WATER AND $\bar{\Sigma}_a^* B = \bar{N} B \quad \sigma_a^* B$ Grams of H_3BO_3 per Gallon of
 H_2O Determined by

Weighing (1)	Titration (2)	Average of (1) and (2)	$\bar{\Sigma}_a^* B$
0.20	0.25	0.225	0.0002143
0.80	0.84	0.82	0.0007813
1.41	1.44	1.43	0.0013625
1.896	1.86	1.88	0.0017893
3.06	2.98	3.02	0.0028774
4.10	3.96	4.03	0.0038398

$$*\bar{\Sigma}_a^* B = \bar{N} B \quad \sigma_a^* B$$

$$= \left(\frac{0.602}{61} \times \frac{\text{gm/gal of } H_3BO_3}{3780 \text{ cc/gal}} \right) \times (\sqrt{H_2O} = 0.63) (\sigma_a^* B = 755 \times 0.767)$$

Table 100G-II

BANK CALIBRATION OF CONTROL RODS

Grams H_3BO_3 per Gallon H_2O	$\bar{\Sigma}_a^* B$	$\delta\rho = \frac{\bar{\Sigma}_a^* B}{\bar{\Sigma}_a + \bar{\Sigma}_a^* B}$	Control Rod Bank Position (inches above bottom of core)
0	0	0	14.17
0.82	0.0007813	0.9215%	16.55
1.43	0.0013625	1.596%	18.62
1.88	0.0017893	2.086%	20.4
3.02	0.0028774	3.312%	25.7
4.03	0.0038398	4.371%	32.32

DETERMINATION OF $\bar{\Sigma}_a$

The present EBWR core consists of fuel elements with thin and thick enriched and natural uranium plates, and fully enriched fuel spikes. With the aid of previous data (Reference 1), it was estimated $\bar{\Sigma}_a \simeq 0.085$.

With the present calibration of the rods, $\bar{\Sigma}_a$ was determined as follows:

With the central rod (No. 9) down, the reactor was just critical when the remaining rods Nos. 1-8 were at a 16.5-in. position above the bottom of the core.

Four positive period measurements were made by moving the central rod into positions of 3.1, 4.2, 4.5, and 5.9 in. from its down position of criticality, Table 100G-III. (These period measurements were separated by about 7 min to minimize any temperature rise in the core that may have occurred during these tests.)

Table 100G-III

FOUR POINT CALIBRATION OF CENTRAL ROD NO. 9 WITH RODS NOS. 1-8
AT 16.5-in. POSITION

Rod No. 9	Reactor On	P_1	P_2	$\ln \frac{P_2}{P_1}$	ΔT , sec	Asymptotic Period, sec	$\frac{\delta \rho^*}{\beta}$	$\delta \rho$
Down	Critical							0
3.1 in.	(+) Period	3	8.15	1	55.4	71	0.122	0.08%
		3	8.15	1	71			
4.2 in.	(+) Period	3	8.15	1	32			
		3	8.15	1	34	34	0.20	0.131%
4.5 in.	(+) Period	3	8.15	1	31			
		3.1	8.45	1	31	31	0.211	0.138%
5.9 in.	(+) Period	2.9	7.9	1	11			
		3	8.15	1	11.8			
		3.3	9.0	1	11.6	10	0.382	0.25%

* Obtained from Inhour Equation, with $\beta = 6.55 \times 10^{-3}$ for spiked core.

The central rod removal worths $\delta \rho$ were evaluated from the inhour versus period curve available for the spiked core with burnup of ~ 9200 Mwd ($\beta = 6.55 \times 10^{-3}$ used).

On Fig. 100G-1 are shown $\delta\rho$ versus central rod removal position with remaining rods Nos. 1-8 at a 16.5-in. position.

Following this partial calibration of the control rod, H_3BO_3 was added in the amount of 0.225 gm/gal to the reactor water, and criticality obtained with the central rod at a 6-in. position, and retaining rods Nos. 1-8 at 16.5 in.

With the aid of the curve of Fig. 100G-1, and Eq. (1), $\bar{\Sigma}_a$ was calculated to be $\bar{\Sigma}_a = 0.0855$, which is in close agreement with the estimated value given above.

Fig. 100G-1

Worth of Central Rod When Rods (Nos. 1-8) Are at a 16.5-in. Position (above the Bottom of the Core)

BANK CALIBRATION OF RODS

Having determined $\bar{\Sigma}_a$, the removal worths of the bank of 9 rods were calculated with the aid of Eq. (1) as shown on Table 100G-II. Results are plotted on Fig. 100G-2.

Fig. 100G-2

Bank Calibration of EBWR Control Rods

DISCUSSION OF RESULTS

On Fig. 100G-2 the results of the present and of a previous calibration of the old control rod system with fueled stainless steel followers are plotted. These curves indicate removal worths of the two systems of rods to be nearly equal in the range of about 14 - 32 in. above the bottom of the core.

The errors in the present calibration are:

- 1. Error in the 4-point calibration of the central rod by period measurements.
- 2. Error in the determination of the content of boric acid in the reactor water.

The first type of error should not be large for the range of periods used; it is estimated to be ~ 3 to 4%.

The content of H_3BO_3 was determined by direct weighing and by titration of samples from the top and bottom of the core (see Table 100G-I). Errors should not exceed 5% except for the initial low concentration of 0.225 gm/gal of reactor water.

It is difficult to explain the nearly equal worth of the two system of rods. More careful measurements should be made by both the period method and the present method used when loading the core for 100-Mw operation.

REFERENCE

- (1) R. Avery, et al., "EBWR Core 1A Physics Analysis," ANL-6305 (February, 1961).

INSPECTION OF THE EBWR CONTROL ROD

(Test Report No. 119A, dated July 6, 1962, by T. L. Kettles)

INTRODUCTION

On April 11, 1962, the No. 3 (2 w/o boron-stainless steel) side rod and No. 9 (hafnium) center rod were examined visually in the reactor vessel. No flaws were discovered.

2 w/o BORON-STAINLESS STEEL ROD

Side rod No. 3, which had been installed in December, 1961, and had undergone 7,780 Mw hours of operation, was raised for examination to a maximum elevation two feet above the surface of the water filled reactor vessel. This permitted the upper poison section of the rod to be observed in air, and the lower portion through a water depth less than that at which weld cracks and failed rivets had been readily discernible in the original control rods. Examination of underwater surfaces was facilitated by a mirror arrangement, a low power periscope, and high intensity light sources.

No cracks could be seen at the corner radii, at the cruciform edges, nor in any other portion of the base material. None of the rivets exhibited failure indications such as loss of heads, opening of the sandwich construction, or the dead non-metallic appearance characteristic of the original failed rivets.

HAFNIUM CONTROL ROD

Center rod No. 9 which was installed in February, 1960, had undergone 15,674 Mw-hr of operation, and had been oscillated for 112 hr during March and April, 1962. Radioactivity was such that all portions of this rod had to be examined underwater.

The spotwelded transition, between hafnium and Zircaloy-2, of the original center hafnium control rod had failed due to the forces imposed during the oscillation tests. However, no distress was observed at this transition on the subject rod; there was no opening of the sandwich construction; the spotwelds were sound; and none of the rivets added to reinforce the transition had failed.

CALIBRATION OF CONTROL ROD SYSTEM IN EBWR - EMPIRICAL DETERMINATION OF β_{eff}

(Test Report No. 100I, dated January 31, 1963, by H. P. Iskenderian)

This report covers a cold calibration of the control rod system with Zircaloy followers. The calibration consists of bank criticality data for various concentrations of H_3BO_3 , and the differential worth of the bank about each of the criticality settings. Specifically, the following data were obtained:

1. Bank calibration of rods Nos. 1-9, from 10 to 22 in.
2. Calibration of central rod No. 9, for positions of the bank of remaining rods (Nos. 1-8).
3. A criticality line for varying bank settings of rods Nos. 1-8, and the corresponding criticality setting of the No. 9 rod.
4. An empirical determination of the core averaged β_{eff} .

BANK CALIBRATION OF RODS NOS. 1-9 WITH ZIRCALOY FOLLOWERS

Bank criticality setting data for H_3BO_3 concentrations of 5.05, 4.85, 4.04, 2.8, 2.18, and 0.35 gm/gal, and values of differential worths at each of these bank settings are given in Table 100I-I. This calibration is given, as previously, in percents of reactivity " ρ " (see Fig. 100I-1).

For the purpose of comparison with the present control rod system the calibration for the previous control rod system with steel fueled followers is also shown on Fig. 100I-1. These calibrations were both obtained by

Table 100I-I

BANK CALIBRATION OF RODS NOS. 1-9
($\beta = 6.55 \times 10^{-3}$ taken)

H ₃ BO ₃ Conc., gm/gal	Critical Bank Pos., in.	Raised Bank Pos., in.	Asymptotic Period, sec	ρ/β	ρ , %	$\therefore$ at Bank Pos., in.	$\rho/\text{in.}$, %
5.05	22.18	—	—	—	—	—	—
4.85	21.6	22.1	23	0.25	0.1637	21.85	0.3274
4.04	18.73	19.20	19.47	0.278	0.182	18.96	0.364
2.8	15.5	15.8	29.6	0.218	0.142	15.65	0.47
2.18	13.91	14.21	26	0.235	0.154	14.06	0.513
0.35	9.86	10.16	20.3	0.27	0.1768	10.01	0.589

Fig. 100I-1.

Calibration of Control Rods
(Zircaloy Followers)

CALIBRATION OF CONTROL ROD NO. 9, AND CRITICALITY LINE

The experimental data, calculated reactivity worths, and differential worths for various settings of the central rod No. 9, for indicated bank settings of rods Nos. 1-8, are presented in Table 100I-II. Calibration of control rod No. 9 is shown on Fig. 100I-2.

The criticality settings of rod No. 9 for indicated bank settings of Rods Nos. 1-8, i.e., the criticality line, are shown on Fig. 100I-3.

integration of the differential worths of the rods, and $\beta = 6.55 \times 10^{-3}$ was assumed. It will be noted, the more recent control rod system with Zircaloy followers has greater worth by 0.85% over the range of 10 - 22 in. of control rod calibration. This increased worth of the new system of rods is mainly due to the replacement of fueled stainless steel followers by Zircaloy followers. It is also, to a lesser degree, due to a reduction in the value of $\bar{\beta}_{\text{eff}}$ with a buildup of plutonium which was not allowed for in these calibration curves.

Table 100I-II

CALIBRATION OF CENTRAL ROD NO. 9 ($\beta = 6.55 \times 10^{-3}$ taken)

1. Rod Bank, Nos. 1-8, @ 25.28 in. H_3BO_3 Conc. 5.25 gm/gal No. 9 Rod in Down Position. Reactor Critical $\rho = 0$

Raised Pos. of Rod No. 9, in.	Asymptotic Period, sec	ρ/β	ρ , %
4	184	0.056	0.0367
6	85	0.105	0.0689
8	43	0.17	0.1113
9.5	27	0.23	0.1508
10.5	20	0.273	0.1788

2. Rod Bank, Nos. 1-8, @ 23.01 in. H_3BO_3 Conc. 4.85 gm/gal No. 9 Rod @ 8.4 in. Reactor Critical $\rho = 0$

Raised Pos. of Rod No. 9, in.	Asymptotic Period, sec	ρ/β	ρ , %
13	29.5	0.215	0.140
14.4	18	0.288	0.1886

3. Rod Bank, Nos. 1-8, @ 23.01 in. H_3BO_3 Conc. 5.05 gm/gal No. 9 Rod @ 14.4 in. Reactor Critical

Raised Pos. of Rod No. 9, in.	Asymptotic Period, sec	ρ/β	ρ , %
18.1	37.4	0.19	0.1244
20.0	21.7	0.26	0.1703
21.5	14	0.325	0.2129

Fig. 100I-2

Calibration of Control Rod No. 9
(Rods, Nos. 1-8, at
23.01-25.98-in. Positions)

Fig. 100I-3

For Criticality:

$$\frac{\delta \text{ Position of Rods, Nos. 1-8}}{\delta \text{ Position of Rod No. 9}} \approx -\frac{1}{9}$$

SEMI EMPIRICAL DETERMINATION OF THE $\bar{\beta}_{\text{eff}}$ OF THE REACTOR

The reactor's averaged delayed neutron fraction $\bar{\beta}_{\text{eff}}$ may be obtained by changing the reactivity of the core by additions of H_3BO_3 , and noting the criticality settings of the control rod system. For two different concentrations of H_3BO_3 , with Σ_a^{B1} and Σ_a^{B2} , the calculated change in reactivity will be:

$$\Delta \rho \approx \frac{\Sigma_a^{B2} - \Sigma_a^{B1}}{\Sigma_a + \Sigma_a^{B2}} \quad (1)$$

where:

Σ_a^{B1} are the thermal poison cross sections due to H_3BO_3

and

Σ_a is the reactor's averaged thermal cross section.

Equation (1) is, in effect, the reactivity worth of H_3BO_3 , which was calculated to be nearly equivalent to 1% ρ for a H_3BO_3 concentration of 1 gm/gal.

The difference in the reactivity worths of the criticality settings of the rods for the two concentrations of H_3BO_3 may also be obtained from a calibration of the rods based on period runs. In dollars, this is given by

$$\Delta \rho / \bar{\beta}_{\text{eff}} \quad (2)$$

Five pairings of criticality data and calculations of $\bar{\beta}_{\text{eff}}$, with an average of $\bar{\beta}_{\text{eff}} = 5.4 \times 10^{-3}$, are presented in Table 100I-III.

This relatively low value of $\bar{\beta}_{\text{eff}}$ is mainly due to the effect of Pu^{239} in the reactor and also to the errors inherent in the method.

Table 100I-III
EMPIRICAL DETERMINATION OF $\bar{\beta}_{\text{eff}}$ FOR LOADING 1A (10/62)

H ₃ BO ₃ Conc., gm/gal	Bank Critical, in.	$\Delta\rho$, * %	$\Delta\rho/\beta$	$\Delta\rho$ (H ₃ BO ₃), %	$\bar{\beta}_{\text{eff}}$
{ 4.85 4.04	21.6 18.73	-- 1.04	-- 1.59	{ -- 0.81	0.0081 1.59 = 5.1 x 10 ⁻³
{ 4.85 2.8	21.6 15.5	-- 2.5	-- 3.82	{ -- 2.05	2.05 3.82 = 5.37 x 10 ⁻³
{ 4.04 2.8	18.73 15.5	-- 1.46	-- 2.23	{ -- 1.24	1.24 2.23 = 5.56 x 10 ⁻³
{ 2.8 2.18	15.5 13.91	-- 0.75	-- 1.145	{ -- 0.62	0.62 1.145 = 5.42 x 10 ⁻³
{ 2.18 0.35	13.91 9.86	-- 2.15	-- 3.283	{ -- 1.83	1.83 3.283 = 5.57 x 10 ⁻³

The average of the above $\bar{\beta}_{\text{eff}}$ value = 5.40 x 10⁻³
average deviation = 0.14

* $\Delta\rho$ was obtained from calibration of rods obtained by Period Runs, where $\Delta\rho$ was calculated for $\beta = 6.55 \times 10^{-3}$.

Disregarding leakage, $\bar{\beta}_{\infty}$ may be calculated for an infinite reactor with EBWR fuel as of October, 1962. For ET elements,

$$\left. \begin{array}{l} \bar{N}^{25} = 0.00008901, \quad \beta^{25} = 0.0065 \\ \bar{N}^{49} = 0.00001526, \quad \beta^{49} = 0.0021 \end{array} \right\} \text{Keepin data}$$

This gives

$$\bar{\beta}_{\infty}(\text{ET}) \simeq 5.77 \times 10^{-3}$$

Without going into an error analysis, the above results are presented to indicate the possibility of obtaining β_{eff} by the method described.

Section V. HYDRAULICS AND HEAT TRANSFER

<u>Title</u>	<u>Page</u>
Temperature Distribution in the EBWR Core (Test Report No. 108A, dated November 18, 1960, by K. Almenas and J. Haugsnes).	375
Preliminary Hydrodynamic Tests on EBWR with Modified Internals (Test Report No. 105, dated August 1, 1961, by M. Petrick and E. A. Spleha. Incorporates "Errata for Test Report No. 105," dated August 16, 1961.)	378
Additional Hydrodynamic Test Data from EBWR at Powers to 60 Mw (Test Report No. 105A, dated October 3, 1962, by M. Petrick)	395
An Explanation of the Behavior of EBWR at the Higher Power Levels (> 65 Mw) (Test Report No. 105B, dated November 29, 1962, by M. Petrick and E. A. Spleha. Incorporates Test Report No. 105B-Addendum, dated December 3, 1962)	398
Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly (Test Report No. 106A, dated December 11, 1961, by G. F. Popper)	404
Further Preliminary Hydrodynamic Data Obtained from an Instrumented Fuel Assembly (Test Report No. 106B, dated January 12, 1962, by G. F. Popper)	414
A Terminal Report on the Hydrodynamic Data Obtained from the EBWR Instrumented Fuel Assemblies (Test Report No. 106C, dated January 31, 1963, by G. F. Popper)	416
Reactor Steam Quality by Radioactive Sodium Tracer (Test Report No. 118, dated March 19, 1962, by W. G. Knapp).	424
EBWR Steam Quality during High Power Operation (Test Report No. 118A, dated January 24, 1963, by W. G. Knapp)	430

TEMPERATURE DISTRIBUTION IN THE EBWR CORE

(Test Report No. 108A, dated November 18, 1960,
by K. Almenas and J. Haugsnes)

SUMMARY

The time rate temperature changes with heating and cooling in the EBWR core were determined. It was established that extensive stratification exists in the vessel and especially in the core. The 100-gpm circulation provided by the cleanup-loop pumps was not capable of removing this stratification in any reasonable amount of time.

EXPERIMENT

The summary of the obtained data is presented on Figs. 108A-1 and 2. The location of the temperature measuring points is given in Fig. 108A-3.

Temperature Distribution

Figure 108A-1 seems to be self-consistent and shows that equalization

Fig. 108A-1. Temperature Distribution in EBWR Core after Non-nuclear Heating

Fig. 108A-2

Thermocouple Temperature Readings

of temperature, though extremely slow, is accomplished eventually. However, it appears that this is as much due to the cooling of the top water layers as to mixing. Also an extreme stratification during heating is demonstrated. Temperature differences of 60°F were present during this relatively slow heating.

Figure 108A-2 contradicts Fig. 108A-1 in that it does not show any tendency towards equalization. Particularly strange is the fact that couples, Nos. 1 and 11, which are located near the bottom of the core and thus should read temperatures close to the temperature of Station 2 actually indicate a temperature 13°C lower. These are couples readings, and as such they are suspect. However, the close agreement of the couples at the top of the core (Nos. 8 and 5) with the temperature of Station 1 seems to indicate a good degree of reliability. Further, the whole thermocouple bank was consistent in that when pulled out, and at room temperature, they indicated the same correct temperature with small variations. Even allowing a conservatively large error of $\pm 5^{\circ}\text{C}$ the couples indicate a surprising degree of stratification in the core.

Cooling by Water Addition

Because of indefinite measurements of the feedwater temperature, the accuracy with which subcooling, by introduction of cold water, can be achieved was not determined conclusively. Indirect measurements indicated that the feedwater used was at $<29^{\circ}\text{C}$. This is quite surprising since the ambient temperature of its storage space was $\sim 20^{\circ}\text{C}$. In any case the following temperature drops were obtained:

theoretical	2.5°C
experimental	2.2°C

Fig. 108A-3. Temperature Sampling Points

CONCLUSIONS

Any future experiments which will require homogeneous temperature or H_3BO_3 distributions will need mixing of the water in the core and positive checks that mixing is complete.

PRELIMINARY HYDRODYNAMIC TESTS ON EBWR WITH MODIFIED INTERNALS

(Test Report No. 105, dated August 1, 1961, by M. Petrick and E. A. Spleha.
Incorporates "Errata for Test Report 105," dated August 16, 1961.)

A series of tests was completed on EBWR, during which the in-reactor instrumentation was thoroughly checked out, and preliminary data was taken on the reactor performance characteristics with the new riser installed. The 32 tests run covered the reactor operating conditions listed in Table 105-I.

Table 105-I

REACTOR OPERATING CONDITIONS DURING PRELIMINARY HYDRODYNAMIC TESTS

Power	Hot Water Level Saturation above Riser, in.	Pressure, psi	Water Injection Point
5	3	600	upper ring
5	↓	↓	lower ring
10	↓	↓	upper ring
10	↓	↓	lower ring
15	↓	↓	upper ring
15	↓	↓	lower ring
20	↓	↓	upper ring
20	↓	↓	lower ring
5	12	↓	upper ring
5	↓	↓	lower ring
10	↓	↓	upper ring
10	↓	↓	lower ring
15	↓	↓	upper ring
15	↓	↓	lower ring
20	↓	↓	upper ring
20	↓	↓	lower ring
5	3	300	upper ring
5	↓	↓	lower ring
10	↓	↓	upper ring
10	↓	↓	lower ring
15	↓	↓	upper ring
15	↓	↓	lower ring
20	↓	↓	upper ring
20	↓	↓	lower ring
5	12	↓	upper ring
5	↓	↓	lower ring
10	↓	↓	upper ring
10	↓	↓	upper ring
15	↓	↓	upper ring
15	↓	↓	lower ring
20	↓	↓	upper ring
20	↓	↓	lower ring

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

The tests were successful in that they demonstrated the workability of the in-reactor instrumentation and provided interesting preliminary data on reactor performance. The location of the probes within the reactor vessel is shown in Fig. 105-I. All but two of the instrument probes performed satisfactorily, as did the instrumented fuel assembly. The two faulty probes were two of the three impact meters for measuring the total recirculation flow rates. They were found to be inoperative due to insufficient cooling. The thermocouples installed within the reactor, after several hours of irradiation, proved to be erratic in operation. The insulation resistance of the couples changed, as well as their calibrations. It was necessary, therefore, to calibrate the couples each day, using the reactor temperature derived from the measured pressure as the base. As a result, some doubt exists concerning the subcooling data taken. This is discussed further in a later section.

The data taken were (1) a series of differential pressure drops in the riser and downcomer from which steam volume fractions were computed, (2) the velocity in the downcomer from which the total recirculation flow rates were computed, (3) subcooling, and (4) interface height. The quantity of steam carryunder was computed from a heat balance on the downcomer using the measured recirculation flow rates obtained from the impact meter and the subcooling. A detailed discussion of the data follows. For convenience, the data is presented and discussed in four sections, namely, Reactor Recirculation Flow Rates, Steam Volume Fractions in Riser and Downcomer, true Interface Height within the reactor, and Steam Carryunder.

REACTOR RECIRCULATION FLOW RATES

The measured recirculation flow rates are plotted as a function of power for water levels of 721 ft-5 in. and 722 ft-2 in., and pressures of 600 and 300 psi in Figs. 105-2 and 3. The downcomer velocity was measured by a Stauschiebe tube located in the northeast sector of the reactor downcomer. The recirculation flow rates are of the magnitude expected based on a previous core analysis which indicated that an approximate doubling of the flow rate could be expected by the addition of the riser. However, the rate of increase of the recirculation velocity with power and decreasing pressure was not as large as expected, based on the aforementioned analysis and previously acquired data from EBWR.^(1,2) A comparison of the measured recirculation flow rates with the data reported by Sutherland^(1,2) is shown

in Fig. 105-4. The data shown represent only the upper injection ring series of runs at a pressure of 600 psi. As can be seen at the lower powers, the expected increase of flow was achieved but the rate of increase of flow with power is much smaller. The lower rate of increase of flow can only be attributed to steam carryunder in the downcomer. This is discussed further in the section on Steam Carryunder.

When the makeup water is injected in the lower feed ring, the reactor appears to behave as if it is in an equilibrium state. The recirculation velocity under the conditions of power, water height, and pressure studied is $\sim 4.7 \times 10^6$ lb/hr (See Figs. 105-2 and 3).

The exception to the fact seems to be the series of runs at 300 psi with an initial water level of 721 ft-5 in. (Fig. 105-3). The equilibrium flow rate corresponds to 1.2 ft/sec in the upper downcomer region. This behavior was not unexpected. With the makeup water injected in the lower feedwater ring, the net driving head of the reactor is essentially

$$L_R (\bar{\rho}_D - \bar{\rho}_R) \quad ,$$

where

$$\bar{\rho}_R = \text{mean density in the riser}$$

$$\bar{\rho}_D = \text{mean density in the downcomer} \quad .$$

Since the voids are not collapsed until they reach the lower injection point below the seven-foot riser, one would expect the reactor to stabilize at a total flow just above the critical value for the initiation of steam carryunder. Once the critical value is passed and steam carryunder commences, small amounts of steam produce large steam volume fractions, since the relative velocity of the two phases is very near zero. This immediately tends to reduce the net driving head by decreasing the density in the downcomer $\bar{\rho}_D$. Thus, the reactor reaches an equilibrium recirculation flow rate. The measured reactor downcomer velocity of 1.2 ft/sec checks very well with laboratory loop data which shows that the threshold velocity for the initiation of carryunder at 600 psi is ~ 1.0 ft/sec.

Steam Volume Fractions in Riser and Downcomer

The steam volume fraction variations in the riser and downcomer are shown in Figs. 105-5 through 10. The riser void fraction steadily increases with reactor power and is rather insensitive to the makeup water injection point and initial liquid levels. For a given power, the steam volume fractions

in the riser are higher for the 300-psi series of runs, see Figs. 105-5 and 8. It can be seen that the 300-psi data fit the 600-psi curve when plotted as $2P_r$, and a smooth extension of the curve is obtained to "40 Mw." This tends to substantiate the opinion that operation of the reactor at a given power at 300 psi is equivalent to operation at approximately twice the power at 600 psi. It is also interesting to note that there is very little difference between the voids measured at the center and edge of the riser (probe positions A and B), which indicates that the differential pressure probes are measuring the average static pressure change and not the true local void fractions. As mentioned previously, when water is injected into the lower feedwater ring, the recirculation flow rate changed but slightly. The reason for this can be seen in Fig. 105-11 which shows the variation of $\alpha_R - \alpha_D$ (a measure of the net driving head) as a function of power. As can be seen, the differential void volume fraction increases very slowly. The small gains in driving head are nullified, however, by the increased flow resistance in the core. The increased flow resistance is caused by the higher steam volume content as the power is increased. Since

$$V = k \sqrt{\frac{\sum \text{driving head}}{\sum \text{flow resistance}}},$$

the recirculation velocity would be expected to rise only very slowly if at all.

The data obtained on the quenching of steam in the downcomer is somewhat muddled by the contradictory results obtained from Probe D which is located in the lower downcomer and Probe C-2 which is located just after the cold makeup water is injected. Probe C-2 indicates that the entrained steam voids are quenched rather rapidly (see Figs. 105-6 and 9, Run C-2). Probe D located even further downstream in a higher velocity sector (see Figs. 105-7 and 10, WCL 721 ft-5 in), in some instances, indicates higher voids than that measured by Probe C-2. It is believed that Probe D is not positioned correctly; that is, it is not located in a constant area sector as it was supposed to be, and therefore is picking up a flow effect. However, the magnitude of the readings obtained indicate that some voids are in the vicinity of Probe D and therefore quenching is apparently not occurring as fast as was hoped for. The quenching problem will be investigated more thoroughly in future tests after the questionable aspects of Probe D are corrected.

INTERFACE HEIGHT

The variation of the height of the water interface as a function of power and water column height is shown in Fig. 105-12. The differential height level (interface height - water column height) is plotted as a function of power in Fig. 105-13. As can be seen, a substantial difference exists due

to the presence of steam voids in the reactor. A linear extrapolation of the data to 100 Mw indicates that the differential height would be ~ 4.5 ft, which is almost equal to the available height above the riser discharge.

The acceptance of such an extrapolation at this time is premature. However, the data seem to indicate that the swelling of the level due to voids may prove to be as serious as previously anticipated. Another interesting characteristic of the steam-water interface is shown in Fig. 105-14, where the difference in interface height at the center of the reactor and the edge of the vessel is plotted as a function of power and water column level. A differential of 9 in. exists for the lower operating level indicating a "hump" above the riser. As can be seen, the hump decreases as the operating water level increases.

Steam Carryunder

The steam carryunder was calculated from the measured recirculation flow rates and subcooling by means of the following heat balance:

$$\frac{W_t}{W_t - W_I} (h_f - h_{in}) - \frac{W_I}{W_t - W_I} (h_f - h_I) + X_D (h_{fg}) = X_R (h_f - h_m) \quad (1)$$

where

W_t = total recirculation flow rate

W_I = ion exchange flow

X_D = quality in downcomer

X_R = quality in riser

h_f = saturation enthalpy

h_m = makeup water enthalpy

h_{fg} = latent heat of vaporization

h_{in} = core inlet enthalpy

h_I = ion exchange enthalpy

It is evident that the quantity X_D/X_R , which is equivalent to the percent of steam carried under, is sensitive to W_T and especially to ΔT_{sub} . Therefore, any error in these measurements is strongly reflected in the calculated steam carryunder rate.

Confidence in the measured flow rates using the Stauschiebe tube was gained from the following considerations:

1. Assuming no steam carryunder in Eq. (1), the recirculation flow rate W_t was computed and compared with the hydrodynamic analysis of EBWR. The estimated maximum total flow rate at 100 Mw, assuming no carryunder, is 11×10^6 lb/hr. The calculated heat balance flow rates, in many instances, exceeded this value and a substantial number of the values approached it very closely even though the maximum power level was only 20 Mw. Thus, it appears that the heat balance flows are false and the low subcoolings are a result of carryunder.
2. The rate of change of flow with power, as seen in Fig. 105-15, for a series of runs at 600 psi is much too great to be justified by any reasonable hydrodynamic analysis.
3. The slip ratio in the riser, which was computed with the flow rate measured by the Stauschiebe tube, checks very well with correlations developed from Laboratory loop studies. Had the heat balance flow rate been used, a large number of the slip ratios would have been less than 1, which is a physical impossibility. A comparison of the reactor slip ratios and the laboratory developed correlations is given in Figs. 105-16 and 17.
4. Under reactor conditions, where little or no carryunder is expected, e.g. 5 Mw, and makeup water is injected through the lower ring, the Stauschiebe flow rates check out well with the heat balance flow rate.

Confidence in the subcooling measurements was gained from the fact that the subcooling data derived from two entirely separate and different thermocouple systems (namely, those installed by Heat Engineering and by G. Popper) checked reasonably well (within 1°F) in the majority of the cases, see Table 105-II.

Table 105-II

COMPARISON OF SUBCOOLING MEASUREMENTS

<u>Run No.</u>	<u>Subcooling (G. Popper)</u>	<u>Subcooling (Heat Engineering)</u>
7	1.48	1.38
8	1.14	2.12
9	2.33	3.18
10	2.87	4.36
11	2.23	2.0

Table 105-II (cont'd.)

<u>Run No.</u>	<u>Subcooling (G. Popper)</u>	<u>Subcooling (Heat Engineering)</u>
12	1.02	1.6
13	1.12	1.15
14	0.51	0.8
22	2.19	1.6
23	2.26	1.33
25	2.8	1.6
26	1.4	1.2
27	1.6	1.77
28	1.5	2.02
29	1.9	3.0
30	1.7	2.35
31	1.6	1.65
32	0.95	0.95
34	0.55	1.38
35	0.78	1.81
36	1.04	2.38
37	1.12	1.15
38	1.21	1.49
39	1.12	1.77
40	1.31	2.18
42	0.85	1.35
43	1.02	1.84
44	1.18	2.3
45	1.80	1.38
46	0.98	2.6
47	2.14	3.96
48	1.37	1.9
49	1.46	1.35
50	1.45	1.0

The calculated values of steam carryunder are presented in Figs. 105-18 and 19. At 600 psi with the makeup water injected in the upper ring, the per cent of steam carryunder remains fairly constant, as the power is increased, and is of the order of 25-35%. The downcomer superficial liquid velocity ranged from 1.2 to 1.35 ft/sec. Laboratory studies showed that the per cent of carryunder decreases as power was increased for a fixed recirculation velocity. The reactor did not follow this pattern, since the recirculation velocity increased slightly. Once the threshold velocity for carryunder is passed, substantial increases in carryunder occur with small increases in velocity. In EBWR, the expected decrease of carryunder with increasing power apparently was nullified by the increase in the recirculation velocity. The over-all effect was that the per cent of carryunder remained relatively constant. It does not follow that this behavior will continue with further power increases. This will depend strongly on the effect the loss of subcooling has on the voids in the downcomer and core and hence on recirculation. At this juncture, a discussion of the possible occurrences would be highly

speculative and of little value. It does appear that the steam carryunder (assuming the present data to be correct) will limit the attainment of very high powers.

When the makeup water is injected into the lower feed ring, the percentage of steam carryunder decreases sharply. The recirculation flow rate also drops somewhat, and the downcomer voids increase. The reactor stabilizes just above the threshold velocity for initiation of carryunder as discussed previously.

A comparison of the carryunder data obtained from EBWR with a correlation developed from laboratory loop data is shown in Fig. 105-20. As can be seen, the data points fit the correlation fairly well even though the data were taken from very small geometries (9-in. diameter and less). However, additional data will have to be obtained before the validity of the correlation is completely established, since the parameter range covered in EBWR was rather small. Also, it should be re-emphasized that the carryunder data is only as good as the subcooling and recirculation flow measurements, which are admittedly subject to question in these preliminary runs.

Based on these preliminary tests, the following conclusions have been drawn:

1. The threshold velocity for the initiation of steam carryunder in the downcomer is approximately 1.1 ft/sec. As a result, the effectiveness of the modified reactor internals should be re-evaluated, since it appears that operation at very high powers (greater than 70 Mw) will be difficult to achieve. Ways and means of reducing the total recirculation coolant flow rate should be investigated without impairing cooling of the hot channels.
2. The true "water" interface within the reactor vessel is substantially greater than the indicated water column water level. In order to maintain the mixture interface at a reasonable level and allow sufficient dome height when the approach to 100 Mw is made, it appears that it would be advantageous to remove the top detachable foot of the riser. Also, the water column level should be maintained at a height very near the top of the riser.

Fig. 105-1. Location of Probes within Reactor Vessel

Fig. 105-2. Total Recirculation Flow Rate as a Function of Reactor Power (600 psig)

Fig. 105-3. Total Recirculation Flow Rate as a Function of Reactor Power (300 psig)

Fig. 105-4. Comparison of EBWR Recirculation Flow Rate Data (600 psig)

Fig. 105-5. Riser Steam Volume as a Function of Reactor Power (600 psig)

Fig. 105-6. Downcomer Steam Volume Fraction as a Function of Reactor Power (600 psig)

Fig. 105-7

Downcomer Steam Volume Fraction as a Function of Reactor Power (Run D, 600 psig)

Fig. 105-8

Riser Steam Volume Fraction as a Function of Reactor Power (300 psig)

Fig. 105-9. Downcomer Steam Volume Fraction as a Function of Reactor Power (300 psig)

Fig. 105-10. Downcomer Steam Volume Fraction as a Function of Reactor Power (Run D, 300 psig)

Fig. 105-11. Steam Volume Fraction Differential between Riser and Downcomer as a Function of Reactor Power (600 psig)

Fig. 105-12

Actual Interface Level as a Function of Reactor Power [Level (Transducer)]

Fig. 105-13

Differential Height between the Actual Interface Level and the Water Column Level as a Function of Reactor Power

Fig. 105-14

Fig. 105-15

Interface Height Difference between the Center and Periphery of Vessel as a Function of Reactor Power

Heat Balance Flow Rates as a Function of Reactor Power (600 psig)

Fig. 105-16. Comparison of Laboratory Correlation with Computed Slip Ratios in Riser

Fig. 105-17. Comparison of Laboratory Correlation with Computed Slip Ratios in Riser

Fig. 105-18

Percentage of Steam Carryunder as a Function of Reactor Power (600 psig)

Fig. 105-19

Percentage of Steam Carryunder as a Function of Reactor Power (300 psig)

Fig. 105-20. Comparison of Reactor Data with Laboratory Correlation

REFERENCES

- (1) EBWR Test Report No. 39A, "EBWR Test Reports," ANL-6229 (November, 1960).
- (2) EBWR Test Report No. 39E, "EBWR Test Reports," ANL-6229 (November, 1960).

ADDITIONAL HYDRODYNAMIC TEST DATA FROM EBWR AT POWERS TO 60 Mw

(Test Report No. 105A, dated October 3, 1962, by M. Petrick)

A second series of tests was completed on EBWR at power levels up to 60 Mw. The highlights of the data taken in the second series of tests are presented below. Prior to these tests the top part (1 ft) of the riser was removed to provide additional steam dome volume. The feedwater was injected through the upper feedwater ring during the entire test series and the initial hot saturated water level was 7 in. above the top of the riser.

As in the first series of experiments, the data taken include: (1) a series of differential pressure drops in the riser and downcomer from which steam volume fractions were computed; (2) the velocity head in the downcomer from which the total recirculation flow rates were computed; (3) subcooling; and (4) interface height. The quantity of steam carryunder was computed from a heat balance on the downcomer using the measured recirculation flow rates obtained from the impact meter and the subcooling.

The two impact probes which were faulty in the first series of tests⁽¹⁾ were modified and performed very satisfactorily. As a result more confidence in the total recirculation flow rates was gained since the three impact meters checked one another quite closely. The calibration of the thermocouples remained constant and this also added to the confidence in the data.

Fig. 105A-1

Reactor Subcooling as a Function of Reactor Power

As can be seen in the figure, the subcooling reached a peak of about 2.5°F at 40 Mw and then began to drop with increasing power. This is, of course, not a healthy nor desirable indication since one would expect the subcooling to

REACTOR SUBCOOLING

The reactor subcooling is plotted as a function of reactor power in Fig. 105A-1. The data shown in the figure comprise all the subcooling readings that have been taken to date. The subcooling measurements taken in the present series of tests are denoted by the open and darkened triangular symbols. Again, the open symbols represent data taken on a second independent set of thermocouples by G. Popper. Considering the fact that several degrees were sought out of approximately 500, the reproducibility of the data was quite good and the scatter, not excessive.

increase rather rapidly as the power was raised. At 60 Mw, if there were no carryunder, the subcooling would be about 8.75°F for the measured total flow rates.

RECIRCULATION FLOW RATE

The variation of the total recirculation flow rate with reactor power is shown in Fig. 105A-2. At the higher powers the flow rates are substantially lower than the calculated values, assuming no carryunder. The lower flow rates are apparently directly due to the effect of steam carryunder. An attempt is being made to corroborate this fact by performing extensive thermal hydraulic performance analysis on the reactor.

VAPOR CARRYUNDER

The vapor carryunder occurring in the reactor was calculated, from a heat balance utilizing the measured subcoolings and flow rate, and is shown in Fig. 105A-3. The quantity X_D/X_R on the ordinate is equivalent to the fractional weight per cent of carryunder. In the figure the carryunder remains fairly constant as the power is increased up to 40 Mw and is in the order of 25%. That is, 25%, by weight, of all the steam generated in the core is being entrained in the downcomer. Above 40 Mw, the carryunder increases rapidly. At 60 Mw, 32% of all the steam generated in the core is carried under. Complete loss of subcooling occurs at 34.6% carryunder.

Fig. 105A-2

Total Recirculation Flow Rate as a Function of Reactor Power

Fig. 105A-3

Steam Carryunder as a Function of Reactor Power

This is somewhat startling, but not wholly unexpected. There are a number of data which demonstrate the existence of the vapor carryunder. Perhaps the most vivid demonstration can be obtained by calculating the

recirculation flow rate by means of a heat balance, assuming no carryunder. Many of the flow rates calculated in this way are far beyond the values obtained from hydrodynamic analyses using the most optimistic assumptions. These results are shown in Fig. 105A-4. The calculated value of the total recirculation flow rates is approximately 12 to 14 million lb/hr at 100 Mw. Also, note that the shape of the curve is not what one would expect for a system of this type. That is, the total flow should be leveling off with power instead of rising sharply.

Further corroborating evidence of the excessive carryunder is given in Fig. 105A-5. Here, the average core void volume fraction, obtained from physics analysis⁽²⁾ is plotted versus the reactor power. The estimated void content at 50 Mw, assuming no carryunder, was 12%, which is in sharp contrast to the 20% shown in this figure. The 20% average core void fraction, which corresponds to 5% ρ , is what was actually expected at 100 Mw. Also shown in the figure are the steam volume fractions, measured at the exit by means of the Pottermeter, and, at 29 in. from the bottom of the core, as determined by ion chambers. These data were obtained from the instrumented fuel assembly.

Fig. 105A-4

Heat Balance Flow Rates as a
Function of Reactor Power

Fig. 105A-5

Calculated Steam Volume Fraction
in Core as a Function of Reactor
Power

ACTUAL INTERFACE HEIGHT

The difference between the true, water interface level in the reactor and the level as indicated by the water column is shown in Fig. 105A-6.

Here the differential height level, that is, the interface height minus the water column height, is plotted as the function of the power. As seen in the figure, a substantial difference exists due to the presence of steam voids in the reactor. A linear extrapolation of the data to 100 Mw indicates that the differential height could be around 4-1/2 ft, which is almost equal to the available height of steam dome itself. If the interface height continues to rise at this present level, serious carryover problems may be expected to occur at the very high power levels.

Fig. 105A-6. Height Differential between Actual Interface Level and Water Column Level as a Function of Reactor Power

REFERENCES

- (1) EBWR Test Report No. 105.
- (2) EBWR Test Report No. 100M.

AN EXPLANATION OF THE BEHAVIOR OF EBWR AT THE HIGHER POWER LEVELS (> 65 Mw)

(Test Report No. 105B, dated November 29, 1962, by M. Petrick and E. A. Spleha. Incorporates Test Report No. 105B-Addendum, dated December 3, 1962.)

The purpose of this report is to explain the behavior of EBWR at the higher power levels (> 65 Mw) based on the data and information that have been derived from the in-reactor instrumentation.

The behavior referred to is the apparent ease with which the higher powers are attained as compared to the previous series of power runs up to 65 Mw. Approximately 5% ρ was utilized or tied up in voids when the reactor power was increased from 0 to 65 Mw. In sharp contrast only an additional 1.2% ρ (rough estimate) was required to go from 65 to 94 Mw. In fact, as shown in Table 105B-I after the reactor power was raised from 85 to 94 Mw, the rods had to be reinserted to maintain the higher power level. (At 88.6 Mw the bank was at 38 in. and the central rod was at 40 in., at 90.7 Mw all the rods were at 37.7 in.)

Table 105B-I

SUMMARY OF REACTOR OPERATING DATA

Date and Time	Flowrate, lb x 10 ³ /hr							Rod Position, in.		Total Power, Mw	Boric Acid, gm/gal
	Reboiler Plant		Reboiler Intermediate			Turbine Plant					
	W _{FW} [*]	W _S [†]	W _{S₁} [‡]	W _{S₂} [§]	W _{FWT}	W _S [†]	W _{FW} [*]	No. 1-8	No. 9		
10/31/62											
13:17	146	—	71	66	136	50	60	35	30	67.3	1.8
13:45	165	165	76	71	146	50	64	36	30	72.0	1.8
14:13	175	—	78	75	152	50	66	37	30	74.8	1.8
15:02	184	check	80	74	154	50	66	37	30	76.9	1.8
15:27	194	—	80	76	156	50	66	36	32	77.6	1.8
15:40	193	—	80	77	156	50	66	35.5	32	77.3	1.8
16:00	193	—	78	76	153	50	66	35.2	32	76.9	1.8
16:20	183	—	77	76	153	50	66	34.6	34.6	75.8	1.8
11/1/62											
08:37	165	—	81	76	154	6.5	11	32.5	32.5	56.2	1.2
09:37	195	—	84	78	163	30	43	36.5	30	70.3	1.2
09:55	195	—	86	81	166	31	44	38.0	30	73.3	1.2
10:20	215	—	88	84	172	30	45	35	35	76.75	1.2
10:40	200	—	90	85	170	30	44	34.4	40	74.4	1.2
11:40	159	170	82	77	160	30	40	33	40	68.4	1.0
11:50	158	165	80	74	156	30	38	33	40	66.2	1.0
12:10	203	198	97	89	188	—	—	33	40	64.9	1.0
12:23	191	203	97.5	90	184	—	—	33	40	65.1	1.0
12:45	220	—	—	—	195	—	—	34.4	40	67.7	1.0
13:38	292	265	105	100	200	31	54	38	40	84.6	1.0
								37.8	—	—	—
13:48	320	280	105	98	200	44	79	38	40	94.4	—
								37.8	—	—	—
14:04	240	240	80	80	160	45.5	63	36.7	36.7	82.0	—
14:30	245	230	90	90	180	45.5	63.5	36.7	36.7	80.6	—
15:10	297	250	93	97	196	45	71	37.7	37.7	90.7	—

* Feedwater

† Steam

‡ Steam (Reboiler No. 1)

§ Steam (Reboiler No. 2)

|| Total Feedwater

This behavior strongly suggests that additional reactivity is being gained as the power is increased. Such a gain in reactivity can be obtained from either a sharp increase in the recirculation flow rate or a sharp increase in reactor subcooling - both of which would tend to reduce the rate of increase of the void fraction in the core, or even possibly decrease the void content at the very high power levels. A sharp increase in flow is ruled out by the fact that neither the impact meters nor the instrumented fuel assembly picked up any such variation in flows, see Fig. 105B-1. On the other hand, a sharp increase in subcooling was measured beyond 65 Mw as shown in Fig. 105B-2. The increase was, at first, thought to be due to a sudden unexplainable drop in vapor carryunder. However, this can be ruled out after careful scrutiny of all the carryunder data. In fact, as will be shown in subsequent reports, the carryunder actually continued to increase beyond the 34% which theoretically eliminates all subcooling.

Fig. 105B-1. Total Recirculation Flow Rate as a Function of Reactor Power

Fig. 105B-2. Reactor Subcooling as a Function of Reactor Power (Increase of Subcooling Due to Carryover)

The sharp increase in the subcooling was traced to the occurrence of liquid carryover in the effluent system.

Data on the magnitude of the liquid carryover were obtained from three sources and were found to check fairly well. First, a measure of the carryover can be obtained from the discrepancy between the feedwater and steam discharge flow rates as shown in Table 105B-I. As the power is increased beyond 65 Mw, more feedwater is injected into the reactor than steam is withdrawn. This discrepancy is actually a measure of the liquid carryover rate and, as can be seen in the table, increased sharply with increasing power. A second check on the liquid carryover was obtained by making a heat balance on the secondary heat dissipation system. After establishing the true reactor power, the carryover can be computed by making a heat balance on the primary reactor system. The liquid carryover obtained in this manner checked fairly well with the first technique. A final check was obtained by analyzing steam samples taken from three sampling probes placed at various positions in the reactor steam dome.

The liquid carryover curve, constructed from these data, is shown in Fig. 105B-3. In the figure, the liquid carryover starts at about a power level of 65 Mw, and the rate increases with increasing power. It is from this excess cold feedwater which is injected into the downcomer that the excess subcooling is suddenly gained. Note that the sharp break in the subcooling curve appears at about the same power level where the liquid carryover begins.

Fig. 105B-3. Percent Carryover as a Function of Reactor Power. Comparison of Three Techniques.

The increase in reactor subcooling, due to carryover, can be predicted by making a heat balance on the downcomer. The results of this analysis are shown by the heavy dotted line in Fig. 105B-2. This line was computed from the following heat balance equation:

$$\Delta T_{\text{sub}} = \frac{\left(1 + \frac{X_s}{1 - X_s}\right) (X_R) (h_f - h_m) - \left(\frac{X_D}{X_R}\right) \left(\frac{Q_T}{W_T}\right)}{1.158 \left(1 - \frac{X_D}{X_R}\right)},$$

where

$$X_s = \frac{\text{lb-Water}}{\text{lb-Water} + \text{lb-Steam}}$$

$$X_R = \frac{\text{lb-Steam}}{\text{lb-Total Recirculation Flow Rate}}$$

$$\frac{X_D}{X_R} = \frac{\text{lb-Steam in Downcomer}}{\text{lb-Steam in Riser}}.$$

Since the subcooling is increasing with increasing power, the rate of increase in the core void fraction would, therefore, be expected to decrease. An indication of the change in core void content with increasing power can be gained from a simple heat balance on the core, as follows:

$$Q_C = W_T (\Delta H + X_{h_{fg}})$$

or

$$X = \frac{\left(Q_C / W_T\right) - \Delta H}{h_{fg}},$$

where X is the average mixture quality at the core exit. The core void fraction α_c is roughly proportional to the exit core mixture quality X_e .

Therefore, the change in quality, and hence the change in core void, was computed over the power level intervals 70-80 Mw, 80-90 Mw, and 90-95 Mw by ratioing the above equation. The results are as follows:

$$\frac{\alpha_{cP=80 \text{ Mw}}}{\alpha_{cP=70 \text{ Mw}}} \sim \frac{X_{eP=80 \text{ Mw}}}{X_{eP=70 \text{ Mw}}} = 1.062 \quad ,$$

$$\frac{\alpha_{cP=90 \text{ Mw}}}{\alpha_{cP=80 \text{ Mw}}} \sim \frac{X_{eP=90 \text{ Mw}}}{X_{eP=80 \text{ Mw}}} = 1.02 \quad .$$

$$\frac{\alpha_{cP=95 \text{ Mw}}}{\alpha_{cP=90 \text{ Mw}}} \sim \frac{X_{eP=95 \text{ Mw}}}{X_{eP=90 \text{ Mw}}} = 0.97 \quad .$$

These figures, although not precise, clearly indicate the decreasing rate of core void fraction as the power is increased. Between 80-90 Mw there is virtually no change in exit quality, and, hence, the core void fraction would tend to remain nearly constant. Between 90-95 Mw it appears that the void content actually decreases and, hence, the reactivity would increase. This behavior is corroborated by the steam volume fraction measurements taken in the riser, as shown in Fig. 105B-4, by the static probes, and by the

Fig. 105B-4. Riser Steam Volume Fraction as a Function of Reactor Power

rod movements described in Table 105B-I. From Fig. 105B-4 it can be seen that the void fraction has leveled off and, in fact, does appear to be decreasing beyond 85 Mw as the above listed values predict. This means that as one would attempt to increase power in this region by pulling the rods, the power would keep increasing unless the rods were reinserted. The final rod position would be at a slightly lower level at the higher power than the rod position was at the lower power from which one started. Such a sequence of events actually occurred at the reactor in the high power range as described previously.

In summary, it appears that the reason EBWR is able to go to the higher powers is due almost entirely to the occurrence of, and the magnitude of, the liquid carryover in the effluent steam. A large excess of cold feed-water is being injected into the reactor which, in turn, increases the reactor subcooling and hence reactivity. The excess water is then expelled in the effluent steam. The carryover phenomenon changes the performance characteristics of the reactor radically.

This analysis, or explanation, is based on preliminary data. The data presented are not as yet precise, but the trends are definitely discernable. Further reactor tests are being run to confirm this behavior. See Reference 1 for a terminal report on the thermal and hydraulic performance characteristics of the EBWR between 0-100 Mw.

REFERENCE

- (1) EBWR Test Report No. 106C.

PRELIMINARY HYDRODYNAMIC DATA OBTAINED FROM AN INSTRUMENTED FUEL ASSEMBLY

(Test Report No. 106A, dated December 11, 1961, by G. F. Popper)

SUMMARY

Preliminary hydrodynamic data with the new riser, the one foot riser extension in place, and core loading No. 59 has been obtained from an instrumented fuel assembly located in EBWR core position 101-266. The inlet

water velocity, the exit liquid velocity, and the reactor subcooling were measured while the reactor was operating at the conditions listed in Table 106A-I. The reboiler portion of the EBWR facility was not used during these tests.

Table 106A-I

REACTOR OPERATING CONDITIONS

Power, Mw	Pressure, psig	Saturated Hot Water Level above Riser, in.	Feedwater Injection Ring
2 to 20	600	3	Lower
5 to 20	600	3	Upper
5 to 20	600	12	Lower
5 to 20	600	12	Upper
5 to 20	300	3	Lower
5 to 20	300	3	Upper
5 to 20	300	12	Lower
5 to 20	300	12	Upper

The two feedwater injection rings and the two initial saturated water levels were used to study the effects of steam carryunder on the system performance. The lower pressure of 300 psig was used to simulate higher power density operation.

DESCRIPTION OF INSTRUMENTED FUEL ASSEMBLY

The instrumented fuel assembly, shown in Fig. 106A-1, is designated as Unit II. It is a standard 1.44% enriched thick plate element which has modified inlet and outlet fittings to accept the special turbine type flowmeters that measure the inlet water velocity and exit liquid velocity. This fuel assembly has the original perforated side plates; and it was built primarily to study the effects of the fluid coupling between adjacent fuel elements. Thermocouples are attached at the inlet of the assembly to measure the reactor subcooling.

Fig. 106A-1. EBWR Instrumented Fuel Assembly (Unit II)

INSTRUMENT PERFORMANCE

Turbine Flowmeters

Both the inlet and exit turbine flowmeters have performed extremely well during and after this series of tests. The integrated neutron exposure of the inlet flowmeter and its accumulated hours of operation above 420°F, as of October 3, 1961, are given in Table 106A-II.

Table 106A-II

NEUTRON EXPOSURE AND HOURS OF OPERATION ABOVE 420°F OF THE INLET TURBINE FLOWMETER

Hours of Operation above 420°F	916.5 hr
Integrated Fast Flux (Above 180 kev)	2.07×10^{19} nvt
Integrated Epithermal Flux (0.625 ev to 180 kev)	1.50×10^{19} nvt
Integrated Thermal Flux (Below 0.625 ev)	2.69×10^{19} nvt

The reproducibility of the measured inlet and exit liquid velocity was found to be within $\pm 1\%$.

Thermocouples

The thermocouples, four at the inlet and six in the steam zone, in general, have also performed satisfactorily. It was found necessary, however, to calibrate the thermocouples each day in the following manner. The reactor power was lowered to that value needed to just maintain system pressure. The feedwater was then shut off and system pressure maintained by power adjustment. After a short stabilization period, the temperature at the core inlet became very nearly equal to the saturation temperature or the zero subcooling condition. The "apparent" subcooling signal was measured under these conditions, and used to correct the subcooling measurements made during the test runs for that day. Day to day variations in this correction factor were noted but, in most cases, they could not be attributed to changes in the thermocouple calibrations.

DISCUSSION OF RESULTS

The test data obtained is presented in Table 106A-III, and will be discussed in three separate sections, specifically, Total Recirculation Flow Rate, Inlet Water Velocity, and Exit Liquid Velocity.

Table 106A-III
SUMMARY OF TEST DATA

Run No.	Reactor Power, Mw	Pressure, psig	Initial Saturated Water Level above Riser Extension, in.	Feedwater Injection Ring	Subcooling, °F	Inlet Velocity, ft/sec	Exit Liquid Velocity, ft/sec	Recirculation Flow Rate Calcu- lated Assuming Zero Carryunder x 10 ⁻⁶ lb/hr	Recirculation Flow Rate Measured with Stauschiebe Tube x 10 ⁻⁶ lb/hr
C-3	2.06	600	3	Lower	-	1.64	1.85	-	-
57	4.88	600	3	Lower	-	2.48	3.08	-	-
C-4	5.57	600	3	Lower	-	2.58	3.14	-	-
45	6.51	600	3	Lower	1.80	2.65	3.25	7.78	4.90
7	6.60	600	3	Lower	1.48	2.59	3.23	7.98	4.59
56	9.77	600	3	Lower	-	2.93	3.65	-	-
8	9.89	600	3	Lower	1.14	2.95	3.65	7.67	4.59
46	12.06	600	3	Lower	0.98	3.08	3.82	7.71	4.97
55	13.98	600	3	Lower	-	3.24	4.03	-	-
9	14.78	600	3	Lower	2.33	3.25	4.05	7.70	4.51
	18.03	600	3	Lower	2.14	3.37	4.23	7.52	4.97
10	20.07	600	3	Lower	2.87	3.42	4.35	7.63	4.51
14	6.78	600	3	Upper	0.51	2.65	3.30	8.05	4.75
50	8.65	600	3	Upper	1.45	3.00	-	8.64	5.37
13	10.41	600	3	Upper	1.12	3.14	3.80	9.03	4.83
49	12.50	600	3	Upper	1.46	3.35	-	9.32	5.32
12	15.60	600	3	Upper	1.02	3.52	4.20	10.26	4.90
48	18.52	600	3	Upper	1.37	3.72	-	10.67	5.25
11	19.88	600	3	Upper	2.23	3.77	4.52	11.11	4.97
19	7.58	600	12	Lower	-	2.77	3.36	-	4.68
18	9.91	600	12	Lower	-	3.00	3.67	-	4.63
52	14.75	600	12	Lower	-	3.28	4.08	-	-
20	15.09	600	12	Lower	-	3.29	4.08	-	4.68
53	19.49	600	12	Lower	-	3.47	4.36	-	-
21	20.19	600	12	Lower	-	3.47	4.37	-	4.70
16	7.24	600	12	Upper	-	2.90	3.45	-	4.67
17	9.73	600	12	Upper	-	3.20	3.76	-	4.80
23	15.63	600	12	Upper	2.26	3.58	4.22	10.23	5.02
54	19.67	600	12	Upper	-	3.83	4.53	-	-
22	20.56	600	12	Upper	2.19	3.85	4.55	11.13	4.97
32	6.33	300	3	Lower	0.95	3.02	3.60	8.69	5.29
31	10.56	300	3	Lower	1.56	3.59	4.18	8.35	5.29
30	14.68	300	3	Lower	1.72	3.88	4.53	8.49	5.45
29	18.58	300	3	Lower	1.92	4.05	4.75	7.87	5.45

Table 106A-III (Cont'd.)

Run No.	Reactor Power, Mw	Pressure, psig	Initial Saturated Water Level above Riser Extension, in.	Feedwater Injection Ring	Subcooling, °F	Inlet Velocity, ft/sec	Exit Liquid Velocity, ft/sec	Recirculation Flow Rate Calcu- lated Assuming Zero Carryunder x 10 ⁻⁶ lb/hr	Recirculation Flow Rate Measured with Stauschiebe Tube x 10 ⁻⁶ lb/hr
26	6.04	300	3	Upper	1.37	3.09	3.62	4.55	5.45
25	11.53	300	3	Upper	2.80	3.80	4.32	7.09	5.59
27	15.38	300	3	Upper	1.55	4.07	4.62	9.00	5.66
28	19.61	300	3	Upper	1.53	4.30	4.94	11.73	5.74
42	5.89	300	12	Lower	0.85	3.06	3.61	9.02	4.54
34	6.03	300	12	Lower	0.55	3.12	3.63	8.98	4.63
35	10.30	300	12	Lower	0.78	3.63	4.20	9.14	4.71
43	10.48	300	12	Lower	1.02	3.58	4.20	8.80	4.71
44	14.83	300	12	Lower	1.18	3.85	4.50	8.57	4.75
36	15.06	300	12	Lower	1.04	3.92	4.54	8.57	4.63
37	6.40	300	12	Upper	1.12	3.12	3.67	4.79	4.54
38	11.17	300	12	Upper	1.21	3.72	4.24	7.04	4.71
39	14.50	300	12	Upper	1.12	3.97	4.55	8.75	4.80
40	19.50	300	12	Upper	1.31	4.20	4.90	11.70	4.80

Total Recirculation Flow Rate

The total recirculation flow rate through the core can be determined by a heat balance method only if the steam carryunder in the downcomer is zero. From a heat balance, on the EBWR system, the recirculation rate is:

$$W_t = \frac{W_m(h_f - h_m) + W_I(h_f - h_I)}{(h_f - h_{in}) + X_D h_{fg}} \quad (1)$$

where

W_t is the recirculation flow rate.

h_f is the enthalpy of saturated liquid at system pressure.

h_m is the enthalpy of the mixed feedwater.

h_I is the enthalpy of the ion exchange outlet water.

- h_{in} is the enthalpy of the core inlet water.
 X_D is the steam carryunder in the downcomer.
 h_{fg} is the enthalpy change during vaporization.
 W_m is the feedwater flow rate.
 W_I is the ion exchange flow rate.

If X_D , the steam carryunder, is assumed to be zero in Eq. (1), then the recirculation flow rate can be calculated if subcooling measurements are made. The subcooling measurements are shown in Fig. 106A-2. The recirculation flow rates calculated from the smoothed subcooling curves, using Eq. (1) and assuming X_D to be zero, are shown in Fig. 106A-3 and are compared to the flow rates measured by the Stauschiebe tubes placed in the downcomer (see Reference 1). The recirculation flow rate at 100 Mw, assuming no steam carryunder, is estimated to be 9.5×10^6 lb/hr.⁽²⁾ The flow rates calculated from the heat balance, assuming zero carryunder, in many instances, approached and exceeded this value, even though the reactor power was only 20 Mw. It, therefore, appears that the measured low values of subcooling are a result of carryunder.

The subcooling measurements made with two separate and independent thermocouple systems (those installed in the instrumented fuel assembly and those installed by the Heat Engineering Section in the downcomer) compared reasonably well (within 1°F) for most runs, and gave a further confidence that the heat balance recirculation flow rates were false. A comparison of these subcooling measurements is given in Reference 1.

Inlet Water Velocity

The inlet velocity at core position 101-266 is shown in Fig. 106A-4. The effects of the initial saturated water levels appear to be small, but the expected increase in the inlet velocity when the upper feedwater injection ring is used was quite evident. Another significant fact is also evident from these curves: the rate of increase of the inlet velocity at higher power is much smaller than one would expect. This small rate of increase of flow with power can again be attributed to steam carryunder in the downcomer.

Another interesting aspect of the inlet velocity plots is that when the system is running in an equilibrium state (in other words, when the lower feedwater ring is being used) the 300 psig data fit the 600 psig curve, when plotted at twice the power, and a smooth extension of the curve is obtained to "40 Mw" operation. These curves, therefore, present evidence that operation of the reactor at a given power at 300 psig is equivalent to operation at approximately twice that power at 600 psig.

Fig. 106A-2. Reactor Subcooling as a Function of Reactor Power

*A-CALCULATED FROM SUBCOOLING ASSUMING ZERO STEAM CARRYOVER

+B-MEASURED WITH STAUSCHIEBE TUBE

Fig. 106A-3. Recirculation Flow Rate as a Function of Reactor Power

Fig. 106A-4. Inlet Velocity as a Function Reactor Power

Exit Liquid Velocity

The exit void fraction of a fuel assembly can be calculated from the measured inlet and exit liquid velocities only if flow continuity between the inlet and outlet is assured. The instrumented fuel assembly presently installed in the core has perforated side plates, and there is no assurance of flow continuity through the element; hence, exit void fractions cannot be calculated with any degree of certainty. It is hoped, however, that the effects of fluid coupling between adjacent fuel elements can be studied with this element. This study will have to be made after the second instrumented fuel assembly is installed in the core so that comparisons of the two types of elements can be made. The measured exit liquid velocities are shown in Fig. 106A-5.

Fig. 106A-5. Exit Liquid Velocity as a Function of Reactor Power

CONCLUSIONS

The only conclusion that can be drawn from these preliminary runs is that the effectiveness of the modified reactor internals should be re-evaluated, since it appears that the steam carryunder problem will make operation at very high powers (greater than 70 Mw) difficult to achieve. It must be pointed out, however, that this conclusion is based on data taken with core loading No. 59 which is not the final 100 Mw configuration.

Attachment A

CORE LOADING

Core loading No. 59 consists of 20 spike elements, 54 enriched thin elements, 50 enriched heavy elements, 1 natural thin element, 8 natural heavy elements, 14 dummy elements, and 1 enriched heavy instrumented fuel assembly. A complete description of core loading No. 59 is found in Reference 2.

Attachment B

REACTIVITY EFFECTS (by K. Almenas)

It was found that the instrumented fuel assembly caused the following reactivity changes when it replaced thin enriched element ET-48.

$$\begin{array}{ll} + 0.07\% & \rho \text{ at room temperature} \\ + 0.17\% & \rho \text{ at } 340^{\circ}\text{F} \end{array}$$

Some reactivity increase at higher temperatures is expected. This is due to the fact that the element is a thermal neutron sink and at elevated temperatures, because of the increasing thermal diffusion length, the flux depression in it becomes less. The direction of the reactivity trend is thus correct, but the change in magnitude is probably due largely to measurement errors.

REFERENCES

- (1) EBWR Test Report No. 105.
- (2) EBWR Test Report No. 110L.

FURTHER PRELIMINARY HYDRODYNAMIC DATA OBTAINED FROM
AN INSTRUMENTED FUEL ASSEMBLY

(Test Report No. 106B, dated January 12, 1962, by G. F. Popper)

Preliminary hydrodynamic data was obtained from the instrumented fuel assembly, located in EBWR core position 101-266, see Reference 1, without the riser and one-foot extension in place. The inlet water velocity, the exit liquid velocity, and the reactor subcooling were measured while the reactor was operating at 600 psig (41.8 atm), and at powers ranging from about 5 to 20 Mw. The data obtained is summarized in Table 106B-I.

The total recirculation flow rates through the core were calculated using the subcooling measurements, assuming zero steam carryunder, and are plotted in Fig. 106B-1 as a function of reactor power. The measured inlet and exit liquid velocities are plotted in Figs. 106B-2 and 3 as a function of reactor power. For comparison these figures also show the flow rates and velocities measured after the riser and one-foot extension were installed above the core.

Table 106B-1

SUMMARY OF TEST DATA

Run No.	Reactor Power, Mw	Pressure, psig	Subcooling, °F	Inlet Velocity, ft/sec	Exit Liquid Velocity, ft/sec	Calculated Recirculation Flow Rate Assuming Zero Steam Carryunder $\times 10^{-6}$ lb/hr
1A	5.13	600	4.95	1.75	2.64	1.15
2A	9.78	600	7.00	2.23	3.35	1.51
3A	15.98	600	7.79	2.42	3.67	2.18
4A	21.75	600	8.68	2.64	3.95	2.60

Fig. 106B-1. Recirculation Flow Rate as a Function of Reactor Power

Fig. 106B-2. Inlet Velocity as a Function of Reactor Power

Fig. 106B-3. Exit Liquid Velocity as a Function of Reactor Power

REFERENCE

- (1) EBWR Test Report No. 106A.

A TERMINAL REPORT ON THE HYDRODYNAMIC DATA OBTAINED FROM THE EBWR INSTRUMENTED FUEL ASSEMBLIES

(Test Report No. 106C, dated January 31, 1963, by G. F. Popper)

SUMMARY

Hydrodynamic data was obtained from two instrumented fuel assemblies, located in core positions 101-266 (Instrumented Fuel Assembly - Unit II; I.F.A. II) and 101-54 (Instrumented Fuel Assembly - Unit III; I.F.A. III). Runs 100 through 111 were taken with core loading No. 62.⁽¹⁾ The remaining data were taken with core loading No. 63.⁽²⁾ The new riser was located above the core but the one-foot riser extension was not used during these tests. The inlet water velocity, the exit liquid velocity, the reactor subcooling, and the neutron flux and cadmium ratio 29 in. above the bottom of the fuel plates was measured. The reactor pressure was nominally 600 psig throughout the tests and the upper feedwater injection ring was used exclusively. The re-boiler portion of the EBWR facility was used during a portion of the experiments. The range of reactor thermal power covered was from 2 to 100 Mw. The steam-water interface level was used to study the changes in reactor performance due to the effects of steam carryunder, in the downcomer, and water carryover, into the steam discharge system.

DESCRIPTION OF THE INSTRUMENTED FUEL ASSEMBLIES

The Instrumented Fuel Assembly, designated as Unit II (I.F.A. II), was previously described in Reference 3. Instrumented Fuel Assembly, designated as Unit III (I.F.A. III), is shown in Fig. 106C-1. This assembly is similar to Unit II with the following exceptions: (1) The fuel element was modified by sealing the perforated side plates. This sealing was done to assure flow continuity between the inlet and outlet flowmeter so that the exit void fraction could be calculated. (2) Two miniature in-core ion chambers, one of which is cadmium covered, were fastened 29 in. above the bottom of the fuel plates and between them to measure the thermal neutron flux and cadmium ratio.

Fig. 106C-1. EBWR Instrumented Fuel Assembly (Unit III)

INSTRUMENT PERFORMANCE

Turbine Flowmeters

I.F.A. II was installed in core position 101-266 on June 6, 1961. Both the inlet and outlet turbine flowmeters performed satisfactorily during the entire duration of these tests. During this period, the instrument leads were removed from the vessel penetrations three times and reinstalled. Only minor problems were encountered during these lead handling operations, and the sheathed cables suffered no ill effects. The integrated neutron exposure and the total accumulated hours of operation at 490°F are given in Table 106C-I. The exit flowmeter was in operation for the same length of time. However, the integrated neutron exposure of the exit flowmeter is less than that of the inlet flowmeter.

Table 106C-I

NEUTRON EXPOSURE AND HOURS OF OPERATION
OF THE INLET TURBINE FLOWMETER

Hours of Operation at 490°F	3,991 hr
Integrated Fast Neutron Exposure (above 180 kev)	3.2×10^{20} nvt
Integrated Epithermal Neutron Exposure (0.625 ev to 180 kev)	2.3×10^{20} nvt
Integrated Thermal Neutron Exposure (Below 0.625 ev)	4.1×10^{20} nvt

The reproducibility of the measured inlet and exit velocity was found to be within $\pm 1.5\%$.

On June 5, 1962, I.F.A. III was installed in core position 101-54. The inlet flowmeter coil was found to be defective immediately after the leads were brought through the vessel penetrations. It is believed that the coil or cable was damaged during the installation of the assembly in the core, as the coil was found to be in working order just prior to the installation. The exit flowmeter on this fuel assembly had accumulated 2537.7 hr of operation at 490°F when the tests were terminated and had performed satisfactorily. The reproducibility of the exit flowmeter was also found to be within $\pm 1.5\%$.

Thermocouples

The thermocouples that were used to measure the reactor subcooling were originally installed in the following locations; four at the inlet of each fuel assembly and eight in the steam zone. Four of each of the inlet and steam zone thermocouples were grounded to the sheath, and the remaining eight were insulated from the sheath. The grounded thermocouples, in general, performed poorly. This type of thermocouple would become an open circuit after several thermal cycles from room to operating temperature. It is believed that thermal stresses would break the thermocouple wires at the hot junction. When I.F.A. III was installed, the two grounded inlet thermocouples of I.F.A. II and two each of the grounded and insulated steam zone thermocouples were permanently removed. The reason that they were removed was to make more room available through the vessel penetration for the installation of I.F.A. III leads. Furthermore, four of the six thermocouples were not functioning. Two additional thermocouples, one grounded and one insulated, were installed in the downcomer area at this time. The two insulated thermocouples installed in I.F.A. II became very sensitive to reactor power after they had accumulated about 3,900 hr of operation. They

were located in a neutron flux equivalent to that of the inlet flowmeter. The data taken with these thermocouples were disregarded at this time.

Because the subcooling was small in magnitude, any slight changes in thermocouple calibration were very noticeable. It was, therefore, necessary to calibrate the thermocouples at every possible opportunity. The calibrations were performed by lowering the reactor power, raising the reactor level, and then turning off the reactor feedwater. After a short stabilization period, the temperature at the core inlet would become very nearly equal to the saturation temperature or the zero subcooling condition. The "apparent" subcooling signal was measured under these conditions and was used to correct the subcooling data.

Ion Chambers

Two miniature in-core ion chambers, one of which was covered with 0.020 in. of cadmium, were fastened in diagonally opposite corners of I.F.A. III, 29 in. above the bottom of the fuel plates. Both chambers performed satisfactorily throughout the tests. During the last set of experiments (Runs 201 to 212), the neutron flux level and cadmium ratios changed radically. The cause of these changes cannot be explained at this time.

RESULTS

The results from this series of tests is presented in Table 106C-II and is discussed in five separate sections.

Table 106C-II

SUMMARY OF TEST RESULTS

Run No.	Reactor Power, Mw	Average Steam-Water Interface Level, ft-in.	Reactor Subcooling Average, °F	Inlet Velocity I.F.A. II, ft/sec	Outlet Liquid Velocity I.F.A. II, ft/sec	Outlet Liquid Velocity I.F.A. III, ft/sec	Thermal Neutron Flux 29 in. above Bottom of Fuel Plate x 10 ⁻¹² nv	Cadmium Ratio 29 in. above Bottom of Fuel Plate	Exit Quality I.F.A. III	Exit Void Fraction I.F.A. III	Void Fraction 29 in. above Bottom of Fuel Plate
201	25.1	17-0	1.2	3.92	4.20	5.95	4.3	5.72	0.0335	0.363	0.206
202	35.0	17-2	0.7	3.92	4.75	7.00	7.1	5.47	0.0481	0.467	0.258
203	43.3	17-2	0.3	3.68	5.12	7.60	8.9	5.33	0.0652	0.548	0.272
101	4.1	17-9	1.2	2.67	3.02	3.10	1.4	7.66	0.0066	0.145	0.095
102	9.9	18-0	-	3.27	3.65	4.03	2.6	7.46	0.0145*	0.201	0.122
105	14.9	18-2	1.6	3.63	3.96	4.77	3.9	7.26	0.0201	0.254	0.149
209	25.6	18-0	2.9	4.10	4.32	5.92	4.2	5.72	0.0298	0.328	0.206
208	35.8	18-0	2.5	4.30	4.77	7.05	5.8	5.60	0.0419	0.415	0.227

Table 106C-II (Cont'd.)

Run No.	Reactor Power, Mw	Average Steam-Water Interface Level, ft-in.	Reactor Subcooling Average, °F	Inlet Velocity I.F.A. II, ft/sec	Outlet Liquid Velocity I.F.A. II, ft/sec	Outlet Liquid Velocity I.F.A. III, ft/sec	Thermal Neutron Flux 29 in. above Bottom of Fuel Plate x 10 ⁻¹² nv	Cadmium Ratio 29 in. above Bottom of Fuel Plate	Exit Quality I.F.A. III	Exit Void Fraction I.F.A. III	Void Fraction 29 in. above Bottom of Fuel Plate
114	37.5	18-1	2.2	4.30	5.10	7.30	10.6	6.72	0.0445	0.438	0.223
204	43.5	17-11	1.6	4.28	5.15	7.88	8.7	5.49	0.0535	0.486	0.245
205	51.6	18-3	1.0	4.25	5.65	8.75	10.6	5.38	0.0655	0.547	0.263
206	60.2	18-2	0.7	4.20	6.00	9.50	11.7	5.28	0.0780	0.592	0.271
207	70.9	18-3	0.4	4.10	6.70	10.78	13.4	5.16	0.0949	0.656	0.300
103	19.5	18-5	1.8	3.87	4.20	5.35	5.3	7.15	0.0240	0.294	0.164
106	24.7	18-6	2.4	4.10	4.47	6.05	6.4	7.07	0.0301	0.342	0.175
100	28.5	18-4	2.5	4.18	4.57	6.30	7.1	7.00	0.0337	0.359	0.185
115	44.6	18-2	2.0	4.45	5.40	8.20	11.5	6.62	0.0522	0.486	0.237
110	48.1	18-7	2.3	4.40	5.50	8.30	11.3	6.51	0.0567	0.499	0.252
116	50.4	18-5	1.7	4.45	5.70	8.60	12.4	6.54	0.0598	0.513	0.248
117	56.3	18-7	1.3	4.45	6.00	9.15	13.6	6.39	0.0677	0.547	0.268
118	61.2	18-8	1.1	4.45	6.30	9.60	14.0	6.31	0.0742	0.571	0.279
107	34.5	19-0	3.1	4.37	4.95	7.20	9.0	6.72	0.0385	0.416	0.222
111	59.5	19-0	1.8	4.40	6.15	9.50	14.1	6.35	0.0717	0.570	0.273
108	40.3	20-0	3.7	4.45	5.25	7.72	10.8	6.50	0.0441	0.449	0.253
124	74.8	20-1	—	4.45	—	10.31	—	—	0.0912*	0.608	—
212	78.8	20-1	3.3	4.50	7.00	11.00	14.9	5.22	0.0913	0.628	0.290
141	100	20-1	—	4.57	—	12.81	—	—	0.1095*	0.683	—
119	65.1	20-7	1.3	4.42	6.35	10.00	14.7	6.20	0.0792	0.592	0.293
120	69.9	20-5	0.7	4.35	6.30	10.00	14.7	6.20	0.0876	0.603	0.293
211	76.4	20-4	5.8	4.65	6.95	10.74	14.2	5.27	0.0814	0.603	0.282
121	76.6	20-4	3.8	4.47	6.70	10.66	15.4	6.09	0.0884	0.618	0.308
122	79.6	20-5	5.2	4.55	6.90	11.14	15.4	5.73	0.0883	0.627	0.358
210	73.7	20-11	4.6	4.60	6.68	10.60	13.3	5.25	0.0811	0.602	0.286
125	82.0	20-10	—	4.55	—	10.33	—	—	0.0890*	0.599	—
126	90.1	20-11	—	4.63	—	11.13	—	—	0.0960*	0.624	—
113	2.0	—	0.5	2.16	2.35	2.44	0.83	—	0.0043	0.115	—

* Estimated

Subcooling

It was shown in Reference 3 that the total recirculation flow rate through the core can be determined by a heat balance method only if the steam carryunder in the downcomer is zero. This report also showed that steam carryunder is present in the downcomer and can be calculated if the recirculation rate is measured by a direct technique, such as a Stauschiebe tube, and if the measured subcooling is used. The results of steam carryunder will be reported elsewhere, and the reactor subcooling is plotted in Fig. 106C-2 as information only.

Fig. 106C-2. Reactor Performance as a Function of Reactor Power for Various Steam-Water Interface Levels

Inlet Water Velocity

The inlet water velocity at core position 101-266 is shown in Fig. 106C-2. The effect of steam-water interface level is clearly shown.

Exit Quality

If the thermal reactor power is assumed to be distributed throughout the core as calculated in Reference 4, and does not change with reactor power, then the exit quality of I.F.A. III can be calculated from,

$$x_e = 5.52 \times 10^{-3} \frac{P}{V_{in}} - \frac{\Delta h}{728} \quad (1)$$

where

x_e is the exit quality of the fuel assembly

P is the total reactor power in Mw

V_{in} is the inlet velocity in ft/sec

Δh is the enthalpy change due to subcooling in Btu/lb.

The exit quality is plotted in Fig. 106C-2 for various steam-water interface levels.

Exit Void Fraction

It has been shown⁽⁵⁾ that the exit void fraction can be calculated from,

$$\alpha_e = 1 - \frac{V_{in}(1 - x_e)}{V_{out}} \quad (2)$$

when flow continuity exists between the inlet and outlet of a boiling channel, and

where

α_e is the exit void fraction of the fuel assembly

V_{in} is the inlet water velocity in ft/sec

V_{out} is the exit liquid velocity in ft/sec

x_e is the exit quality of the fuel assembly.

The exit void fraction of I.F.A. III is plotted in Fig. 106C-2 for various steam-water interface levels.

Local Void Fraction

It has also been shown⁽⁵⁾ that the local void fraction can be calculated from,

$$\alpha = 1 - \frac{Cd R - 1}{Cd R_o - 1} \quad (3)$$

where

α is the void fraction 29 in. above the bottom of the fuel plate

$Cd R$ is the cadmium ratio at power

$Cd R_o$ is the cadmium ratio at zero power

The local void fraction is plotted in Fig. 106C-2 for various steam-water interface levels.

DISCUSSION OF RESULTS

The behavior of EBWR at high power has been explained in some detail in Reference 6. The data taken and the curves presented here do, however, substantiate the conclusions arrived at in Reference 6.

REFERENCES

- (1) EBWR Test Report No. 110Q.
- (2) EBWR Test Report No. 110R(A).
- (3) EBWR Test Report No. 106A.
- (4) R. Avery, K. Almenas, C. Carson, H. Iskenderian, and C. Kelber, "EBWR Core 1A Physics Analysis," ANL-6305 (1961).
- (5) G. F. Popper, "Lecture Notes on In-Core Instrumentation for the Measurement of Hydrodynamic Parameters in Water-Cooled Reactors," ANL-6452 (1961).
- (6) EBWR Test Report No. 105B.

REACTOR STEAM QUALITY BY RADIOACTIVE SODIUM TRACER

(Test Report No. 118, dated March 29, 1962, by W. G. Knapp)

INTRODUCTION

This study was conducted to evaluate the effect of a removable riser extending 5 ft-8 in. above the fixed core shroud in the EBWR reactor vessel on the quality of the steam effluent of the reactor at operating powers to 20 Mw. Subsequent to this study, the upper one-foot high section of the removable riser was removed and all future power operation of the reactor is to be with only a 4-ft-8-in. removable riser section.

Under the above conditions the effect of varying several reactor parameters was studied. These parameters included power levels of 5 to 20 Mw, reactor pressures of 600 and 300 psi, and feedwater additions by means of the 3- and 4-in. feedwater rings.

Though original plans were to conduct a single test of a few days duration, the widely variable and erratic results obtained in the first test necessitated (Test A) that the test be repeated. Since the widely variable results were not obtained in the rerun test (Test B), the cause of the abnormal and erratic behavior of the operating reactor during Test A was confirmed to have been a water trap in the pipe vent connection between the top of the water column and the reactor. This connection had been modified after Test A and prior to Test B. Difficulty in control of water level, and unexplainable high level scrams experienced during Test A were thought to be caused by a trap inadvertently formed by a dip in this vent pipe and the modification was made. This was the only modification of consequence made between these tests.

The results of the original test, indicating such excessively abnormal reactor behavior, particularly in view of the results of the test rerun, are of no quantitative value to this report. The cause of this erratic behavior, however, is of interest and is discussed in Attachment A of this report. Otherwise, this report is devoted entirely to the reliable and reproducible quantitative results of the rerun test.

Assuming that sampling and analysis conditions for this test are essentially the same as for the previous test reported in Reference 1, in which steam quality from EBWR in the absence of a removable shroud was studied, it is interesting that the steam quality values obtained in the present test are at least one magnitude higher than for the previous study. A variation in analytical technique, such as failure to correct for background, etc., could be responsible for the lower values previously obtained, or some physical characteristic of the operating reactor, comparable to that encountered in the first test, may have contributed to greater carryover.

PROCEDURE

The method for determining the quality of steam effluent from the reactor was essentially the same as that used in the previous study referenced above. It involved the addition of normal, chemically pure sodium sulfate, as an aqueous solution, to the water of the operating reactor; allowing sufficient time for activation of the sodium to a suitably high level for radioanalysis; simultaneously drawing samples of reactor water and condensed steam; and the determination of relative Na^{24} activity levels in the respective samples.

The predetermined procedure for sodium addition and for sampling of reactor water and steam was followed with no significant deviations. Samples were taken in 60-ml polyethylene bottles. The capacity of these bottles was sufficiently uniform to preclude volumetric measurement of each sample.

Steam samples were drawn, by means of coolers, directly from the steam lines entering and leaving the steam dryer. During the first test, water samples were drawn from a sample line in the reactor purification system. Prior to the rerun test, a sample line was prepared for drawing water directly from the downcomer in the reactor vessel. This change resulted in more reliable and representative water samples.

Subsequent to attainment of a desired stable operating power level in the reactor, 30 to 60 min was allowed before drawing samples. These samples were kept for 24 hr prior to radioanalysis of the 15-hr sodium activity, to allow short-lived activities to decay. The characteristic 1.387 Mev gamma activity of Na^{24} was measured with a 3 x 3 in. NaI (TI) scintillation detector coupled to a 200-channel pulse analyzer. Geometry and analyzer parameters remained constant throughout the analyses, so that direct comparisons of samples for a specific power run could be made.

After correcting for background radiation levels, the activity of the steam, due to Na^{24} , was found to be uniformly very low, and the activity of samples of condensed steam from the effluent of the steam dryer was so low that radioanalysis was of no quantitative value and therefore discontinued.

RESULTS

Of the two tests conducted, the first indicated an abnormal and erratic condition in the reactor which caused very high carryover of water, and consequently of sodium activity, with no particular reproducibility or dependency on power levels. Since this test gave no quantitative information, it is not extensively discussed in this report, but interesting results and conclusions of this test are given in Attachment A of this report.

During the second test, several parameters were varied, including reactor power, pressure, and feedwater injection point. Powers of 5, 10, 15, and 20 Mw were used with pressures of both 300 and 600 psi, and feedwater additions using both the 3-in. feedwater ring and the new 4-in. feedwater ring. The 3-in. ring is located at a 7-ft-8-in. level, about even with the top of the reactor fuel plates, while the 4-in. ring is located at a 15-ft-3/8-in. level, about level with the top of the 4-ft-8-in. removable shroud. Results of these tests are given in Table 118-I.

Table 118-I

RESULTS OF TEST B

Run No.	Power, Mw	Pressure, psi	F.W. Ring, in.	DF _A *	H ₂ O* Entrainment, %	Quality,* %
1**	20.2	600	3	1.4×10^3	0.07	99.93
2	15.4	600	3	1.1×10^3	0.09	99.91
3	10.1	600	3	9×10^2	0.11	99.89
4	5.3	600	3	9×10^2	0.11	99.89
5	10.1	600	3	8×10^2	0.13	99.87
6	5.9	600	4	1×10^4	0.01	99.99
7	20.3	600	4	8×10^3	0.01	99.99
8	15.2	600	4	1.3×10^3	0.08	99.92
9	10.3	600	4	9×10^2	0.11	99.89
10	5.4	600	4	8×10^2	0.12	99.88
11	6.0	600	4	∞	0	100
12	20.0	300	4	2×10^5	0	100
13	15.6	300	4	∞	0	100
14	10.1	300	4	5×10^3	0.02	99.98
15	5.2	300	4	7×10^3	0.01	99.99
16	5.4	300	3	8×10^3	0.01	99.99
17	20.0	300	3	3.3×10^3	0.03	99.97
18	15.0	300	3	7.3×10^3	0.01	99.99
19	10.6	300	3	7×10^3	0.01	99.99
20	5.7	300	3	6.7×10^3	0.01	99.99

* Steam before dryer. There were no samples for DF_B (steam after dryer) as no significant H₂O entrainment could be detected.

** All runs made at water level of 16 ft-10 in. on reactor water sight gauge.

It will be noted in the table that the steam quality is very high, in every case, with no entrainment detected in three runs. The fact that quality does not reflect the operational power level of the reactor, at which the sample was taken, may be due to the fact that the activity of the steam samples (before dryer) was very low (0 to 120 counts/min, after background correction, on the 60-ml samples) for the 1.387-Mev Na²⁴ peak, giving a large degree of relative variation between runs.

Surprisingly, the steam quality obtained in this test are at least one magnitude higher than those previously reported.⁽¹⁾

The activity of the reactor water varied from 6.0×10^5 to 9.6×10^5 counts/min for the 60-ml samples, depending on concentration and burnup of the sodium and activation of the sodium as a function of reactor operating parameters.

Attachment A

TEST A

On June 26, 1961, at 04:00, sodium sulfate was added to the EBWR reactor water. The reactor was operated at 5.0 Mw until about 09:35 at which time 50-ml samples were taken of reactor water from the purification system of condensed steam before the steam dryer, and of condensed steam after the steam dryer. Subsequently, the reactor power was raised to 10.0 Mw, and higher powers, and samples were taken after 30 min of stable operation at each reactor power level. The samples were retained for 24 hr and analyzed for Na^{24} activity in accordance with the procedure given in the body of this report. Samples, both before and after the dryer, had sufficient activity for radioanalysis and are shown in Table 118-II.

Table 118-II

RESULTS OF TEST A

Run No.	Power, Mw	Pressure, psi	Feed-Water Ring, in.	Water Level*, ft-in.	DF _B [‡]	H ₂ O Entrainment, [†] %	Quality A, [‡] %	DF _A [†]	Quality B, [‡] %
1	5.0	600 ↓	3	17-0	58	1.73	98.27	140	99.27
2	10.0		3	17-1	38	2.65	97.35	83	98.79
3	15.7		3	17-1	66	1.51	98.49	120	99.15
4	20.4		3	17-1	100	1.00	99.00	150	99.35
5	21.0		4	17-1	63	1.60	98.40	130	99.24
6	19.4		3	16-5	28	3.57	96.43	300	99.67
7	14.8		3	16-5	130	0.80	99.20	130	99.23
8	11.2		3	16-6	140	0.70	99.30	360	99.72
9	4.7		3	16-6	93	1.07	98.93	310	99.68

* Sight gauge level readings.

‡ Steam after dryer.

† Steam before dryer.

Obviously, the results of this test, if valid, indicated that higher powers would be impractical with the modified EBWR, and that even powers up to 20 Mw gave such low quality steam as to be of little value. Fortunately,

the case of the low quality steam was traced, somewhat 'after the fact' since the cause was not pinpointed until after it had already been corrected, to a condition in the vent line connecting the top of the water column (and sight gauge) to the reactor. This condition contributed a significant and variable error to the visual and instrumental water level measuring devices attached to the water column. The cause of this error was a small (about 6-in. deep) trap in the vent line which occurred at a dip in this line between the water column and the point at which the line entered through the reactor shielding to the reactor vessel. The line had a continuously upward slope from this dip to the point where it entered the reactor vessel, about 2 in. below the head gasket surface of the vessel.

This dip was removed subsequent to Test A, covered in this attachment, but prior to Test B, covered in the body of this report. The results of this modification are quickly apparent by comparison of Tables 118-I and II. The decontamination factor for steam (before the dryer) was increased one or two magnitudes due to the modification.

Several erratic operating conditions, of a minor nature, actually led to the location and removal of the dip in the vent line. Surges of water were encountered in the vapor recovery system and in the stack blower fan. High level scrams were obtained for no apparent reason. On at least two occasions during reactor operation, when it was desired to raise the reactor water level, the feedwater valve had to be opened wide, manually, to accomplish this. This valve is normally only partially opened, and then controlled automatically to maintain a desired level. On these two (or more) occasions, when the desired level was attained, the valve was closed to its previous setting, and finally completely closed, but the water level continued to rise, eventually giving a high level scram. The only methods of observing reactor water level depend on the water column and include the sight gauge, the instrumented water level recorder, and the high and low level scram instrumentation.

It is apparent, from the carryover studies reported here, that the removal of the trap condition had a profound effect on quality. The erratic conditions in the operation of the reactor, as mentioned above, have not been encountered since the vent line was modified. It appears that sufficient water accumulated in this line to lower the water column level below the actual reactor water level an appreciable amount. This would require that the reactor be operating and have a sufficient amount of void fraction, due to steam, to cause the water-steam interface to be at or above the water column-reactor connection at 19 ft-5 in. (This line was checked and determined to be free of obstructions.) This would create a hydraulic block that would prevent venting of steam and non-condensable gases from the water column. The 6 to 12 in. of water in the vent line trap could be increased by condensation during reactor operation. It is possible, that in this quasi-stable state, the water level in the reactor was two or three feet higher than it would be for

normal operating conditions with the same sight glass readings. Consequently the carryover of water (and radioactive sodium) became exceptionally high as shown in Table 118-II. With the exception of Run No. 6, it is interesting that lowering the water level 6-8 in. did significantly improve the quality of the steam effluent.

When additional feedwater was added to raise the reactor water level, it probably upset the quasi-stable condition by forcing the water column level to rise, which in turn spilled some of the water from the vent line into the reactor. This would sufficiently disturb the equilibrium of the water column to reactor water level so that the vent line would continue to empty to the reactor, allowing the water column level to approach the extraordinarily high reactor water level. This is probably what was encountered in those cases where the sight glass level was observed to continue rising after feedwater additions had been completely stopped.

It is interesting that such a small condition as this dip in the vent line could have such a profound effect on the operation of the reactor. Probably at a reactor water level 12 in. lower this would have never occurred, since the 19 ft-5-in. line would have provided unobstructed venting to the water column.

That the condition occurred as it did points up the danger of blocking the vent line in any manner, whether it be a slug of water or otherwise, since all instrumentation is dependent on this line being open at high water levels in the reactor. Water level instrumentation located in the reactor vessel may be desirable for prolonged runs at high levels.

Attachment B

WATER COLUMN AND SIGHT GAUGE WATER LEVELS

During previous operations, prior to 100-Mw conversion, the connections between the water column and the reactor, and between the sight gauge and the water column were limited to two each, one at the top in each case, and one at the bottom.⁽²⁾ This resulted in considerable disagreement between actual reactor water levels, water column levels, and sight gauge levels due to temperature differences in the different columns. The differences as a function of reactor parameters were studied experimentally by Mittle and Theys and reported (with correction factors) in Reference 3. Thus, it appears that actual reactor water levels during previous steam quality tests were 15 ft to 15 ft-6 in. for 20 Mw, and higher for higher powers, until at 40 Mw it was probably 16 ft. The steam line was located at 19 ft-5 in. so that the water level to steam line distance was about 4 ft.

In the present studies the water levels in the reactor, the water column, and the sight gauge should agree fairly closely, since several new

gauge lines have been installed. Hot reactor water can now circulate through these external columns eliminating the difference due to higher water densities at lower temperatures. It is now probable, though not proven experimentally, that the sight gauge level is a correct indication of the reactor water level (assuming absence of voids and foam). The steam exit is at 21 ft-7-1/2 in., or about 4 ft-8 in. above the sight gauge level, in both Tests A and B. The upper surface of the foam may be 2 ft-6 in. or more above the sight gauge level with the present reactor operation parameters.

With the one-foot riser extension removed, the water level in the reactor will be about 16 ft, under normal operating conditions, giving nearly 6-ft distance from the sight gauge level to the steam exit. This may be sufficient for most or all the rise in power to 100 Mw. Previous studies showed an increase of 6 in. from 20 Mw to 40 Mw, probably due to void fraction. Thickness of the foam layer is difficult to predict or measure since the upper surface probably does not exit, per se, nor would the foam density be flat across the surface. The foam, undoubtedly, is subject to local turbulences which could result in a slug of water being entrained in the steam effluent. It would appear that sufficient freeboard may be available to give high steam quality without the demister at high powers. This study will be continued to higher power operations.

REFERENCES

- (1) EBWR Test Report No. 37, "EBWR Test Reports," ANL-6229 (November 1960).
- (1) "EBWR - The Experimental Boiling Water Reactor," ANL-5607 (May 1957).
- (3) EBWR Test Reports Nos. 62, 62A, and 62B, "EBWR Test Reports," ANL-6229 (November 1960).

EBWR STEAM QUALITY DURING HIGH POWER OPERATION

(Test Report No. 118A, dated January 24, 1963, by W. G. Knapp)

INTRODUCTION

The steam quality studies initiated at lower powers⁽¹⁾ on the EBWR were extended and expanded during higher power operation of the reactor to nearly 100 Mw(t). Contrary to the high quality encountered with lower power operation, the quality quickly deteriorates at powers greater than about 65 Mw(t) to a point where the reactor operation more nearly resembles the well known dual cycle system such as Dresden, than it does the direct cycle design. The scope of this report is limited, however, to reporting the results

of radiochemical and chemical tests conducted to establish, with a degree of confidence, the actual carryover of the reactor water into the steam system. The results given here do not always agree closely with values calculated from heat balance observations. The discrepancy may be due, in part, to indeterminates, such as how much water is preferentially shunted through the steam drier, leakage past various valves, and the like.

In any event, the values obtained by comparing the radioactive sodium present in a sample of feedwater (no contamination by purification system water being returned by means of the feedwater line to the reactor) with that present in a sample of the reactor water is considered, by this author, to be, probably, the most accurate and reliable method for the determination of carryover. This does not necessarily represent the quality of the steam at the turbine inlet or the reboiler building inlet, but in current studies, the subcooling effect due to reactor water being circulated and cooled through the condenser, in a "dual-cycle-like" manner, has been of significant importance in the operation of the EBWR at powers of up to 100 Mw.

PROCEDURE

Due to the high levels of Na^{24} radioactivity present in the reactor water during high power operation, probably due to an $\text{Al}(n, \alpha)$ reaction in the crud accumulated on the fuel plates, it was not necessary to add sodium sulfate as in previous studies. In other aspects, however, the method used here paralleled that described in Reference 1, except that samples were drawn from additional locations, and accurate sampling rates were attempted for steam samples. The samples used in these studies were designated throughout with sampling point numbers identifying the location from which the sample was drawn. These numbers will be identified here and used for convenience in the report.

<u>Sampling Point No.</u>	<u>Sample Point</u>
1	ASME steam sample probe in reactor vessel, about 8 in. below lower lip of steam outlet duct opening.
2	Same as above but ~14 in.
3	Same as above but ~20 in.
4	Reactor water from downcomer area - described in Reference 1.
5	ASTM steam sampling probe in 10-in. line from reactor.
6	ASTM steam sampling probe in 6-in. line before the Dryer.

<u>Sampling Point No.</u>	<u>Sample Point</u>
7	Same as Sampling Point 6 but in 6-in. line after the Dryer (Sampling Points 6 and 7 were used for work reported in Reference 1.)
8	Purification system inlet sampling point (limited use for work in Reference 1.)
9	Feedwater sample point after the full-flow feedwater filters but ahead of the addition point for purification system return.

Sample points identified as 1, 2, 3, and 4 terminated in coolers on the heat transfer test control panel on the main floor. No flow rate control was attempted, other than to guarantee sufficient cooling of the sample water.

Sample points 5, 6, and 7 terminated in a manifold and cooler, complete with a flowmeter, on the East Instrument Gallery Platform. Calculations indicated a required flow rate of 48.2 ml/min/Mw(t) of power for the 10-in. sample probe, and 50.7 ml/min/Mw(t) of power for the 6-in. probes. Since the 6-in. probes would not normally be sampling a flow rate greater than that equivalent to 20 Mw(t), the maximum sampling rate would be just over 1 liter/min. The maximum sample rate from the 10-in. line would be nearly 4 liter/min when 80 Mw(t) is being dissipated by the reboiler plant. The maximum sample flow rate that could be attained from all three probes was 1 liter/min, suitable for the 6-in. probes, but too low for reliable sampling of the 10-in. line. Even with only 50%, or even 25%, of the calculated flow rate from the 10-in. probe, the sodium content was comparable to that from the 6-in. (before dryer) probe.

Though the ASTM did list the vertical probe location in a horizontal pipe as preferable, this was not a preferable location in the 6-in. lines when the carryover rate was very high. On several occasions the water content of the sample from the 6-in. probe (after the dryer) was higher than any other point, and was assumed to be due to water flooding the lower orifice in this probe. The probe in the 10-in. line is horizontal, and this is apparently best in the presence of very low quality steam, as in the case of EBWR.

Sample points 8 and 9 were located on the pump floor at or near the sample station. No sample coolers were required.

After some experimentation, it was found that the quality, as determined from the activity of the feedwater (sample point 8), agreed very well with the qualities obtained with the probes in the steam lines (sample points 5 and 6). Since access to the plant was limited to short time intervals, it was found to be much more convenient to use this feedwater sample point for quality determinations.

RESULTS

The qualities determined from samples, from the points previously defined, are given below in tabular form (Table 118A-I). The reactor parameters are also included in the table.

These results generally support the radiochemical determination values.

CONCLUSIONS

There is evidence, from this and other tests, that the carryover of reactor water in the EBWR during power operation in excess of 65 Mw(t) does occur on a large scale. The carryover was shown to be a function of the reactor water level and of reactor power. Further elaboration will be left to other reports covering higher power operation of the EBWR.

It has been shown, both by identification of the gamma energy emitted and the observed half life decay curve of the radioisotope, that Na^{24} used for these studies, though not originating from added sodium ions; was, in fact, that isotope. Samples of crud were examined to rule out preferential deposition or filterability of the Na^{24} . There is no doubt that the Na^{24} observed in the steam samples, or in the feedwater samples was an exact and reliable indication of the amount of entrainment of particulate - including liquid droplet - matter in the steam discharged from the reactor during operation. All sample points were in flowing streams, and they represented, in most cases, an integrated sample of the material flowing through the line at the time of sampling. The only deviation would be those cases where entrained water may have preferentially deposited on inner walls of the piping, which particularly would affect the 10-in. steam probe, or the flooding of the lower portion of a steam line with water, which eventually reached the level of and flowed through the lower orifice of a steam probe, as is believed to have occurred in the 6-in. steam line on the effluent of the steam dryer.

The feedwater sample probably gave the best over-all integrated value of carryover, or, if one prefers, steam quality.

REFERENCE

- (1) EBWR Test Report No. 118.

Table 118A-I
STEAM QUALITY AS A FUNCTION OF REACTOR OPERATING PARAMETERS

Date	Reactor Power, Mw	To Condenser, Mw	To Reboilers, Mw	Water* Level, ft-in.	Quality, %						
					Sampling Point No.						
					1	2	3	5	6	7	9
10/30	65†	~18	~47	15-11	97	86.7	46.3	99.42	99.6	99.89	---
10/31	74.8‡	~20	~55	16-0	75.4	25.0	8.0	75.3	75.4	73.0	---
10/31	75.8§	~22	~54	15-11	69.4	(3.9)	8.8	73.2	79.9	67.1	---
11/12	60	~18	~42	16-1	99.74	87.8	92.2	99.65	99.7	99.59	99.63
11/12	~79	~20	~59	15-11	66.5	16.8	10.0	---	---	---	78.3
11/14	~74	~21	~53	15-5	---	---	---	---	---	---	99.78
11/15	79.8	~22	~58	15-3	---	---	---	---	---	---	96.9
11/15	96.9	~28	~68	15-8	---	---	---	---	---	---	77.4

The two sets of samples taken on 10/31 were sent to the Chemistry Division for accurate analysis of the boric acid content. When calculations were made comparable to the Na²⁴ quality measurements but using these boric acid concentrations, the following results were obtained:

10/31	74.8‡	~20	~55	16-0	72.7	25.5	7.9	73.4	72.4	69.2	---
10/31	75.8§	~22	~54	15-11	68.3	8.7	8.7	71.2	77.6	65.4	---

* As indicated by the sight gauge on the water column.

† Rods banked at 33 in.

‡ Rods Nos. 1-8 at 37 in.; rod No. 9 at 30 in.

§ Rods banked at 34.6 in.

|| Rods banked at 48 in.

Section VI. COMPONENTS

<u>Title</u>	<u>Page</u>
Volume of Water in the EBWR Reactor (Test Report No. 103, dated May 10, 1960, by C. E. Carson)	437
EBWR Turbine Seal Leak Tests, Preliminary Results (Test Report No. 101A, dated April 12, 1960, by R. J. Gariboldi, D. R. Jacobson, and H. M. Sroka)	439
Relative Effectiveness of Sb-Be Source and Core-Induced Neutrons (Test Report No. 102, dated April 14, 1960, by A. Hirsch)	442
Source Level at Startup Counters and Sensitivity of Counters (Test Report No. 102A, dated April 28, 1960, by A. Hirsch)	444
Instrumentation for Startup (Test Report No. 102B, dated May 5, 1960, by A. Hirsch).	445
Effect of Source Installed in EBWR on November 8, 1960 (Test Report No. 102C, dated November 10, 1960, by A. Hirsch)	447
Frequency Response of EBWR High Flux Trip Circuit (Test Report No. 102D, dated January 31, 1963, by A. Hirsch and W. C. Lipinski) . .	448
EBWR Building Shell Leak Rate Test (of October 1960) (Test Report No. 125, dated May 24, 1962, by E. A. Wimunc)	453
EBWR Building Shell Leak Rate Test (of August 1962) (Test Report No. 125A, dated August 23, 1962, by E. A. Wimunc and V. M. Kolba) . .	455
Radiochemical Analysis of Samples from the EBWR Shock Shield (Test Report No. 104, dated May 12, 1960, by R. J. Armani).	457
EBWR Feedwater Filters - Cleanliness Determination (Test Report No. 116, dated May 19, 1961, by E. Swanson).	459
EBWR Subcooler Cleaning (Test Report No. 117, dated May 24, 1961, by J. Matousek)	462
Cleaning of Vent Condenser (Test Report No. 117A, dated July 13, 1961, by J. Matousek)	464
Performance Investigation of EBWR Purification and Decay Heat Removal Systems (Test Report No. 117B, dated April 3, 1962, by T. L. Kettles)	465

Section VI. COMPONENTS (Cont'd.)

<u>Title</u>	<u>Page</u>
Test of Control Boric Acid Injection System (Test Report No. 109, dated March 8, 1962, by E. A. Wimunc)	471
Test of Control Boric Acid Addition to the EBWR during Power Operation (Test Report No. 109A, dated June 19, 1962, by W. G. Knapp. Includes Memo, "Boric Acid Concentration in EBWR Condenser Hotwell Prior to and during Reactor Operation," dated April 3, 1962, by W. G. Knapp.)	474
Safety Aspects of Operating Borated EBWR (Test Report No. 120, dated January 15, 1962, by K. Almenas, N. Suda, and N. Ito).	478
Heat Balance Data on EBWR 100-Mw Plant (Test Report No. 122, dated January 28, 1963, by E. Martinec).	487
Radiation Levels in Reboiler Building and Sub-Reactor Room (Test Report No. 126, dated November 27, 1962, by E. L. Martinec)	504
Short Duration Test of Demister Material (Test Report No. 111, dated February 13, 1962, by V. M. Kolba).	509
Second Test of Demister Material (Test Report No. 111A, dated July 12, 1962, by V. M. Kolba)	513

VOLUME OF WATER IN THE EBWR REACTOR

(Test Report No. 103, dated May 10, 1960, 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 old feed-water 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 Volumes

In the pressure vessel

Note: (+) indicates a containment of H₂O.
 (-) indicates a displacement of H₂O.

	<u>Vol. (ft³)</u>	<u>Vol. (gals)</u>	
Clean pressure vessel (to height 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 Control rod drive thimbles and housing (net)	5.7	42.6	(+)
Thermal shield	20.45	153.0	(-)
Shock shield (to 12-ft level)	0.2	1.5	(-)
Core grid plate	4.2	31.4	(-)
Core hold-down assy.	0.67	5.0	(-)
2 forced circ. outlet pipes (12-in. dia)	19.3	144.4	(+)
4 forced circ. inlets (6-in. dia)	6.3	<u>47.1</u>	(+)
		3239.1 gal	(+)

<u>External to pressure vessel</u>	<u>Vol. (ft³)</u>	<u>Vol. (gals)</u>	
Water column (to 12-ft level)	2.4	18.0	(+)
input and drain lines	0.3	2.2	(+)
3-in. pipe between valve and old feedwater ring	1.55	11.6	(+)
<u>Shutdown cooling</u>			
2-in. pipes between reactor and purification system (incl. 10 ft of reactor drain line)	2.4	18.0	(+)
1-1/4-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 gal	(+)

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	0.11 gal/element
Neutron source	1.6 gal

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	0.1324 gal/in. (+)
Water column	0.1249 gal/in. (+)
4-in. pipe between startup heater and steam drier	0.0552 gal/in. (+)

Adding these changes algebraically results in a net change of 24.0 gal/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:

$$H_2O \text{ Vol.} = (3201 + 24 h) \text{ gal; where } h = (\text{water level} - 144) \text{ with} \\ \text{water level in inches.}$$

NOTE

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

EBWR TURBINE SEAL LEAK TESTS, PRELIMINARY RESULTS

(Test Report No. 101A, dated April 12, 1960,
by R. J. Gariboldi, D. R. Jacobson, and H. M. Sroka)

The EBWR turbine, feed pumps, throttle valve, bypass valve and turbine inlet valves employ seals designed especially for D_2O operation. A detailed description of the seals is presented in Reference 1. 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 D_2O reactors, a summary of the data from the completed tests is reported herein.

Summary of Test Results

	<u>Measured[†]</u>	<u>Design⁽¹⁾</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	0.11 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>~0.07</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
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>~0.04</u> scfm	<u>0.20</u> scfm
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</u> lb/mo	
Total	1.3 lb/mo	

[†]scfm corrected to 0°C and 760 mm Hg.

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_2O , approximately 0.7 lb per month of D_2O would be lost to the containment shell and 0.7 lb per month of H_2O would enter the primary system as a contaminant.

6. Moisture recovered from the seals

	<u>Actual</u>	<u>Design⁽¹⁾</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

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 $+40$ to $+60^{\circ}F$. The dew point of the makeup air was measured as $-50^{\circ}F$ and the dew point of the air mixture was $+20$ to $+40^{\circ}F$. 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/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 D₂O 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, ~1 Mw(e)
 - b. Total steam flow rate ~20,000 lb/hr
 - c. Reactor pressure 600 psi
 - d. Seal steam pressure 2 psi

REFERENCE

- (1) "EBWR - The Experimental Boiling Water Reactor," ANL-5607 (May 1957).

RELATIVE EFFECTIVENESS OF Sb-Be SOURCE AND CORE-INDUCED NEUTRONS

(Test Report No. 102, dated April 14, 1960, 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	0.5 mc	64 x 1	170	500
No. 2	USE	0.5 mc	64 x 1	200	500

Data as taken: (Counts reduced to counts/min) shown in Table 102-I.

Table 102-I

COUNTING DATA

Condition	No. 1	No. 2
Source in place	3073 $\pm$ 29	3284 $\pm$ 31
Tool on antimony	3070 $\pm$ 55	3262 $\pm$ 57
Removed antimony	1581 $\pm$ 40	1780 $\pm$ 42
Reinsert antimony	3083 $\pm$ 55	3243 $\pm$ 57
Removed antimony	1566 $\pm$ 40	1647 $\pm$ 41
Reinsert antimony		
Moved source 2/3 out	1466 $\pm$ 27	1533 $\pm$ 28
Moved source full out	1438 $\pm$ 38	1505 $\pm$ 39

Fig. 102-1

Counts Per Minute Versus
Pulse Height Selector

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

Data as taken: (Counts reduced to counts/min.) shown in Table 102-II.

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 hr of decay time.

Table 102-II
COUNTING DATA

Condition	No. 1	No. 2
Source full out	1497 $\pm$ 38	1511 $\pm$ 39
Source raised to top of shroud	1460 $\pm$ 38	1444 $\pm$ 38
Antimony in dummy 0-333 Source		
Be at top of shroud	1408 $\pm$ 37	1504 $\pm$ 39
Source re-established in original position	3150 $\pm$ 40	3269 $\pm$ 41

Counting rate of plant counters is negligible.

SOURCE LEVEL AT STARTUP COUNTERS AND SENSITIVITY OF COUNTERS

(Test Report No. 102A, dated April 28, 1960, by A. Hirsch)

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 (see Table 102A-I).

Table 102A-I
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 counts/min)	1 counts/5 min (0.2 counts/min)
Ratio of sensitivity of $\frac{\text{No. 1}}{\text{H.P.}} = \frac{8950}{2300} = 3.89$ Sensitivity of H.P. counter = 134 counts/min/nv Sensitivity of No. 1 counter = 521 counts/min/nv		

The 4 counts/min of the No. 1 counter with the source removed is certainly on upper limit. Usually the counting rate is less. Using 4 counts/min and the 521 counts/min/nv sensitivity, an upper limit of:

$$\frac{4 \text{ counts/min}}{521 \text{ counts/min/nv}} = 0.0077 \text{ nv in the hole,}$$

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 counts/min
H.P. tube	1594 counts/min

$$\text{Ratio of sensitivity} \quad \frac{4913}{1594} = 3.14$$

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

(Test Report No. 102B, dated May 5, 1960, by A. Hirsch)

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 -67-1/2 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 counts/min/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.

Fig. 102B-1

In-core Chamber Current and $1/\Sigma_{1+2}$ cr Versus Rod Position (Criticals 57-8-2 and 3, April 28, 1960, EBWR 8-Spike Core)

Figure 102B-1 shows the plot of in-core chamber current and $1/\Sigma_{1+2}$ cr over the sum of the counting rates of No. 1 and No. 2 counters versus the rod position. From the curve of $1/\Sigma_{1+2}$ 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-sec 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

$$\frac{0.372 \times 10 \times 10^{-5} \text{ amps}}{4 \times 10^{-14} \text{ amps/nv}} = 9.3 \times 10^8 \text{ nv}$$

Keithley

$$\frac{0.758 \times 10 \times 10^{-10} \text{ amps}}{2 \times 10^{-14} \text{ amps/nv}} = 3.76 \times 10^4 \text{ nv}$$

L & N micro-microammeter

$$\frac{0.55 \times 2 \times 10^{-9} \text{ amps}}{2 \times 10^{-14} \text{ amps/nv}} = 5.5 \times 10^4 \text{ nv}$$

Counter No. 1

$$\frac{745.967 \text{ counts/min}}{521 \text{ counts/min/nv}} = 1.43 \times 10^3 \text{ nv}$$

(The counter has undoubtedly exceeded its limit of resolution.)

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.

EFFECT OF SOURCE INSTALLED IN EBWR ON NOVEMBER 8, 1960

(Test Report No. 102C, dated November 10, 1960, by A. Hirsch)

An antimony γ source was irradiated in CP-5. This source was then put into the EBWR neutron source Be block dummy after removal of the old Sb source.

In-core instruments were used during this change of sources. As a result of jamming of the in-core counters, after the new source was installed, the evaluation of the counts has a wide error. The readings and calculations obtained are included in Table 102C-I.

Table 102C-I

COUNTS, CHAMBER CURRENTS, AND NEUTRON FLUX FOR VARIOUS SOURCES

Source	Plant Counter No. 1, counts/min	Plant Counter No. 2, counts/min	Keithley Linear Flux Current, amp	0-395 In-Core Chamber Current, amp	0-371 In-Core Counter, counts/min
Old Source	2.42 ± 0.04	2.05 ± 0.04	$0.290 \times 3 \times 10^{-12}$	6.55×10^{-9}	2197 ± 7
No Source	1.04 ± 0.1	0.97 ± 0.1	$0.289 \times 3 \times 10^{-12}$	6.50×10^{-9}	217 ± 3.3
New Source	1227 ± 8	1072 ± 7	$0.555 \times 3 \times 10^{-12}$	3.12×10^{-8}	2,280,000 Jamming

Table 102C-1 (Cont'd)

Source	Plant Counter No. 1, counts/min	Plant Counter No. 2, counts/min	Keithley Linear Flux Current, amp	0-395 In-Core Chamber Current, amp	0-371 In-Core Counter, counts/min
Ratios Old Source None	2.33	2.12	1	1	10.1
New Source Old Source	508	524	Meaningless due to in- complete compensation		1040
Flux at Counter No. 1	$\frac{1227 \text{ counts/min}}{521 \text{ counts/min/nv}} = 236 \text{ nv}$				
Flux at Keithley Chamber	$\frac{0.795 \times 10^{-12} \text{ amp}}{2 \times 10^{-14} \text{ amp/nv}} = 40 \text{ nv}$				

CONCLUSIONS

The core is approximately as strong a source as the old Sb-Be source. This is the same conclusion as drawn in Reference 1.

Replacing the old Sb source with the new Sb source affects the plant counters by a factor of approximately 500 and offers a suitable counting rate for startup. Using the counter sensitivity, as reported in Reference 2, 521 counts/min/nv, a flux of 2.36 nv is present in the counter holes.

From the increase in current shown by the Keithley and the chamber sensitivity, a flux of approximately 40 nv can be assigned in the tangential holes. The difference of fluxes between the counter holes and the chamber holes can probably be explained by their relative proximity to the source.

REFERENCES

- (1) EBWR Test Report No. 102.
- (2) EBWR Test Report No. 102A.

FREQUENCY RESPONSE OF EBWR HIGH FLUX TRIP CIRCUIT

(Test Report No. 102D, dated January 31, 1963,
by A. Hirsch and W. C. Lipinski)

During 100-Mw operation of EBWR an apparent power oscillation was observed on the Brush recorder connected to the Linear Flux Channel. There is some doubt as to whether the oscillation did occur in the reactor. It may

be possible that the recorder malfunctioned as the period of time during which the oscillations were observed occurred when the recorder was switched to 1 mm/sec chart speed. The oscillations were observed for approximately 30 sec and the recorder switched back to 10 mm/sec chart speed. After the change, no oscillations of such magnitude were observed.

If the recording were, in fact, a true indication of reactor power oscillation, then the reactor peaked to about 135 Mw. The trip level was set at about 125 Mw. Since no trip occurred, it was thought advisable to look at the frequency response of the Safety Trip Circuits.

The No. 1 Safety Trip Circuit, Model CD-106, Serial 56174, was removed from the control racks and

bench tested. Figure 102D-1 is a plot of the meter reading of the circuit and the helipot setting for "trip" versus the current input. It will be observed that the two curves are straight line functions. The span of the meter reading is not correct, as it just falls outside of the accuracies of the type test instruments and circuit meter used. Since the trip circuit is used as a relative rather than absolute instrument, this is of no consequence. The trip level is also a straight line function but is about 0.03 unit higher than the meter reading. This is a safer condition, meaning that the "trip" would occur at a lower meter reading than the level set in the helipot dial.

Fig. 102D-1

Meter Reading and Helipot Setting
for Trip Versus Current Input

Next, a sine wave generator was added to the current source circuitry. An oscilloscope was used to read the peak voltage excursions of the generator. The d-c or average power component was established by adjusting the current input to give a meter reading of 0.7 unit. Referring again to Fig. 102D-1, it can be seen that a current of 0.76 unit was the d-c level. The trip level pot was set to 0.900 unit which corresponds to a trip with 0.935 unit of current input. Therefore an increase of 0.175 unit

of current would induce a trip. Using various frequencies, the voltage of the sine wave generator was increased until a trip occurred. The ratios of the peak value of the sine wave to the d-c value (0.715 unit) required for trip is plotted in Fig. 102D-2. Within the limits of the measurement, it can be seen that this curve approximates a single time constant response with the break-point at $\omega = 3.3$ rad/sec.

Fig. 102D-2. Ratio of Peak Signal to d-c Signal to Produce Trip Versus Frequency

The basic circuit for the high flux trip circuit is as shown:

102D-SK-1.

C_1 - ionization chamber and cable capacity, approximately 2000 mmf.

R_1 - range resistor. Selected to give $IR_1 = e_1 = 10$ v.

For $I = 10^{-4}$ amp, $R_1 = 10^5 \Omega$.

The transfer function for the input amplifier is:

$$\frac{e_1(s)}{R_1 I(s)} = - \frac{G}{1+G} \left[\frac{1}{1 + S \frac{R_1 C_1}{1+G}} \right]$$

$$\begin{aligned} R_1 C_1 &= 10^5 \times 2000 \times 10^{-12} \\ &= 2 \times 10^{-4} \text{ sec} \\ &= 200 \mu \text{ sec.} \end{aligned}$$

If the input amplifier has $G = |>10|$ then the effective time constant for the amplifier is

$$\frac{200}{1 + |>10|} = |<18| \text{ sec.}$$

Therefore, this time constant does not cause the reduced response.

The transfer function of interest is the one obtained for the filter circuit following the input amplifier:

$$\frac{e_2(s)}{e_1(s)} = \frac{1}{1 + s R_2 C_2} \quad .$$

As designed:

$$R_2 = 2.2 \text{ M}\Omega$$

$$C_2 = 0.1 \mu \text{ f}$$

$$R_2 C_2 = 0.22 \text{ sec and breakpoint } \omega = 4.55 \text{ rad/sec}$$

This is reasonable agreement with the empirical value. The transfer function of the filter is:

$$\frac{e_2(s)}{e_1(s)} = \frac{1}{1 + s(0.22)} \quad .$$

A step input applied to this circuit has the following time response:

$$e_2(t) = e_1 \left[1 - e^{-\frac{t}{0.22}} \right] \quad .$$

For $e_1 = 1$ and a trip setting of $e_2 = 0.9 e_1$, we have:

$$0.9 e_1 = \left[e_1 1 - e^{-\frac{t}{0.22}} \right]$$

$$t = 0.5 \text{ sec}$$

A step change of zero to full scale requires 500 msec for trip action.

Another case to consider is operation at constant average power with a superimposed variation.

102D-SK-2

The signal for $n(t)$ is:

$$e_1(t) = e_0 + \Delta e \sin \omega t$$

If this signal is passed through the R_2C_2 filter, the output of this filter is:

$$e_2 = e_0 + \frac{\Delta e}{\sqrt{1 + (\omega R_2 C_2)^2}} \sin \omega t$$

The EBWR resonance occurs at approximately $\omega = 10 \text{ rad/sec}$. Substituting for ω results in:

$$\begin{aligned} e_2 &\approx e_0 + \left[\frac{1}{\sqrt{1 + (\omega R_2 C_2)^2}} \right] \Delta e \sin \omega t \\ &= e_0 + \left[\frac{1}{\sqrt{1 + (10 \times 0.22)^2}} \right] \Delta e \sin \omega t \\ &= e_0 + (0.414) \Delta e \sin \omega t \end{aligned}$$

At 10 rad/sec, the sinusoidal component is reduced to 41% of its original value.

As an example, consider $n_0 = 100 \text{ Mw}$ and a trip setting of 135 Mw. If a 10 rad/sec sinusoidal variation with a peak value of 35 Mw were present, trip would not occur because the 35-Mw signal peak would be reduced to

$35 \times 41\% = 14.5$ Mw-signal level. The relay amplifier would only see a total peak signal of $100 + 14.5 = 114.5$ Mw. In order for a trip to occur, a sinusoidal variation of

$$\frac{35 \text{ Mw}}{41\%} = 84.5 \text{ Mw}$$

would be needed. The total peak power would be $100 + 84.5 = 184.5$ Mw.

In designing trip circuits, it is not sufficient to simply consider the delay of trip action for step inputs. Where the possibility of spontaneous oscillations exist due to feedback instability, the trip circuits must be designed to reproduce the oscillation signal without attenuation.

The response of the EBWR trip circuit can be improved by making $R_2 = 140 \text{ k}\Omega$. Then:

$$\begin{aligned} R_2 C_2 &= 140 \times 10^3 \times 0.1 \times 10^{-6} \\ &= 14 \text{ msec} \end{aligned}$$

The attenuation of a 10 rad/sec sinusoidal signal will be:

$$\frac{1}{\sqrt{1 + (10 \times 0.014^2)}} = 0.99$$

Under these conditions, the sinusoidal component would have to increase by

$$\frac{1}{0.99} = 1.01$$

or be greater by 1% to initiate a trip. Assuming a trip setting of 135-Mw and 100-Mw average power, then $35 \times 1.01 = 35.35$ Mw or a total peak power of $100 + 35.35 = 135.35$ Mw would be required to initiate a high flux trip.

EBWR BUILDING SHELL LEAK RATE TEST (OF OCTOBER 1960)

(Test Report No. 125, dated May 24, 1962, by E. A. Wimunc)

A leak rate test of the EBWR containment shell was made from 14:00 hr October 12 to 24:00 hr on October 13, 1960. The leakage rate was 946 ft^3 of air per day at STP. The curves of the allowable leakage rate of 1000 ft^3 of air per day at STP and the leakage rate determined from the data obtained during this test are shown on Fig. 125-1.

Fig. 125-1. Results of EBWR Leakage Rate Test

Equipment used for the leak rate test was the same as that used in April of 1959 and reported in Reference 1. Data was processed by the IBM-704.

The factor of 10^3 in the nomenclature of the ordinate of Reference 1 should be omitted. The correct amount of equivalent air at STP should be about 739,000 cu ft. All leak rate tests performed on the EBWR containment shell are based on the equivalent volume of dry air at STP. Inasmuch as the shell is pressurized to ~ 15 psig during these tests, the equivalent volume at STP will be greater than 700,000 ft³.

REFERENCE

- (1) EBWR Test Report No. 78C, "EBWR Test Reports," ANL-6229 (November 1960).

EBWR BUILDING SHELL LEAK RATE TEST (OF AUGUST 1962)

(Test Report No. 125A, dated August 23, 1962,
by E. A. Wimunc and V. M. Kolba)

A building leak rate test was conducted from 13:00 on 8/4/62 to 13:00 on 8/6/62.

Pumpup of the building containment shell was begun on 7/30/62; however, because of severe oil pumping by the compressor, causing excessive oil in the building, pumping was discontinued and the building pressure of ~ 3 psig relieved. A new filter was ordered which arrived 8/1/62 and pumping was again attempted. Again, severe oil fogging of the building occurred; pumping was discontinued and the building depressurized. A new compressor was rented which arrived on 8/3/62. Using a high (85 - 90 lb) back pressure on the compressor, pumping was again started at $\sim 14:00$ on 8/3/62. No oil vapors were noted. Building pressure of 15 psig was obtained and pumping discontinued at 12:00 on 8/4/62. No leaks were noted in the check made 8/4/62 @ 12:00. Readings were begun at 13:00 on 8/4/62 and continued to 13:00 on 8/6/62. However, at $\sim 10:45$ on 8/6/62 a major leak at a conduit in the emergency escape hatch was noted by the sound of escaping air. This leak was repaired. When data was first sent (8/5/62) to the computer, the RE-151 code would not "run" because of "Temperature Too High" stop. This was remedied on 8/7 and 8/8/62 by Miss M. Schlapkohl of AMD who revised the code to permit building temperatures of up to 100°F. A plot of the STP data from RE-151 versus time is shown in Fig. 125A-1. A general fit of the data to a polynomial of degree 1 was run using the RE-167 code. This gave a line having a slope of -1055 cu ft/24 hr. This data included the known time, i.e. from 10:45 to 13:00, of the leak discovered on 8/6/62. Without this leak it appears that the building would be within the allowable

Fig. 125A-1. Results of EBWR Containment Shell Leak Rate Test

1000 cu ft/day loss rate. The building test was, therefore, considered to be acceptable, and the testing was discontinued.

Date was taken every 15 minutes. After each hourly reading the ice bath was checked. This permitted the cold junction to stabilize again prior to the quarter hour reading.

RADIOCHEMICAL ANALYSIS OF SAMPLES FROM THE EBWR SHOCK SHIELD

(Test Report No. 104, dated May 12, 1960, by R. J. Armani)

Two samples were submitted from the EBWR shock shield for radiochemical analysis. Approximately 0.007 in. was ground from the top surface and 0.007 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. The isotopes 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/gm of iron, and for the chips sample, 1306 d/s/gm of iron. The specific activity on July 15, 1959, for the grindings sample was 4165 d/s/gm of iron and 5627 d/s/gm of iron for the chips samples. The results of the radiochemical separation are given in Table 104-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 104-I
RADIOCHEMICAL SEPARATION DATA

<u>Isotope</u>	<u>Grindings Samples</u>	<u>Chips Sample</u>
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 104-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 $\pm 5\%$ at best. So the indicated values in Table 104-I could easily vary $\pm 5\%$.

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 Co⁵⁸ and Co⁶⁰. The activity ratio of Co⁵⁸ to Co⁶⁰ 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% Co⁵⁸ and 10.6% Co⁶⁰. The smear sample was taken and analyzed by personnel of the Industrial Hygiene and Safety Division.

Previous investigations indicate that 90% 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.

EBWR FEEDWATER FILTERS - CLEANLINESS DETERMINATION

(Test Report No. 116, dated May 19, 1961, by E. Swanson)

Fig. 116-1

Orifice Differential Versus Flow

GENERAL

EBWR has four parallel feedwater filters. Each filter has a flow orifice connected to a differential pressure (D.P.) cell to determine the relative flow when more than one filter is in service. Data taken on May 3, 1961 determined the operating characteristics of the system as shown on Figs. 116-1, 2, and 3. The following procedures pertaining to filter cleanliness were prepared as temporary operating instructions.

Filters subjected to greater than 30 psig differential may start to distort the cartridge which impedes performance. More than 50 psi differential may result in damage to the filter assembly.

DETERMINING FILTER CLEANLINESS

1. Be sure that the number of filters in service are such that usable data can be taken (Fig. 116-3).
2. Read the filter differential with the 0 - 25 psig D.P. cell. This reading should be above 6 psi.

3. Read the feedwater flow in the control room.
4. Use Fig. 116-2 (depending on the number of filters in service) to determine over-all cleanliness.

Fig. 116-2. Filter Operation

5. If the differential approaches the dirty filter curve, record the flow through each filter using Fig. 116-1 and the 0 - 150 in. H_2O D.P. cell reading. These readings should be above 6 in. H_2O .
6. If the individual filter flows vary by more than 15% change the cartridges of the filter with the lowest flow.

Fig. 116-3. Number of Filters Versus Total Feedwater Flow

NOTE 1

Do not allow the filter differential to exceed 17 psi when 4 filters are in service. When the dirtiest filter is removed from service, to change the cartridges, the differential across the other filters will be to approximately 30 psi.

NOTE 2

If the filter differential exceeds 25 psi it will be necessary to measure this with a test gage. A new D.P. cell can be purchased if this becomes a problem, which is quite unlikely.

NOTE 3

The present D.P. cell is capable of a maximum of 25 psi across the filters. If it is necessary to operate the filter with a ΔP greater than 25 psi, the D.P. cell must be isolated and the equalizing line open to prevent damage to the cell. Also, removal of one filter when the ΔP over all filters is 17 psi will result, at worst, in a ΔP of 30 psi over the remaining filters. Therefore, when replacing filters, isolate the D.P. cell and open the instrument equalizing line before isolating the filter to be replaced.

EBWR SUBCOOLER CLEANING

(Test Report No. 117, dated May 24, 1961, by J. Matousek)

INTRODUCTION

Because of fouling, the performance of the two subcoolers, of the water purification system, was unsatisfactory. In order to improve the heat transfer characteristics, it was decided to perform a cleaning operation on the shell side (cooling water side) of the subcoolers. Pursuant to this decision subcooler No. 1 was cleaned May 3, 1961 and subcooler No. 2 was cleaned May 4, 1961.

CLEANING SOLUTION

Cleaning Compound

Sulfamic acid (NH_2HSO_3) obtained from the Dearborn Chemical Company under the trade name of "Scale-cleen" was selected as the cleaning agent. The compound as supplied contains an inhibitor (1% MgO), a wetting agent (Sharples PF-2266), and a color indicator.

Concentration Used

Both subcoolers were cleaned with an 8.25% solution of sulfamic acid; this corresponds to 0.75 lb of acid per gal of water.

APPARATUS AND FLOWSHEET

The apparatus for cleaning consisted of an open 40-gal stainless steel mixing tank, a small bronze turbine pump, 1/2-in. brass discharge control valve, and connecting rubber hose. Hose connections were made by slipping the hose over 1/2-in. pipe and securing it with hose clamps.

The solution in the tank was pumped to the drain connection located midway in the subcooler, passed upward through the shell, and removed through the relief valve line. The relief valve had been previously removed in order to provide a connection for the rubber hose conveying the solution back to the mixing tank.

After the cleaning operation was completed, the pump was stopped, and the rubber hose returning to the tank was connected to another hose which discharged to the sump. After neutralization with caustic, the contents of the tank were discharged to the sump via the subcooler.

CLEANING PROCEDURE

The following sequence was used for cleaning:

1. Subcooler shell isolated.
2. Subcooler shell drained.
3. Fittings and hose connections made to subcooler.
4. Solution prepared and heating coils turned on.
5. Pump started and discharge valve opened.
6. Recirculation continued for 1 hour and 20 minutes.
7. Pump stopped and cleaning solution neutralized with caustic.
8. Contents of tank emptied to sump.
9. 30 gal of 1-2% caustic prepared and recirculated through subcooler.
10. Caustic rinse discharged to sump.
11. Subcooler flushed with demineralized water.
12. Subcooler restored to original condition.

Each subcooler was cleaned individually with an 8.25% solution of "Scale-clean" by adding 22.5 lb of the compound to 30 gal of water. (Although the sulfamic acid is convenient to handle in the dry form, it is advisable to wear a dust mask since the fines are excessive. During weighing and mixing operation the acid dust acts as a lung irritant. Since the acid exhibited no heat of solution, it is suggested that water be added to the acid after the weighing to form a slurry which can be added to the mixing tank.)

As soon as the water was added to the mixing tank the 220 v heater coil was turned on. It is estimated that the heater had a capacity of about 5-6 kw. Immediately after the solution was prepared, the pumps were turned on and the solution was recirculated for 1 hour and 20 minutes. During that time the temperature of the solution reached an estimated 100°F. No recording of temperature was made during the cleaning operation.

(The compound contains a color indicator which indicates depletion of the acid. Upon mixing, the solution is purple. Complete depletion is indicated by yellow and partial by light red and orange.) During the cleaning of both units no depletion was noted; the color indicator remaining purple throughout the operation.

After the cleaning cycle was completed, the pump was shut off and the solution in the tank neutralized with caustic flakes. Neutralization was determined with pH paper. After neutralization, the pump was restarted

and the solution piped to the sump via the subcooler. Thereafter, 30 gal of 1-2% solution of caustic was prepared and recirculated through the subcooler for several minutes. The caustic rinse was then discharged to the sump and demineralized water was added to the tank and pumped through the subcooler. The water rinse was continued until the discharge from the hose to the sump showed the water to be neutral.

After the final wash, the hose connections were removed and the subcooler restored to its original condition.

CONCLUSION

Although no visual means was available to ascertain the effectiveness of the cleaning, the dirty condition of the cleaning solution and rinses was indicative of some success. Unfortunately, the subcoolers are of all welded construction and the before and after condition will never be known except as exhibited by heat transfer performance.

Prior to the cleaning operation, the reactor water flow in each subcooler was limited to approximately 2.5 gpm with an exit temperature of 120°F. After cleaning, the No. 1 subcooler was capable of handling up to 8 gpm with an exit temperature of 117°F. Hence, on the basis of performance, the sulfamic acid cleaning was successful.

After the cleaning operation, several steel fittings, the hose, and the brass valve were visually examined. No adverse effect or pitting was noted. Exposure of the metal furnaces and the hose to sulfamic acid for this time and concentration appears safe.

RECOMMENDATIONS

It is recommended that only 20 gal instead of 30 gal of acid solution be prepared for cleaning of the subcoolers. This amount of solution will require only 15 lb of sulfamic acid.

CLEANING OF VENT CONDENSER

(Test Report No. 117A, dated July 13, 1961, by J. Matousek)

On July 6, 1961, an attempt was made to clean the cooling water side (tube side) of the vent condenser. Prior to the cleaning, the vent condenser was examined, and it was noted that the inside surface of the tubes was covered with at least a 1/16 in. deposit. Many outcroppings within the tube were of even greater thickness. Some of the tube openings were almost completely plugged. A sulfamic acid solution (3/4 lb sulfamic/gal water) was recirculated through the vent condenser at about 130°F for one hour. After

recirculation, the solution was neutralized to pH of 7 and dumped. Next, demineralized water was flushed through the condenser for about 5-10 minutes.

Following the one hour cleaning, the vent condenser channel box was removed and the tubes were examined. It was concluded that the one hour cleaning was unsatisfactory and most of the deposit was still intact, even though it was soft.

On July 7, 1961, a second attempt was made to clean the vent condenser except that the cleaning solution was recirculated for six hours at approximately 130°F. The concentration of the solution was the same as used for the first cleaning. The same procedure for neutralization and flushing were used.

After completion of the six hour cleaning operation, the channel box was removed and the vent condenser was examined. Although considerable cleaning had been accomplished, the result was still not satisfactory. The entrances to several tubes were wiped with a cloth for a more minute inspection, and it was discovered that what was believed to be a white scale, during previous examinations (before 1st cleaning), was actually acid attack. Further examination revealed the extent of the attack which, although considerable, was not sufficient to remove the condenser from service. It is the writer's opinion that this condition existed before use of the sulfamic acid cleaning. This opinion is based on the original observation of the tubes before cleaning and the later conclusion that the surface remained unchanged and was not attacked during the sulfamic acid cleaning. Admittedly, the examination was qualitative only, but the original inspection was made difficult by the dirty condition of the condenser.

It is recommended that further acid cleaning, of any sort, be discontinued for the vent condenser. If leaks are encountered in the future, it is suggested that the unit be retubed. Retubing should not be difficult or expensive since only 16 aluminum tubes are installed in the condenser.

PERFORMANCE INVESTIGATION OF EBWR PURIFICATION AND DECAY HEAT REMOVAL SYSTEMS

(Test Report No. 117B, dated April 3, 1962, by T. L. Kettles)

INTRODUCTION

The original EBWR purification and decay heat removal complex consisted of two identical systems each designed to:

1. Remove ionic impurities and particulate matter from reactor water by circulating up to 10 gpm of water from the reactor vessel bottom through a prefilter, mixed bed ion exchanger, and

afterfilter, returning to the reactor vessel through the 3-in. feed-water ring. To prevent ion exchange resin breakdown, the water is cooled to a maximum temperature of 120°F before entering the prefilter by heat removal in a "regenerative cooler" and a "subcooler."

2. Dissipate decay heat from the reactor core by circulating up to 50 gpm of water from the reactor vessel bottom through the subcooler, returning to the reactor vessel through the 3-in. feed-water ring.

Soon after EBWR started up it became apparent that a flow of 10 gpm, below 120°F, could not be attained and that performance was deteriorating with time. When EBWR was shut down for revision to 100-Mw operation in July, 1959, the temperature limitation prevented flow greater than about 4 gpm. This low flow was sufficient to maintain good quality water for 20-Mw operation but might not be sufficient for 100-Mw operation.

During June, 1961, heat balance tests revealed the regenerative coolers performed satisfactorily but the subcoolers did not have their design cooling capacity. During September, 1961, the shell side of the No. 2 subcooler was cut open and steam cleaned. Upon reactor startup in March, 1962, it was determined that cleaning had improved the subcooler capacity but only to 6.5 gpm.

The heat balance tests revealed an additional problem; pressure drop across the prefilter, afterfilter, and related piping limited the circulating pump capacity to 8 gpm. During October, 1961, the No. 2 pump was cut out of the system. A bench test of the pump revealed maximum flow at full-open outlet was 17 gpm compared with the rated capacity of 50 gpm against a head of 110 ft. The low pump capacity had not affected reactor operation at 20 Mw since purification flow had been limited by temperature, and decay heat removal had been controlled sufficiently with the available pump capacity.

However, operation of the purification system was modified to permit boric acid removal in the range to 100-Mw power output. It became necessary for one pump to circulate water through two ion exchange units coupled in series. The additional pressure drop may determine that pump capacity, rather than water temperature, will limit purification flow unless pump performance can be improved.

The above information is amplified under the headings; Regenerative Coolers, Subcoolers, and Circulating Pumps.

REGENERATIVE COOLERS

The regenerative coolers consist of a single pass tube side and single pass shell side (see Reference 1). Water from the reactor passes through

the tubes of the regenerative cooler to the subcooler and returns to the reactor, from the purification system, through the shell side of the regenerative cooler. When the reactor water is circulated for removal of decay heat the regenerative cooler is bypassed.

Thermodynamic data determined during the heat balance tests are tabulated in Table 117B-I. The coefficient of heat transfer, as interpolated to 10 gpm, is revealed to be 32% higher than design which alleviated somewhat the poor performance of the subcooler during purification operation.

SUBCOOLERS

The subcoolers consist of a single pass tube side and a single pass shell side (see Reference 1). Reactor water flows from the regenerative cooler through the tubes of the subcooler to the circulating pump. The reactor water is cooled by cooling tower water passing through the shell side of the subcooler.

Thermodynamic data determined during the heat balance tests are tabulated in Table 117B-II. The coefficient of heat transfer is shown to have been only 30% of the design figure. It was suspected the poor performance was primarily due to sludge or scale deposited by cooling tower water on the shell side; therefore, two 4-in. cleanout connections were installed through the No. 2 subcooler shell.

Inspection revealed the tubes were covered uniformly with a deposit, approximately 1/8-in. thick. The deposit was soft (possibly due to the sulfamic acid treatment described in Reference 2) and was easily removed by flushing with site steam, throttled to about 100 psig, through a copper tubing lance. Heat balance tests revealed the steam cleaning had increased the coefficient of heat transfer by 25% but the new value was still only 37% of the design figure. The actual heat transferred was, however, 72% of rated design because the cooling water was colder and it was maintained at a higher flow rate than design criteria.

Inspection of the subcooler tube side would require that one subcooler head be cut off then rewelded in accordance with ASME code, Section VIII requirements - a major task. However, a dirty tube side is not suspected as the cause of the poor subcooler performance because the turbine condenser steam side, which includes temperature zones in the same range as the subcooler, has presented no fouling problems. It is believed that the subcoolers never did meet design criteria, which conclusion appears substantiated by the fact that their poor performance was noted soon after the purification system was first placed in service, see Reference 3.

Examination of the subcooler physical design reveals a low length-to-diameter ratio, closely packed spiral tubes, and cooling water inlet and outlet

Table 117B-I

HEAT BALANCES ON EBWR PURIFICATION SYSTEM REGENERATIVE COOLER
BEFORE AND AFTER CLEANING SHELL SIDE OF SUBCOOLER

Flow, gpm	Water Temperature, °F				A, ft ²	LMTD, # °F	Q, 10 ³ Btu/(hr)	U, Btu/(hr) (ft ²) (°F)	Comment
	T ₁ [*]	T ₂ [†]	T ₃ [‡]	T ₄ [§]					
10	486	210	110	397	57.5	95	1464	268	Design Rating Interpolated
10	---	---	---	---	↓	--	1480	355	
8	486	218	150	419		73	1126	270	
6	486	210	135	412		83	871	182	
4	486	211	119	386		96	580	105	
10	---	---	---	---	↓	--	1570	355	Interpolated **
8	486	206	129	415		78	1185	264	
6	486	203	118	406		86	895	182	
4	486	207	104	384		100	595	104	

* To regenerative cooler (tube side).

† From regenerative cooler (tube side).

‡ To regenerative cooler (shell side).

§ From regenerative cooler (shell side).

Log mean temperature difference.

**Data obtained after shell side of subcooler was steam cleaned.

Note: See Reference 1 for schematic arrangement and physical description of coolers.

Table 117B-II

HEAT BALANCES ON EBWR PURIFICATION SYSTEM AND DECAY HEAT REMOVAL SUBCOOLERS
BEFORE AND AFTER CLEANING SHELL SIDE

Flow, gpm	Water Temperature, °F				A, ft ²	LMTD, # °F	Q, 10 ³ Btu/hr	U, Btu/(hr) (ft ²) (°F)	Comment
	T ₁ '*	T ₂ '†	T ₃ '‡	T ₄ '§					
10	210	110	85	110	52.8 ↓ ↓	54	498	175	Design Rating Interpolated
10	---	---	---	---		--	310	52	
8	218	150	62	67		116	268	44	
6	210	135	61	66		116	222	36	
4	211	119	62	65		95	182	36	Interpolated ** ** **
10	---	---	--	--		---	358	65	
8	206	129	56	61		104	306	56	
6	203	118	57	61		96	253	50	
4	207	104	56	59		89	206	44	

* To subcooler (tube side).

† From Subcooler (tube side).

‡ To Subcooler (shell side).

§ From subcooler (shell side).

Log mean temperature difference.

**Data obtained after shell side was steam cleaned.

connections both located at the top of the shell. The total arrangement appears conducive to cooling flow short circuits, making it doubtful that a major portion of the tube surface is effective. It is also believed that subcoolers of improved design would be required to attain the original purification heat transfer design criteria.

CIRCULATING PUMPS

These pumps are a centrifugal type, vertically mounted with 5-3/8-in. diameter closed impellers, a 1-1/2-in. pipe size inlet and outlet connections, and rated, on cold water, to provide 50 gpm against a 110-ft head, with a net positive suction head of 5 ft. They are powered with canned rotor motors rated for 3 hp, 3500 rpm, 3 phase, 440 v, and 5.8 amps. The rotor and impeller assembly are designed to float within the housing during operation to minimize friction and wear.

During October, 1959, the stator winding of the No. 1 pump burned out. Upon dismantling, it was determined that the top surfaces of the impeller had been rubbing against the housing cover. Evidently the rotor and impeller assembly was not floating as intended; so, on recommendation of the pump manufacturer, four, 3/16-in. diameter holes were drilled vertically through the outboard bearing to relieve the differential water pressure causing upward thrust. The stator was rewound and the pump returned to service with no further difficulty to date.

During May, 1961, strange noises were heard emanating from the No. 2 pump, and its motor was found to be overheating. Upon dismantling, it was discovered the rotor cladding had split and had been rubbing against the stator can. Whether the cladding developed a leak to cause swelling and interference with the stator can, or whether bearing wear or rotor thermal expansion had produced the interference to cause cladding failure is not known.

The rotor, stator can, wearing ring, inboard bearing, and outboard bearing were replaced, and the pump returned to service. Within a week the stator winding burned out. Upon disassembly it was discovered the impeller top surfaces had been rubbing the housing cover. The impeller was machined to clear the housing cover; the rotor and impeller assembly was balanced statically and dynamically; the stator was rewound; a 1/8-in. air bleed valve was welded to the housing cover to permit priming; and four No. 10-24 holes were tapped through the inboard bearing (rather than the outboard bearing which was relatively inaccessible) to relieve upward thrust. The last modification employed threaded holes to simplify reduction in size should operation prove water circulation through the holes to be excessive.

Before returning the No. 2 pump to service a bench test was run to compare performance with designed rating. With water at 100°F and a net positive suction head of 5 ft, the relation of head to capacity varied almost

linearly from 120 ft at no flow to 23 ft at 17 gpm. It is suspected this poor performance is caused by water from the impeller short circuiting back to the suction inlet due to excessive clearance between impeller and wearing ring. However, the pump was returned to service without further investigation since performance was acceptable for existing requirements.

Since rotation of a canned rotor pump cannot be determined visually, one additional item of information obtained from the bench test is of interest. With the pump outlet closed and correct impeller rotation, line amperage averages 4.9 amps. With pump outlet closed and incorrect impeller rotation, line amperage averages 8.2 amps.

REFERENCES

- (1) "EBWR - Experimental Boiling Water Reactor," ANL-5607, pp. 88-89 (May, 1957).
- (2) EBWR Test Report No. 117.
- (3) EBWR Test Report No. 13A, "EBWR Test Reports," ANL-6229 (November, 1960).

TEST OF CONTROL BORIC ACID INJECTION SYSTEM

(Test Report No. 109, dated March 8, 1962, by E. A. Wimunc)

The main objective of this test was to determine the effectiveness of the control boric acid injection system and the length of time required to poison the reactor after initiating pumping on 2/23/62.

Prior to turning on the H_3BO_3 injection pumps, the condition of the reactor was as follows:

1. Reactor was critical with a control rod bank position of 9.65 in. The Keithley instrument read 10×10^{-10} amp which corresponded to a power level of about 2 kw.
2. Boric acid concentration was 0.42 gm/gal of water in the reactor vessel.
3. Both purification pumps were in operation (total flow of ~ 14 gpm) taking water from the bottom of the reactor, pumping the water thru the boric acid monitor and then back to the reactor thru the 3-in. feedwater ring; the demineralizers were bypassed. Some circulation, and possibly the only circulation, of water in the reactor was due to these pumps.

4. The reactor had not been in operation, since 10/4/61, at 5 Mw except for a number of low power criticals. Hence there was essentially no decay heat available for natural circulation.
5. The volume of water in the pipe between the boric acid pumps and the reactor vessel was 7.67 gal.
6. The 3-in. pipe (7.2 gal) contained boron-free water since 1000 gal of demineralized water had been pumped through this pipe prior to turning on the boric acid injection pumps.
7. Riser was not in the reactor. Water level in the reactor was 716 ft-8 in. (about 3100 gal), 2 in. above the control rod guide structure.

At a time reference of zero, both boric acid injection pumps were turned on simultaneously. The combined capacity of the pumps was 3 gpm and the H_3BO_3 solution contained 181 gm/gallon. The solution was pumped into the 3-in. feedwater ring.

The following was observed:

Time, min-sec

0 - 0	Both pumps turned on simultaneously.
2 - 35	Linear flux level began to drop (see Fig. 109-1). (The 2 min and 35 sec also corresponds to the length of time required to completely displace the clean water from the lines between the injection pumps and the reactor vessel.)
6 - 35	Power of reactor dropped about 3 decades or approximately equivalent to the rods being fully inserted. (Control rods still banked at 9.65 in.)
12 - 35	Both boric acid injection pumps turned off. (If complete mixing is assumed, concentration would be 1.792 gm/gal.) Counter 1 639 and 440 counts/min Counter 2 815 and 729 counts/min
14 - 35	Air was bubbled into reactor to assist mixing. At this moment a very slight rise was noted on the linear flux recorder which returned to a base level within 30 sec (see Fig. 109-1).
20 - 35	Air bubbling stopped. Counter 1 679 and 645 counts/min Counter 2 927 and 965 counts/min

Fig. 109-1. H_3BO_3 Injection - Linear Flux Record

Time, min-sec

23 - 0 Rods lowered from 9.65 in. to "full in" position.
 Counter 1 301 and 277 counts/min
 Counter 2 372 and 398 counts/min

25 - 0 Air bubbled through reactor water to assist mixing.
 Counter 1 264 and 294 counts/min
 Counter 2 372 and 366 counts/min

Both H_3BO_3 injection pumps were turned on for 12 min to increase the H_3BO_3 concentration in the reactor to ~ 4.2 gm/gal.

The chronology of observations indicates that the injection system is highly effective and poisoned the reactor (without the riser) within 7 min.

Effectiveness of the injection system may be altered when the core riser is installed, for a greater volume of water (~ 4100 gal) will be required in the reactor vessel. Some circulation will also be available in the presence of any decay heat after operation of the reactor at power. Reactor power dropped by a factor of at least 500 which was approximately equivalent to having the rods dropped to the "full in" position (although the control rods still remained banked at 9.65 in.).

TEST OF CONTROL BORIC ACID ADDITION TO THE EBWR DURING POWER OPERATION

(Test Report No. 109A, dated June 19, 1962, by W. G. Knapp. Includes Memo, "Boric Acid Concentration in EBWR Condenser Hotwell Prior to and during Reactor Operation," dated April 3, 1962, by W. G. Knapp.)

A previous test was made using boric acid additions via the Chemical Control system to shut down the EBWR from a cold critical operating condition.⁽¹⁾ The object of the test reported here was to determine the effectiveness of this soluble poison addition system for shutting down the reactor during power operation. Of interest was the rate at which the power output of the reactor was reduced during the addition of the soluble poison at a fixed rate.

This test was conducted at the conclusion of several weeks of power operation during which time extensive rod oscillation studies at several power levels had been made. Immediately prior to initiating the soluble poison addition for this test the condition of the reactor was as follows:

1. The EBWR was operating at a power level of 20 Mw. The control rods were positioned in an unusual manner since the normal drive for No. 9 rod was disconnected for the rod oscillation test. Hence,

rod positions were as follows: rods Nos. 1 and 2 at 26.5 in., 3 to 8, inclusive, at 26.0 in., and rod No. 9 at 0 in. The boric acid concentration in the reactor water was 40 ppm. The Keithley linear flux level was 2.5×10^{-5} amp.

2. The No. 2 purification system pump was in operation with all resin units bypassed. The continuous monitor became inoperative, shortly before the test was started, due to electronic failure; however, the conductivity cell, located at the inlet to the purification system, was operative and was used in this test.
3. The soluble poison addition was made to the 3-in. feedwater ring which was used for feedwater flow. The feedwater flow rate was about 61,000 lb/hr.
4. The reactor was loaded with Core 1A and the riser was in place. The water level was at 15 ft-10 in.

A final concentration of about 1000 ppm was desired. The estimated pumping time for a single unit was 63 min. The addition was initiated, using only one pump, and the second pump was started later, when the initial effect of one pump had been observed. The calibrated combined delivery of the two pumps against a 600 psi head was 2.80 gpm of boric acid stock solution with a concentration of 4.8×10^4 ppm.

The following was observed:

Time,* min-sec

0	No. 1 addition pump started.
2-15	Linear flux level shows first definite decrease on linear flux recorder. Figure 109A-1 shows the corresponding log flux recording.
6-0	No. 2 addition pump started.
11-0	Reactor pressure dropped from 600 psig to 592 psig.
14-0	Reactor power estimated to be 8 Mw.
19-0	Reactor pressure increased by manual control to 600 psig.
21-30	Reactor power estimated to be 2 Mw.
30-0	Reactor pressure decreased to 578 psig.
34-30	Both pumps off. The reactor pressure was 561 psig; the linear flux was 4.2×10^{-12} , or essentially at the normal shutdown level.

* Zero time reference here corresponds to 22:00, CST, 4/6/62.

Fig. 109A-1. Water Conductivity and Log Flux Record of Shutdown from 20 Mw with Soluble Poison

Subsequently rods Nos. 1 through 8 were dropped individually with no observable effect on the flux level.

The final concentration of boric acid was determined to be 985 ppm.

The shutdown of the operating reactor with soluble poison occurred smoothly and uneventfully. The rate at which the reactor power was decreased was approximately 0.86 Mw/min, due entirely to the addition of boric acid. This was not a typical core configuration. With a more uniform rod arrangement, that is, all rods at one bank position, the response of the reactor to such an addition might occur at a different rate.

Attachment A

BORIC ACID CONCENTRATION IN EBWR CONDENSER HOTWELL PRIOR TO AND DURING REACTOR OPERATION

Prior to the recent startup of the EBWR, the conductivity of the hotwell water was observed to be unusually high. A sample of the water was taken for titrimetric determination of boric acid. The results of this test indicated a concentration of 300 ppm (1.13 gm/gal).

The source of this boric acid was, at first, thought to be a leaky valve in the line from the boric acid addition pumps to the condenser hotwell. This was dismissed when a study of demineralized water additions to the reactor via the hotwell showed that these had been made subsequent to operation of the boric acid addition pumps (which had been tested under operating conditions), so that any significant quantity of boric acid from this source would have been flushed into the reactor.

A study of the plant piping revealed that a considerable length of pipe from the top of the steam dryer to the steam relief valves would drain into the hotwell. This line was filled solid with water containing about 1060 ppm boric acid (4 gm/gal) during the hydrostatic test. This line was drained by opening a relief valve, allowing the water to flow through a trap to the hotwell. Study of the control room records revealed the increase in conductivity occurred simultaneously with the draining of this line. It is estimated that this line has a volume of 50 to 60 gal. In addition several small lines and traps are also drained to the hotwell so that the total volume may be 70 to 80 gal. This quantity of water containing 1060 ppm of boric acid added to the hotwell, giving 240 to 300 gal final volume, would give a boric acid concentration of 300 ppm, as observed.

This will be a normal condition following a hydrostatic test. Since the total inventory is small, it will not adversely affect the reactor during startup. It will provide an added load for the anion resin removal of boric acid, and will load up about one quarter of a cubic foot of resin. Therefore, if shim

control is not to be used during operation, it would be advantageous to replace the hotwell water, after a hydrostatic test, with demineralized water prior to operation.

The concentration of the boric acid in the condenser hotwell, during reactor power operation, was analyzed and found to agree very well with predicted values. A D.F. of 17 was predicted. A D.F. of 15.2 was found at 20 Mw, 600 psi operation.

REFERENCE

- (1) EBWR Test Report No. 109.

SAFETY ASPECTS OF OPERATING BORATED EBWR

(Test Report No. 120, dated January 15, 1962,
by K. Almenas, N. Suda, and N. Ito)

SUMMARY

The influence of H_3BO_3 in the reactor water on standard EBWR operational procedures (startups, addition of feedwater) was investigated. It was found that, for the range of concentrations used (up to 4 gm H_3BO_3 /gal H_2O), the present operational procedures are entirely safe and no modifications or special precautions, during operation with H_3BO_3 , are necessary. More specifically:

1. The peak power reached during a startup is increased by a factor of ~ 1.6 for an H_3BO_3 concentration of 4 gm/gal. Even for the minimum practical periods (~ 15 sec), this results in a safe peak power of > 2 Mw; and for the fastest possible period (5 sec, which is the limit due to the period scram circuit) the peak power would not exceed 4 Mw.
2. Addition of unborated feedwater prior to a startup will induce power oscillations when natural circulation sets in; however, these oscillations are of an entirely safe magnitude ($\sim 20\%$ of total power) and they die out within ~ 1 min. The peak powers reached do not exceed those of normal equilibrium condition startups.
3. Addition of unborated cold feedwater during operation will produce smooth controllable power increases not exceeding 10 Mw.

The above conclusions apply to H_3BO_3 concentrations of 4 gm/gal and below; for grossly larger concentrations, the tests should be repeated.

EXPERIMENTAL PROCEDURE

The influence of H_3BO_3 in the reactor water on three separate aspects of EBWR operation were investigated, these were:

- A. Equilibrium startups - (reactor at equilibrium in all aspects - temperature and H_3BO_3 concentration - prior to startup).
- B. Startups after cold unborated feedwater addition.
- C. Addition of cold, unborated feedwater to operating reactor.

Test A

The reactor was started up on three different periods (15, 25, and 40 sec) in the H_3BO_3 free and 3.66 gm H_3BO_3 /gal containing case. The purpose of the test was to see what influence the addition of H_3BO_3 has on the shutdown coefficient and the peak power. For ready comparison, the equivalent periods for the borated and unborated startups were within ~ 1 sec of each other.

The startups were conducted in a standard way except that pains were taken to ensure equilibrium conditions in the core and that additional instrumentation was used. The steps were as follows:

1. A bank critical was obtained with reactor power at ~ 10 kw.
2. No. 9 rod was inserted.
3. The control rod bank was raised an amount determined by the required period.
4. No. 9 was brought up flush to the control rod bank and the power allowed to rise and even out by itself. After this step the rods were not moved for ~ 15 -20 min.

Variables recorded were: power, core water temperature, inlet and outlet flows of the instrumented fuel assembly, and pressure.

Test B

The object of this test was to determine whether cold unborated feedwater added to the reactor vessel, just prior to a startup, would settle essentially unmixed in the bottom of the vessel and then enter the core suddenly as natural circulation sets in. Even for small unborated feedwater additions, the worst hypothetical condition (no mixing whatever) results in very sizable positive reactivity surges. For this reason the test was conducted in three successive steps: namely 50, 150, and 240 gal were added

before three different startups. Because of hotwell capacity and high-low level scram limits, the largest addition (240 gal) was a practical maximum for the present EBWR.

After addition of the feedwater, the critical rod position was known only within ~ 0.5 in. (uncertainty due to incomplete mixing); therefore, for the last two tests the criticals were achieved by simply raising the control rod bank gradually until the reactor was on a satisfactory period. So that comparison with the smooth startups would be possible, the three periods used were of similar magnitude namely 15, 28, and 34 sec.

Test C

The purpose of this test was to measure the power surge resulting from the addition of unborated feedwater to an operating borated reactor. The reactor was operated at ~ 5 Mw and a maximum possible quantity (250 gal) of feedwater was added as rapidly as possible. The feedwater valve was opened wide and the addition was stopped when the reactor was just short of being scrammed by the high water level trip.

INSTRUMENTATION

The temperature and flow rates shown in Figs. 120-1 through 4 were measured with the instrumented fuel assembly. The assembly was located in the central portion of the core. The instruments themselves - a

Fig. 120-1. Unborated Cold Feedwater (50 gal) Addition Prior to Startup. Startup Period 15 sec.

Fig. 120-2. Unborated Cold Feedwater (150 gal) Addition Prior to Startup.
Startup Period 28 sec.

Fig. 120-3. Unborated Cold Feedwater (240 gal) Addition Prior to Startup.
Startup Period 34 sec.

Fig. 120-4. Unborated Cold Feedwater (250 gal) Addition during 7-Mw Operation

turbine-type flow meter and chromel-alumel thermocouples - were located at the bottom of the assembly. Thus the temperature of the water entering the core, rather than the core temperature itself, was monitored. This means that a time lag of ~ 10 sec (time required for core water to complete natural circulation loop) exists in the temperature readings. This is illustrated in Fig. 120-1 which shows a temperature rise only after the peak power was reached and after the positive startup reactivity was compensated by a rise in core temperature. The flow meter readings were not linear for the low powers encountered in Test B and can be interpreted only very qualitatively.

DISCUSSION

Equilibrium Startups (A)

Only the general safety aspects of EBWR startup transients are considered here.

Figure 120-5 shows the power traces of the three unborated startups 1A, 2A, and 3A (15, 25, and 40 sec respectively) and Fig. 120-6 represents

Fig. 120-5. EBWR Startup Transients for Various Startup Periods (No H_3BO_3)

Fig. 120-6. EBWR Startup Transients for Various Startup Periods (3.66 gm/gal H_3BO_3)

the equivalent startups (1B, 2B, and 3B) for a reactor containing 3.66 gm/gal of H_3BO_3 . An additional and no less important difference between the two groups of startups is that for the first group (unborated) the control and bank is at 19 in. and for the second, at 39 in. (almost all the way out of the core). As seen, the borated startups (Fig. 120-6) reach higher peak powers. This was to be expected since both the addition of H_3BO_3 and the removal of control rods from the core lower the absolute value of the negative temperature coefficient which, for this range of power transients, provides the dominating shutdown effect. Since the temperature rises required to compensate for the positive reactivity of the startup period are not large (~ 5 to 10°C). One would expect the temperature coefficient to be linear; and therefore, the peak power to be directly proportional to the reactivity to be compensated (which in turn is proportional to the reciprocal period). That this is so is shown in Fig. 102-7 which gives the peak power versus the reciprocal startup period. The SPERT⁽¹⁾ experiments found that this direct proportionality holds also for periods with smaller orders of magnitude (2 to 0.2 sec). One can therefore extrapolate with confidence to the short periods which would not normally be encountered in EBWR operation. Thus, the peak power expected if the reactor were started up on a 5-sec period with a 3.7 gm/gal concentration of H_3BO_3 would be ~ 4 Mw and this could be handled safely. It is concluded therefore that for H_3BO_3 concentrations of 4 gm/gal the fast period trip is presently set at a safe short period (5 sec).

Fig. 120-7. Peak Power Versus Reciprocal Startup Period

Another encouraging aspect of the presented startups is the absence of any power oscillations. This is due to the installed riser which has greatly

improved the hydrodynamic characteristics of the core and to the fact that the reactor was in temperature equilibrium prior to the startups.

Startups After Cold Unborated Feedwater Addition (B)

Power and temperature traces for the three tests (50, 150, and 240 gal) are given on Figs. 120-1 through 3. Oscillations are clearly shown even for the 50-gal test, though their magnitude is quite low ($\sim 20\%$ of total power). A considerable amount of mixing evidently does take place even in the short times allowed in these (5-10 min). A rough estimate of the extent of this mixing can be had from the 50-gal test. In that test, the critical bank position was determined before the feedwater was added, so that the increase in reactivity due to the feedwater addition could be determined from the known rod worth and the startup period. The reactivity in question was $0.09\% \rho$, which means (assuming uniform dilution) that 11 gal of feedwater had entered the core. This is $\sim 23\%$ of the total feedwater added, and since the total volume of the core water is $\sim 8\%$ of the water contained in the reactor vessel, the conclusion is that when cold feedwater is added, it preferentially enters the core. This result is important since it means that the first reactivity change when natural circulation starts is negative, that is, more highly borated water enters the core rather than the converse as was feared. This is born out by Figs. 120-1 through 3, since in all cases the initial peak power does not exceed what would be expected from an equilibrium startup have the given H_3BO_3 concentrations and the given periods. Table 120-I compares the initial power peaks with those of the equilibrium startups.

Table 120-I

COMPARISON OF INITIAL POWER PEAKS WITH THOSE OF EQUILIBRIUM STARTUPS

Startup Type	H_3BO_3 Conc. gm/gal	Peak Powers, Mw
Equilibrium	0	0.78 to 1.1
Feedwater added	1.85	1.45 to 1.6
Equilibrium	3.66	1.2 to 1.85

The results above are not too surprising since the core is situated in the lower third of the reactor vessel, the added colder and more dense unborated feedwater settles preferentially in the lower region of the vessel, thus also in the core.

Temperature and H_3BO_3 concentration gradients thus exist between the lower and upper halves of the reactor vessel water, and the initiation of natural circulation provides the agitation necessary for mixing. As seen from the power and temperature traces of Figs. 120-1 through 3, three, more or less, defined peaks exist, all separated by about the time (10-20 sec) that it takes water to complete a natural circulation loop.

Further comparing these startup tests with the equilibrium startups, a general much more rapid fall of inpower is noted, i.e., for an equilibrium startup (25 sec period, curve 2A, Fig. 120-5) the power drops $\sim 10\%$ 150 sec after the peak power has been reached. For the equivalent (Fig. 120-2) Test C startup, the power drop is seen to be $\sim 60\%$. The cause of this is again the strongly negative reactivity effect arising as the more highly borated water in the upper regions of the vessel (not containing the core) is mixed with the somewhat diluted water in the lower regions of the vessel.

To sum up: unborated feedwater addition does not present any additional hazards, and the power peaks reached during such an equilibrium startup will not appreciably exceed the equilibrium power peak. The general rate of power drop will be higher. As shown in Figs. 120-1 through 3 increasing the volume of the added feedwater does not appreciably increase the magnitude of the power fluctuations.

Feedwater Addition During Operation (C)

Figure 120-4 shows the power and temperature variation resulting from the addition of cold unborated feedwater. As shown, the power increases rather smoothly from ~ 7 to ~ 15 Mw during the addition, while the accompanying temperature dip is $\sim 4^\circ\text{C}$. The flow rate does not show any observable fluctuation even though the power has increased by a factor of 2. If the flowmeter is to be believed in this case, it would mean that most of the power increase goes into the heating up of the additionally subcooled water or is taken up by other fuel elements.

Comparing the obtained power increase with a predicted value of 7 Mw, it is seen that the estimated maximum just matched the actual conditions. The maximum was estimated on the basis of complete stratification of borated and unborated water, a condition which could not very well be matched in practice. The explanation for this is that for the test conditions (cold feedwater being added) the power coefficient is much smaller than for actual operating conditions. That is, in the latter case most of the reactivity compensated is due to the higher equilibrium voids at the higher powers. Here, as shown by the flow rate, the power goes to heat up the more highly subcooled water and the increase in voids is small. Thus, a small reactivity increase would result in an appreciable power increase. Since at least temperature-wise the test conditions were as severe as could be encountered in practice (feedwater was at $\sim 92^\circ\text{F}$ and the feedwater valve was wide open), further reactivity additions would result in higher equilibrium voids and consequently much smaller power rises. From the safety aspect it then can be predicted that if the test were to be repeated at higher H_3BO_3 concentration (i.e., 5 gm/gal) the power rise would not appreciably exceed the present 8 Mw.

REFERENCE

- (1) S. G. Forbes, et al., "Analysis of Self-Shutdown Behavior in the SPERT-I Reactor," IDO-16528, Reactors - General, TID-4500 (14th Ed.), (July 23, 1959).

HEAT BALANCE DATA ON EBWR 100-Mw PLANT

(Test Report No. 122, dated January 28, 1963, by E. Martinec)

INTRODUCTION

Data has been collected on the EBWR 100-Mw plant since its initial stages of operation in July, 1962. This data was used in obtaining a heat balance on all of the heat dissipation equipment that was installed for 100-Mw operation. Had this data not been accumulated, tabulated, and a heat balance calculated therefrom, it is doubtful if meaningful results could have been obtained from the operation of the plant.

The data enabled the calculation of moisture carryover with a fair degree of accuracy. A relationship between water level, carryover, and reactor power was uncovered. The heat dissipation capabilities of various pieces of equipment were also investigated.

HEAT BALANCE DATA

The very first time primary steam was admitted to the primary reboilers was the signal of the start of the shakedown operation of the reboiler plant and eventually reactor operation at 100 Mw. Data collection for heat balance calculation began at the start of the shakedown operation.

At the beginning, when only one reboiler was in operation, or total reactor power was below 50 Mw, there was close agreement in the heat balance data between the primary and intermediate circuits. As the reactor power was increased, discrepancies were noted between primary feedwater flow and primary steam flow. These discrepancies were sufficient to cause argument as to which flow to use for heat balance calculations used in determining reactor power. It was recognized that there was some moisture carryover with the steam but it was not believed to be sufficient to produce the flow measurement differences.

On October 19, 1962, the reactor had achieved a power of 85 Mw based on the old method of calculation. The reactor was shut down at that time for equipment maintenance. During this shutdown period, more rigid heat balance calculations were made which indicated that the true reactor power was closer to 80 Mw. This was based on intermediate system heat balance data.

There was a close correlation between the heat balance around the shell side of the primary heat exchangers and the air-cooled heat exchangers. However, this did not agree with the primary side heat balance (see Table 122-Ia).

As the intermediate circuit heat balance was consistently correct, within 2% error (see Table 122-Ic, Section 7), while the primary side heat balance was very erratic, it was decided to alter the reactor power calculations to reflect the intermediate circuit heat dissipation. It was considered that the erratic behavior of the primary heat balance data was due to the neglect of the effect of moisture carryover, which was an unknown factor. This enabled the correct calculation of the power being dissipated by the reboiler plant but no moisture correction could be established for the turbine plant. It was decided to use the turbine steam flow rather than feedwater flow in calculating total power as the turbine steam flowmeter does not measure the total moisture carryover. This would produce a lower, or minimum, total power which could be corrected later after measurement or calculation of the moisture carryover (see Table 122-II).

The power is calculated using the heat removed by the primary reboilers and drain coolers in the intermediate system plus the heat removed by the deaerator and subcooler. This eliminates the effect of the moisture carryover in the primary system, as the intermediate system heat balance is utilized and only condensate is entering the deaerator.

The heat balance data around the individual pieces of heat exchange equipment in the reboiler plant are tabulated in Tables 122-Ia through Ic. A running heat balance was not performed as the time for gathering data was limited. This allowed only a sample of the performance of the equipment. From all indications, the primary reboilers and drain coolers performed as specified. It is not known whether the subcooler performed satisfactorily as insufficient instrumentation prohibited gathering information for a heat balance.

Only the air-cooled condenser failed to perform as specified, and this was very serious as it limited the power dissipation capabilities of the reboiler plant. On October 18, 1962, the reboiler plant was unable to maintain intermediate system steam pressure at 350 psig although the air-cooled condenser was in full operation and the control valve wide open. There was insufficient heat dissipation for the steam flow being generated by the primary reboilers. The air-cooled condenser was specified for a capability of 58 Mw at 95°F ambient temperature, but reached its limitation of 47 Mw with a 65°F ambient temperature.

Time limitations on the project prohibited any effort being expended on determining the cause of the decreased rating of the condenser. The pitch angle of the fans was checked twice to be certain that the correct angle had been maintained as recommended by the manufacturer for the required air flow through the unit. No change in angle was ever encountered.

Table 122-1a

EBWR REBOILER PLANT EQUIPMENT HEAT BALANCE

	Date	7/10/62	7/23/62	7/23/62	7/26/62	7/27/62	7/27/62	7/27/62	9/3/62	9/9/62	10/18/62	10/19/62	11/1/62	11/1/62	11/12/62	11/12/62	11/15/62	11/15/62
	Time	---	1530	2000	2000	1500	1530	1545	1200	1200	---	---	1223	1348	1440	1600	1104	1200
1(a) Primary Reboiler - Primary Fluid																		
1. Total Steam Flow, lb/hr x 10 ⁻³	30	63	85	47	92	100	120	125	165	232	222	191	320	244	335	276	317	
2. Pressure, psig	460	475	505	452	450	510	540	470	490	515	515	---	---	---	---	---	---	
3. Temperature (In), °F	450	460	453	458	450	450	470	455	460	470	485	468	463	466	465	466	466	
4. No. 1 Reboiler Temperature (Out), °F	440	450	460	140	125	123	130	450	455	465	480	465	462	462	462	460	463	
5. No. 2 Reboiler Temperature (Out), °F	85	---	---	452	450	462	475	450	458	470	480	471	471	471	471	474	468	
6. Enthalpy (In), Btu/lb	1203	1204.6	1204.6	1204.6	1204.6	1204.6	1204.3	1204.6	1204.6	1204.3	1204.3	1203.3	1204.5	1204.4	1204.5	1204.4	1204.4	
7. Enthalpy (Out), Btu/lb	418.9	430.1	433.3	432.3	430.1	444.1	458.6	430.1	436.8	447.1	464.4	447	444	444	444	442	444.6	
8. Enthalpy Difference Btu/lb	784.1	774.5	771.3	772.3	774.5	760.5	745.7	774.5	767.8	757.2	738.9	757.3	760.5	760.4	760.5	762.4	759.8	
9. Heat Removed, Btu/hr x 10 ⁻⁶	23.53	48.79	65.56	36.30	71.25	76.05	89.48	96.81	126.69	175.67	164.04	144.64	243.36	186.30	254.76	210.42	240.85	
10. Heat Removed, Mw	6.89	14.3	19.21	10.64	20.88	22.28	26.22	28.36	37.12	51.47	48.06	42.36	71.28	54.56	74.61	61.63	70.54	
<div> <div>Only No. 1 Reboiler Operating</div> <div>Only No. 2 Reboiler Operating</div> <div>Both No. 1 and No. 2 Reboilers Operating</div> </div>																		
1(b) Primary Reboiler - Intermediate Fluid																		
1. No. 1 Reboiler Steam Flow, lb/hr x 10 ⁻³	30	62	84	---	---	---	---	61	74.5	102	95	97.5	105	95	98	94	91	
2. No. 2 Reboiler Steam Flow, lb/hr x 10 ⁻³	---	---	---	51	90	100	115	54	78	99	91	90	98	97	102	96	94	
3. Pressure, psig	353-360	358	360	350	350	350	360	350	350	355	350	---	---	---	---	---	---	
4. No. 1 Reboiler Temperature (In), °F	420	425	430	210	200	200	200	430	435	440	440	433	438	438	438	438	440	
5. No. 2 Reboiler Temperature (In), °F	80	---	---	430	430	438	440	430	435	440	440	441	447	444	444	445	445	
6. Temperature (Out), °F	440	435	435	435	415	435	429	430	435	435	440	436	438	438	438	438	438	
7. Enthalpy (In), Btu/lb	396.77	402.3	407.8	407.8	407.8	416.7	418.9	407.8	413.3	413.3	418.9	415.6	421.3	419.5	419.5	419.7	421.5	
8. Enthalpy (Out), Btu/lb	1204.3	1204.1	1204.1	1204.2	1203.8	1204.2	1204.3	1203.8	1204.1	1204.3	1204.3	1204.2	1204.3	1204.3	1204.3	1204.3	1204.3	
9. Enthalpy Difference, Btu/lb	807.5	801.5	796.3	796.4	796	787.5	785.4	796	790.8	791	785.4	788.6	783	784.8	784.8	784.6	782.8	
10. Total Heat, Btu/hr x 10 ⁻⁶	24.22	49.71	66.89	40.62	71.64	78.75	90.3	91.54	120.6	159.8	146.1	147.86	158.95	150.68	156.96	149.07	144.81	
11. Total Heat, Mw	7.09	14.56	19.6	11.90	20.99	23.07	26.46	26.82	35.33	46.8	42.8	43.31	46.55	44.13	45.97	43.66	42.41	
Calculated "U" Value	404	497	815	550	894	729	652	508	641	668	429	576	811	725	770	746	740	

Table 122 lb

EBWR REBOILER PLANT EQUIPMENT HEAT BALANCE

Date	7 10/62	7 23 62	7/23 62	7 26 62	7 27 62	7 27 62	7 27 62	7 27 62	9 3 62	9 9 62	10 18 62	10 19 62	11 1 62	11/1 62	11 1 62	11 1 62	11 12 62	11 12 62	11 12 62	11 12 62	11 15 62	11 15 62	11 15 62	11 15 62
T me		1530	2000	2000	1500	1530	1545	1200	1200				1223	1223	1348	1348	1440	1440	1600	1600	1104	1104	1200	1200
2(a) Primary Drain Cooler Primary Fluid																								
1 Total Steam Flow lb/h x 10 ⁻³	30	63	85	47	92	100	120	125	165	232	222		97.37	93.63	115.4	204.6	117.2	126.8	135.96	199.04	109.54	166.46	107.35	209.65
2 Pressure ps g	460	475	505					470	490	532	515													
3 Temperature (In) F	440	450	460	452	450	462	475	450	455	465	480	465	465	462	462	462	462	462	462	462	460	460	463	463
4 No. 1 Drain Cooler Temperature (Out) F	200	180	190					185	205	205	220	179			240		213		234		213		232	
5 No. 2 Drain Cooler Temperature (Out) F				220	210	190	215	185	205	200	220		185		317			231		294		267		308
6 Enthalpy (In) Btu lb	419	430.1	433.3	432.3	430.1	444.1	458.6	430.1	435.7	447.1	464.4	447.0	447.0	444.0	444.0	444.0	444	444	444	441.8	441.8	444.7	444.7	
7 Enthalpy (Out) Btu lb	168	147.9	158.0	188.1	178.1	158.0	183.1	152.9	173.0	173.0	188.1	147.0	152.9	208.4	287.2	181.1	198.9	202.4	263.5	181.1	235.8	201.0	277.9	
8 Enthalpy Difference Btu lb	251	282.2	275.3	244.2	252.0	286.1	275.5	277.2	262.7	274.1	276.3	300.0	294.1	235.6	156.8	262.9	245.1	241.6	180.5	260.7	206.0	243.7	166.8	
9 Heat Removed Btu/h x 10 ⁻⁶	7.53	17.78	23.4	11.48	23.18	28.61	33.06	34.65	43.34	63.6	61.3	29.21	27.53	27.2	32.08	30.81	31.08	32.84	35.92	28.55	34.29	26.16	34.97	
10 Heat Removed Mw	2.20	5.21	6.86	3.36	6.79	8.38	9.69	10.15	12.7	18.63	17.97	8.55	8.06	7.97	9.4	9.02	9.10	9.62	10.52	8.36	10.04	7.66	10.24	
<div><div>Only No. 1 Reboiler Operating</div><div>Only No. 2 Reboiler Operating</div><div>Both No. 1 and No. 2 Reboilers Operating</div></div>																								
2(b) Primary Drain Cooler Intermediate Fluid																								
1 Total Flow lb/h x 10 ⁻³	30							115	152.5	202	185													
2 No. 1 Reboiler Steam Flow lb/hr x 10 ⁻³	30	62	84					61	74.5	102	95	97.5		105		95		98		94		91		
3 No. 2 Reboiler Steam Flow lb/hr x 10 ⁻³				51	90	100	115	54	78	99	91		90		98		97		102		96		94	
4 Pressure ps g	353	370	370					350	350	355	350													
5 Temperature (In) F	162	150	141	187	170	140	162	145	165	145	145	125	125	122	122	122	122	126	126	117	117	117	117	
6 No. 1 Drain Cooler Temperature (Out) F	420	425	430					430	435	440	440	433		438		438		438		438		440		
7 No. 2 Drain Cooler Temperature (Out) F	80			430	430	438	440	430	435	440	440		441		447		444		444		445		445	
8 Enthalpy (In) Btu lb	129.9	118	109	154.9	137.9	107.9	129.9	112.9	132.9	112.9	112.9	93.0	93.0	90.1	90.1	90.1	90.1	93.7	93.7	85.0	85.0	85.0	85.0	
9 Enthalpy (Out) Btu lb	396.8	402.3	407.8	407.8	407.8	416.7	418.9	407.8	413.3	418.9	418.9	411.2	419.8	417.0	426.8	417.0	423.4	417.0	423.4	417.0	424.5	418.6	424.5	
10 Enthalpy Difference Btu lb	266.9	284.3	298.8	252.9	269.9	308.8	289.0	294.9	280.4	306	306	318.2	326.8	326.9	336.7	326.9	333.3	323.3	329.7	332.0	339.0	333.6	339.5	
11 Total Heat Btu/hr x 10 ⁻⁶	8.01	17.63	25.1	12.9	24.29	30.88	33.24	33.91	42.76	61.8	56.6	31.02	29.41	34.32	32.99	31.05	32.33	31.68	33.62	31.21	32.54	30.36	31.91	
12 Total Heat Mw	2.34	5.16	7.35	3.78	7.12	9.05	9.74	9.94	12.53	18.11	16.59	9.08	8.61	10.05	9.66	9.09	9.47	9.28	9.85	9.14	9.53	8.89	9.34	
Calculated U Value	242.14	544.97	547.56	403.12	713.57	737.52	836.64	498.02	627.99	655.49	575.14	626.54	634.19	476.56	398.25	524.17	542.01	480.70	424.54	526.87	457.85	450.01	368.92	

Table 122-Ic

EBWR REBOILER PLANT EQUIPMENT HEAT BALANCE

Date:	7/10/62	7/23/62	7/23/62	7/26/62	7/27/62	7/27/62	7/27/62	9/3/62	9/9/62	10/18/62	10/19/62	11/1/62	11/1/62	11/12/62	11/12/62	11/15/62	11/15/62
Time:	---	1530	2000	2000	1500	1530	1545	1200	1200	---	---	1223	1348	1440	1600	1104	1200
3. Dearator - Subcooler																	
1. Condensate Flow, lb/hr x 10 ⁻³	36	48	80	47	92	100	120	122	154	230	220	191	320	244	335	276	317
2. Temperature (In), °F	200	180	190	220	210	190	215	185	205	205	220	182	290	222	270	246	282
3. Temperature (Out), °F	114	132	116	118	120	117	118	120	116	120	118	117	127	118	126	121	129
4. Enthalpy (In), Btu/lb	168	148	158	188.1	178.1	158	183.1	152.9	173	173.0	188.4	149.9	259.3	190.2	238.8	214.4	251.1
5. Enthalpy (Out), Btu/lb	81.9	100	84	85.9	87.9	84.9	85.9	87.9	83.9	87.9	85.9	84.7	95.2	85.9	93.8	88.0	96.9
6. Enthalpy Difference, Btu/lb	86.1	48	74	102.2	90.2	73.1	97.2	65.0	89.1	85.1	102.2	65.2	164.1	104.3	145.0	126.4	154.2
7. Power, Btu/hr x 10 ⁻⁶	2.6	3.0	6.3	4.8	8.3	7.3	11.7	8.12	14.7	19.7	22.7	12.44	52.36	25.46	48.58	34.75	49.14
8. Power, Mw	0.76	0.89	1.8	1.4	2.4	2.1	3.4	2.4	4.3	5.8	6.65	3.65	15.34	7.46	14.22	10.19	14.39
4. Flash Tank and Air Cooled Subcooler																	
1. Condensate Flow, lb/hr x 10 ⁻³	30	62	84	51	90	100	115	114	156	202	185	184	200	190	200	185	183
2. Temperature (In), °F	285	310	305	290	218	240	270	240	250	290	275	261	259	246	264	249	240
3. Temperature (Out), °F	162	146	141	190	170	140	162	145	165	145	145	125	122	122	126	117	117
4. Enthalpy (In), Btu/lb	254.2	280	274.7	259.3	186.1	208.3	238.8	208.3	218.5	238.8	228.6	229.6	227.7	214.4	232.7	217.8	208.4
5. Enthalpy (Out), Btu/lb	130	114	109	152.9	137.9	107.9	129.9	112.9	132.9	112.9	112.9	92.8	92.1	90.3	95.6	86.9	85.0
6. Enthalpy Difference, Btu/lb	124.2	166	165.7	106.4	48.2	100.4	108.9	95.4	85.6	125.9	115.7	136.8	135.6	124.1	127.1	130.9	123.4
7. Power, Btu/hr x 10 ⁻⁶	3.73	10.3	13.9	5.43	4.34	10.04	12.5	10.97	13.35	25.43	21.4	25.17	27.12	23.57	25.42	24.21	22.58
8. Power, Mw	1.09	3.02	4.08	1.59	1.27	2.94	3.67	3.2	3.9	7.45	6.3	7.37	7.94	6.91	7.44	7.09	6.61
5. Air-Cooled Condenser																	
1. Flow, lb/hr x 10 ⁻³	30	62	84	51	90	100	115	114	156	202	185	184	200	190	200	185	183
2. Pressure, psig	353-360	354	354	350	350	350	360	350	350	355	350	---	---	---	---	---	---
3. Steam Temperature, °F	440	435	435	440	430	440	440	430	435	435	440	436	438	438	438	438	438
4. Reduced Pressure, psig	180	308	210	242	200	238	340	200	210	335	234	---	---	---	---	234	---
5. Condensate Temperature, °F	387	425-430	390	400	370	400	435	390	395	435	400	399	408	388	406	400	381
6. Enthalpy (In), Btu/lb	1204.3	1204.1	1204.1	1204.3	1203.8	1204.2	1204.3	1203.8	1204.1	1204.1	1204.3	1204.2	1204.3	1204.3	1204.3	1204.3	1204.3
7. Enthalpy (Out), Btu/lb	360.95	405	364	375.0	342.8	375.0	413.3	364.2	369.5	413.3	375.0	374.0	383.6	362.0	381.4	375.2	354.6
8. Enthalpy Difference, Btu/lb	843.35	799.1	840.1	829.3	861.0	829.2	791.0	839.6	834.6	790.8	829.3	830.2	820.7	842.3	822.9	819.1	839.7
9. Heat Removed, Btu/hr x 10 ⁻⁶	25.30	49.54	70.57	42.29	77.49	82.92	90.96	95.7	130.2	159.7	153.4	152.75	164.14	160.04	164.58	151.53	153.66
10. Heat Removed, Mw	7.41	14.52	20.68	12.39	22.70	24.30	26.65	28.04	38.15	46.8	44.95	44.74	48.07	46.87	48.20	44.38	45.01
6. Air-Cooled Drain Cooler																	
1. Flow, lb/hr x 10 ⁻³	30	62	84	51	90	100	115	114	156	202	185	184	200	190	200	185	183
2. Pressure, psig	180	308	210	242	200	238	340	200	210	335	234	---	---	---	---	---	---
3. Temperature (In), °F	387	425-430	390	400	370	400	435	390	395	435	400	399	408	388	406	400	381
4. Temperature (Out), °F	285	308	300	290	218	240	270	240	250	290	275	261	250	246	264	249	240
5. Enthalpy (In), Btu/lb	360.95	405	364	375	342.8	375	413.3	364.2	369.5	413.3	375	374.0	383.6	362.0	381.4	375.2	354.6
6. Enthalpy (Out), Btu/lb	254.17	277.9	269.6	259.3	186.1	208.3	238.8	208.3	218.5	259.3	243.9	229.8	227.5	214.0	232.7	217.9	208.4
7. Enthalpy Difference, Btu/lb	106.78	127.1	94.4	115.7	156.7	166.7	174.5	155.9	151.0	154.0	131.1	144.2	156.1	148.0	148.7	157.3	146.2
8. Heat Removed, Btu/hr x 10 ⁻⁶	3.20	7.88	7.93	5.90	14.10	16.67	20.07	17.77	23.56	31.1	24.2	26.53	31.22	28.12	29.74	29.1	26.75
9. Heat Removed, Mw	0.93	2.31	2.32	1.73	4.13	4.88	5.88	5.21	6.90	9.11	7.11	7.77	9.14	8.23	8.71	8.53	7.83
7. Total Heat Removal Items																	
A. (3 + 4 + 5 + 6), Mw	11.04	20.74	28.92	17.12	30.53	34.26	39.62	38.84	53.27	67.96	64.98	64.69	83.14	70.10	81.02	71.96	76.25
B. (1a + 2a + 3), Mw	9.87	20.7	27.9	15.41	30.10	32.80	39.33	40.89	54.13	75.88	72.68	62.78	104.0	80.14	108.97	90.22	102.83
C. (1b + 2b + 3), Mw	10.21	20.61	28.8	17.09	30.54	34.26	39.62	39.14	52.17	70.69	66.04	64.81	84.25	70.15	79.32	72.52	75.03
D. Turbine Plant, Mw	10.21	20.61	28.8	---	---	---	---	0.05	9.08	---	---	13.25	---	14.7	13.19	17.44	21.85
E. Total Power (C + D), Mw	---	---	---	17.09	30.54	34.26	39.62	39.19	61.25	70.69	79.29	64.81	98.95	83.34	96.76	94.37	100.98
Accuracy, % - A/B	111.92	100.17	103.6	111.09	101.43	104.45	100.74	95.99	98.41	89.56	89.4	103.04	79.11	86.78	72.71	78.22	72.45
- A/C	108.17	100.61	100.45	100.17	99.97	100.0	100.0	99.23	102.11	96.14	98.39	99.81	98.68	100.0	102.10	99.4	101.5
- B/C	96.65	100.4	96.94	90.17	98.56	95.74	99.27	104.47	103.76	107.34	110.05	96.86	124.74	114.11	136.23	123.82	135.89

Table 122-II
MOISTURE CARRYOVER

Date and Time	$W_P \times 10^{-3}$	$Q_P \times 10^{-6}$	$W_I \times 10^{-3}$	$Q_I \times 10^{-6}$	X	1-X	Water Level, (ft-in.)	P_T	P_{TPC}	P_{TC}
11/12/62										
0910	129	99.5	121	97.5	0.980	0.02	---	58.0	17.33	58.6
1000	129	99.5	122	99.1	0.996	0.004	---	58.3	17.65	59.3
1100	126	96.8	120	97.6	---	---	---	---	---	---
1200	138	105.9	134	106.1	---	---	---	---	---	---
1235	173	132.3	150	120.4	0.909	0.091	---	68.5	19.05	71.3
1300	168	128.4	157	125.9	0.980	0.020	---	70.3	19.87	73.9
1410	227	172.0	185	146.6	0.852	0.148	---	78.0	13.62	80.6
1440	244	185.6	190	150.6	0.810	0.190	---	78.6	13.19	81.8
1505	258	196.2	186	149.8	0.762	0.238	---	79.9	12.69	82.6
1535	264	200.5	191	156.0	0.778	0.222	---	84.8	16.17	88.1
1600	335	254.7	200	156.8	0.613	0.387	---	91.0	17.44	95.4
1605	340	260.0	194	152.1	0.581	0.419	---	94.4	20.47	98.2
1607	360	275.2	200	156.8	0.566	0.434	---	96.8	21.24	101.3
1610	310	237.0	200	156.6	0.658	0.342	---	95.9	22.23	101.4
1625	320	242.4	190	154.4	0.636	0.364	---	90.9	19.46	94.7
11/13/62										
1030	119	91.8	115	93.4	---	---	---	55.8	16.50	55.8
1200	197	149.8	177	133.3	0.889	0.111	---	73.0	13.35	75.5
1225	237	180.0	186	147.3	0.817	0.183	---	78.8	15.97	83.5
1240	276	211.0	187	146.6	0.692	0.308	---	86.1	19.65	90.8
1300	282	215.0	183	144.7	0.671	0.329	---	90.5	20.67	91.4
1315	300	228.8	182	145.7	0.634	0.366	16-0	90.2	21.07	91.7
1405	160	122.8	150	123.1	---	---	15-5	71.0	18.80	71.0
1435	157	120.0	148	119.7	0.997	0.003	---	69.8	19.62	70.6
1455	140	107.5	124	99.1	0.921	0.079	14-10	61.6	18.62	61.4
1525	124	95.7	114	94.6	0.989	0.011	14-8	58.8	19.84	59.8
11/14/62										
0845	150	114.4	137	110.1	0.962	0.038	---	67.7	21.74	68.7
0910	180	137.3	168	133.8	0.975	0.025	15-10	68.7	12.09	69.6
0950	160	122.2	152	123.2	---	---	15-5	73.2	20.2	73.2
1030	161	123.1	150	119.4	0.970	0.030	15-5	71.9	21.34	73.0

Date and Time	$W_P \times 10^{-3}$	$Q_P \times 10^{-6}$	$W_I \times 10^{-3}$	$Q_I \times 10^{-6}$	X	1-X	Water Level, (ft-in.)	P_T	P_{TPC}	P_{TC}
11/15/62										
0840	125	96.5	114	95.0	0.984	0.016	15-10	59.1	21.08	60.2
0930	188	143.5	170	136.5	0.951	0.049	15-3	79.8	24.33	84.0
0947	140	108.0	150	125.5	---	---	14-8	72.5	20.1	72.5
1000	178	135.9	160	131.7	0.969	0.031	15-3	75.3	21.19	76.4
1015	207	158.2	160	138.4	0.874	0.126	15-5	77.1	21.13	78.2
1030	230	176.1	175	133.3	0.754	0.246	15-7	83.2	21.10	83.5
1050	250	191.4	182	145.1	0.755	0.245	15-7	87.8	25.01	91.8
1104	276	210.7	185	149.0	0.704	0.296	15-7	91.1	21.85	92.2
1120	285	217.6	178	141.8	0.649	0.351	15-8	96.0	25.07	94.5
1137	298	227.2	184	145.4	0.637	0.363	15-8	96.3	23.29	93.1
1200	317	240.8	183	144.7	0.599	0.401	15-8	99.1	25.95	98.5
1210	316	239.7	190	148.6	0.618	0.382	15-8	101.3	26.37	101.2
1212	317	241.7	190	148.5	0.610	0.390	15-8	101.4	26.19	101.2
1213	316	239.7	185	146.9	0.611	0.389	15-8	99.7	26.20	99.3
11/16/62										
1400	124	95.3	115	92.2	---	0.033	16-0	60.0	22.28	61.7
1450	101	77.6	95	76.5	---	0.014	15-3	53.2	21.77	54.4
1540	111	85.3	103	82.9	---	0.028	15-3	56.0	21.87	57.3
1520	124	95.3	116	93.0	---	0.024	15-3	60.7	21.57	61.7
11/19/62										
1120	191	144.7	183	144.8	---	0	15-3	75.6	19.33	82.4
1142	197	149.2	188	148.7	---	0.003	15-3	78.1	13.62	79.4
11/20/62										
1030	218	165.6	170	133.8	---	0.192	16-0	74.7	15.89	76.8
1115	226	171.7	185	145.6	---	0.152	16-0	80.3	16.21	82.7
1300	193	146.8	160	126.1	---	0.141	16-1	70.1	15.89	72.2
1330	195	147.3	185	145.8	---	0.010	15-3	77.3	15.78	79.8
1407	167	126.5	155	122.3	---	0.033	15-3	66.8	13.95	67.5
1450	200	150.9	184	144.8	---	0.040	15-3	76.6	15.31	78.6
1520	170	128.8	155	122.1	---	0.052	15-3	66.8	13.78	67.3
1550	82	63.4	80	64.4	---	---	15-3	40.9	13.3	40.9
1610	106	81.5	100	80.5	---	0.012	16-0	47.0	14.35	48.1

Legend:

W_P = Primary reboiler feedwater flow, lb/hr

Q_P = Heat removed in primary reboilers (theoretical), Btu/hr

W_I = Intermediate feedwater flow, lb/hr

Q_I = Heat removed by intermediate fluid, Btu/hr

X = Primary steam quality

1 - X = Moisture carryover

P_T = Total reactor power, Mw

P_{TPC} = Turbine plant power (corrected for moisture carryover), Mw

P_{TC} = Total reactor power (with corrected turbine plant power), Mw

One interesting phenomenon that was noticed during operation was the wide variations of the outlet temperatures for the two primary drain coolers. At first this was neglected but after calculating the amount of carryover in the primary system it appeared that the carryover might be the cause of this temperature variation. Based on certain assumptions (see Attachment A) the flow division was calculated and is tabulated with the heat balance data for the drain coolers in Table 122-Ib. During high power runs it is estimated that the flow through the No. 2 reboiler and drain-cooler circuit was twice that of the No. 1 circuit. This is possible as there is a natural separation that can occur because of the configuration of the primary piping. The No. 1 reboiler steam line comes off a lateral on the main header which makes the steam turn 45 degrees to enter the No. 1 reboiler, while the steam to the No. 2 reboiler continues straight ahead. Most of the moisture would continue straight ahead into the No. 2 reboiler. This would also account for lower intermediate steam output on the No. 2 reboiler.

MOISTURE CARRYOVER

It was known from the beginning of operation that there would be some moisture carryover with the reactor steam. However, the amount of carryover and the effect of reactor water level were not foreseen.

Fig. 122-1

Reactor Water Level Versus Reactor Power. Effect of Level on Carryover

leaves in the saturated condition and, therefore, does not enter into the heat balance. However, a correction is made for some slight subcooling of the condensate should this condition exist. Carryover is plotted against reactor power for various reactor water levels in Fig. 122-1.

At high power levels it was known that there was considerable carryover and it was considered that the only positive method of determining carryover was by a heat balance around the primary reboilers. The method of calculation is shown in Attachment A.

This method assumes that all the primary fluid entering the reboiler is steam and leaving is saturated water. The intermediate fluid heat gain is a means of measuring the actual heat removed from the primary fluid. This heat can only be removed from the primary steam that is present as the moisture carried along with the steam both enters and

Fig. 122-2

Effect of Reactor Water Level and Boric Acid on Control Rod Position for Various Power Levels

Fig. 122-3

Primary Flow Versus Reactor Power

This plot indicates that the reactor water level has a material effect upon the amount of moisture carryover.

The effect of water level and boric acid concentration upon control rod position (banked) and reactor power is shown in Fig. 122-2. The effect of water level on primary feedwater flow and reactor power is shown in Fig. 122-3. All of these effects are due to hydraulic phenomena which are more completely explained in Reference 1.

CONCLUSIONS

There is sufficient data available in this report to make a better analysis of the performance of various pieces of heat exchange equipment, with the exception of the air-cooled equipment. However, time allows only a sampling of the data from which an inference of the capabilities can be drawn. All of the equipment performed satisfactorily except for the air-cooled condenser.

It appears that the air-cooled condenser is undersized. Firm proof of this can only be obtained through more thermodynamic information. The information needed is on the air side of

the exchanger. This would require velocity thermocouples to determine air temperatures, pitot tubes for a traverse of the air distribution and flow rate, and certain atmospheric data. From this data a recommendation may be formulated for the changes necessary to bring the exchanger up to design rating.

From the data, it is estimated that the limit of the reactor power is set by the moisture carryover. It was required that the turbine-generator be shut down during high power operation because of the high moisture content of the steam causing excessive turbine vibration. If a limit of 5% moisture carryover is established to enable operation of the turbine-generator, then the reactor power is limited to 65 Mw at a reactor water level of 16 ft and 80 Mw at a water level of 15 ft-3 in. It may be possible to install additional moisture separators and allow higher power capabilities. Operation with 10% moisture and above, especially above 40%, on a continuous long-term basis is not recommended for the damaging effect it can have on pipe elbows and other places of change in flow direction.

The power log and other pertinent data are included here as Attachment B.

REFERENCE

- (1) EBWR Test Report No. 105B.

Attachment A

CALCULATION OF REACTOR POWER

$$P = W_I(h_1 - h_2) + W_P(h_3 - h_4) + W_T(h_5 - h_6)$$

where:

P = reactor power, Btu/hr

W_I = intermediate system feedwater flow, lb/hr

W_P = reboiler plant primary feedwater flow, lb/hr

W_T = turbine plant primary steam flow, lb/hr

h_1 = enthalpy of the intermediate steam, Btu/lb

h_2 = enthalpy of the intermediate feedwater, Btu/lb

h_3 = enthalpy of primary condensate to the deaerator, Btu/lb

h_4 = enthalpy of primary feedwater from subcooler, Btu/lb

h_5 = enthalpy of reactor steam, Btu/lb

h_6 = enthalpy of primary feedwater to reactor, Btu/lb

CALCULATION OF CARRYOVER

Assume some degree of subcooling

$$Q_{P_T} = W_P \Delta h = W_P (h_1 - h_3)$$

$$Q_{P_A} = XW_P (h_1 - h_2) + W_P (h_2 - h_3)$$

$$Q_{P_A} = Q_I = W_I \Delta h = W_I (h_4 - h_5)$$

$$\frac{Q_{P_T}}{Q_I} = \frac{W_P (h_1 - h_3)}{XW_P (h_1 - h_2) + W_P (h_2 - h_3)}$$

$$X(h_1 - h_2) + (h_2 - h_3) = \frac{Q_I}{Q_P} (h_1 - h_3)$$

$$X = \frac{Q_I (h_1 - h_3)}{Q_P (h_1 - h_2)} - \frac{(h_2 - h_3)}{(h_1 - h_2)}$$

Assume no subcooling

$$Q_{P_T} = W_P (h_1 - h_2)$$

$$Q_{P_A} = XW_P (h_1 - h_2)$$

$$\frac{Q_{P_T}}{Q_I} = \frac{W_P (h_1 - h_2)}{XW_P (h_1 - h_2)}$$

$$X = \frac{Q_I}{Q_{P_T}} = \frac{W_I (h_4 - h_5)}{W_P (h_1 - h_2)}$$

where:

Q_{P_T} = theoretical heat removal from primary system, Btu/hr

Q_{P_A} = actual heat removal from primary system, Btu/hr

W_P = reboiler plant primary feedwater flow, lb/hr

X = steam quality

W_I = intermediate system feedwater flow, lb/hr

h_1 = enthalpy of primary steam, Btu/lb

h_2 = enthalpy of saturated primary water, Btu/lb

h_3 = enthalpy of subcooled primary water, Btu/lb

h_4 = enthalpy of intermediate steam, Btu/lb

h_5 = enthalpy of intermediate feedwater to primary reboiler, Btu/lb

CALCULATION OF FLOW DISTRIBUTION TO PRIMARY DRAIN COOLERS

$$Q_1 = UA \Delta T_{L1} = W_1 C_P \Delta t_1$$

$$Q_2 = UA \Delta T_{L2} = W_2 C_P \Delta t_2$$

$$\frac{UA \Delta T_{L1}}{UA \Delta T_{L2}} = \frac{W_1 C_P \Delta t_1}{W_2 C_P \Delta t_2}$$

Assume that the over-all heat transfer coefficient, U , and the specific heat, C_P , are equal and can be cancelled from the equation. The areas, A , of both drain coolers are equal.

$$\therefore \frac{\Delta T_{L1}}{\Delta T_{L2}} = \frac{W_1 \Delta t_1}{W_2 \Delta t_2}$$

$$W_1 = W_2 \frac{\Delta T_{L1} \Delta t_2}{\Delta T_{L2} \Delta t_1} = K W_2$$

$$W_T = W_1 + W_2 = W_2 (1 + K)$$

$$W_2 = WT/(1 + K)$$

$$W_1 = KWT/(HK)$$

where:

Q = heat transferred in drain cooler, Btu/hr

W = primary condensate flow to drain cooler, lb/hr

ΔT_L = logarithmic mean temperature difference, °F

Δt = temperature difference across drain cooler, °F

subscript 1 refers to drain cooler No. 1

subscript 2 refers to drain cooler No. 2

Attachment B

PLANT TEMPERATURES AND REACTOR POWER LOGS

REBOILER PLANT SYSTEM TEMPERATURES, °F

Date and Time	Thermocouple No.*																Date and Time	Thermocouple No.*																	
	1	2	3	4	5	6	7	13	14	15	16	17	18	19	22	23		1	2	3	4	5	6	7	13	14	15	16	17	18	19	22	23		
11/1/62																	11/14/62																		
1223	468	465	179	125	125	433	436	471	185	441	399	261	255	126	435	444	0845	465	462	201	129	150	433	436	471	211	446	375	246	240	215	435	444		
1348	463	462	240	132	122	438	438	471	317	447	408	259	249	126	437	444	0910	465	462	195	131	150	432	436	470	234	446	369	186	180	153	434	443		
11/12/62	0910	459	453	180	126	135	426	435	460	183	430	387	210	200	135	431	441	0950	463	460	150	122	102	425	436	463	152	428	366	201	195	99	434	443	
	1000	459	453	177	125	129	426	435	460	180	429	378	215	198	132	430	441	1030	462	459	179	122	111	428	437	464	168	432	378	228	206	111	434	443	
	1100	459	455	176	125	129	423	436	460	180	429	372	210	201	132	430	441	11/15/62	0840	456	453	180	123	138	425	435	459	189	432	372	220	205	138	432	440
	1200	462	456	183	129	138	426	436	462	192	432	392	231	219	141	431	442		0930	465	460	177	123	117	430	435	470	183	435	370	234	222	114	432	440
	1235	465	459	208	134	156	436	436	466	213	441	374	249	237	159	431	444		0947	460	453	147	121	115	419	433	465	171	426	358	209	209	110	433	441
	1300	465	459	208	135	158	436	436	463	221	442	380	255	249	159	431	442		1000	464	460	153	120	101	423	435	465	183	433	359	214	200	100	434	441
	1410	466	465	213	132	132	438	438	470	204	444	384	250	241	131	433	444		1015	465	459	177	123	111	434	435	471	207	441	375	236	222	111	434	443
	1440	466	462	203	132	122	438	438	471	231	444	388	246	240	123	435	444		1030	465	458	207	126	120	438	438	471	237	444	384	245	234	120	432	444
	1505	466	462	204	132	120	438	436	470	228	444	398	249	240	123	433	444		1050	466	458	210	129	119	438	438	470	234	444	390	240	228	114	434	444
	1535	466	463	216	134	119	438	438	470	234	444	388	246	235	120	435	444		1104	466	460	213	132	117	438	438	474	267	443	400	249	236	120	436	444
	1600	468	462	234	138	126	438	438	471	294	444	406	264	252	129	435	444		1120	465	460	218	132	114	438	438	471	290	445	382	234	220	113	435	444
1605	465	459	220	140	129	438	436	467	314	444	405	264	255	132	432	444	1137	466	461	215	134	111	441	438	471	285	445	384	234	222	111	435	447		
1607	465	459	226	140	129	438	435	465	345	444	399	259	252	132	432	442	1200	466	463	232	138	117	440	438	468	308	445	381	240	231	117	435	444		
1610	465	459	228	141	123	440	436	470	344	444	392	243	240	123	433	444	1210	468	464	227	137	120	441	438	470	280	445	390	243	230	120	435	445		
1625	468	465	234	138	120	438	438	471	291	444	393	250	240	123	435	444	1212	468	461	227	138	120	441	438	471	303	445	388	246	235	120	435	445		
																	1213	468	464	224	137	120	441	438	471	310	445	387	243	234	120	435	445		
11/13/62																																			
1030	457	453	189	129	147	426	435	466	195	432	387	248	234	150	430	441																			
1200	466	462	216	131	153	438	438	461	234	444	393	255	246	156	434	444																			
1225	468	463	211	132	123	438	438	472	246	445	408	258	246	125	435	444																			
1240	465	459	219	135	123	438	436	465	264	444	406	255	246	123	434	444																			
1300	465	462	219	135	120	440	438	470	270	445	405	255	242	120	435	444																			
1315	466	462	222	138	123	438	438	470	273	445	410	259	246	125	435	444																			
1405	463	456	150	123	104	426	436	465	171	429	368	212	200	102	435	444																			
1435	463	459	150	123	101	427	438	463	158	430	363	203	192	99	434	443																			
1455	462	456	141	123	95	425	438	463	147	430	352	183	168	93	434	443																			
1525	459	453	138	123	99	423	438	460	150	427	330	180	168	98	434	444																			

Legend: *

1. Steam to primary reboilers.
2. Condensate from primary reboiler No. 1.
3. Condensate from primary drain cooler No. 1.
4. Condensate from deaerator.
5. Feedwater to primary drain cooler No. 1.
6. Feedwater to primary reboiler No. 1.
7. Intermediate steam.
8. Condensate from secondary reboiler.
9. Condensate from secondary drain cooler.
10. Feedwater to secondary drain cooler.
11. Feedwater to secondary reboiler.
12. Heating steam.
13. Condensate from primary reboiler No. 2.
14. Condensate from primary drain cooler No. 2.
15. Feedwater to primary reboiler No. 2.
16. Condensate from air-cooled condenser.
17. Condensate from air-cooled drain cooler.
18. Condensate from flash tank.
19. Condensate from air-cooled subcooler.
20. Flash tank vent steam.
21. Condensate from air-cooled flash condenser.
22. Primary reboiler No. 1 intermediate steam.
23. Primary reboiler No. 2 intermediate steam.
24. Secondary reboiler.

* Note: Readings on thermocouples Nos. 8 to 12, 20, 21, and 24 were not taken as the systems were not in operation.

EBWR POWER LOG

Date and Time	Reboiler Plant Primary Flow, lb/hr x 10 ³		Reboiler Intermediate Flow, lb/hr x 10 ³			Reboiler Plant Temperatures, °F			Turbine Plant Flow, lb/hr x 10 ³		Linear Flux		Control Rod Position, in.		Reactor Power, Mw				Power Meter, Mw	Noise Meter, %	H ₃ BO ₃ gm/gal	Water Level, ft.-in.
	Feed-Water	Steam	No. 1 Reboiler Steam	No. 2 Reboiler Steam	Feed-Water	T ₁₉	T ₃ or T ₁₄	Sub-Cooler Outlet	Steam	Feed-Water	Band	Mean	Nos. 1-8	No. 9	Reboiler Int'r. Side	Dearator Sub-Cooler	Turbine	Total				
10/31/62																						
1317	146	—	71	66	136	135	195	118	50	60			35	30	44.0	3.3	20.0	67.3			1.8	
1345	165	—	76	71	146	155	225	118	50	64			36	30	46.5	4.2	21.3	72.0			1.8	
1413	175	—	78	75	152	148	205	120	50	66			37	30	48.4	4.4	22.0	74.8			1.8	
1502	184	—	80	74	154	150	225	119	50	66			37	30	49.1	5.8	22.0	76.9			1.8	
1527	194	—	80	76	156	150	220	120	50	66			36	32	49.6	6.0	22.0	77.6			1.8	
1540	193	—	80	77	156	150	220	119	50	66			35.5	32	49.6	5.7	22.0	77.3			1.8	
1600	193	—	78	76	153	150	225	117	50	66			35.2	32	48.8	6.1	22.0	76.9			1.8	
1620	193	—	77	76	153	150	210	117	50	66			34.6	34.6	48.8	5.0	22.0	75.8			1.8	
11/V/62																						
0837	165	—	81	76	154	145	190	118	6.5	11			32.5	32.5	49.4	3.5	3.3	56.2			1.23	
0937	195	—	84	78	163	115	195	117	30	43			36.5	30.0	53.5	4.5	14.3	70.3			1.23	
0955	195	—	86	81	166	130	200	118	31	44			38	30	53.9	4.7	14.7	73.3			1.23	
1020	215	—	88	84	172	130	215	119	30	45			35	35	55.7	6.1	15.0	76.8			1.23	
1040	200	—	90	85	170	125	190	119	30	44			34.4	40	55.5	4.2	14.7	74.4			1.23	
1140	159	170	82	77	160	125	175	115	30	40			33.0	40	52.2	2.9	13.3	68.4			1.07	
1150	158	165	80	74	156	121	172	118	30	38			33	40	51.0	2.5	12.7	66.2			1.07	
1210	203	198	97	89	188	122	179	121	—	—			33	40	61.5	3.4	—	64.9			1.07	
1223	191	203	97.5	90	184	122	180	117	—	—			33	40	61.5	3.6	—	65.1			1.07	
1245	220	—	—	—	195	130	190	119	—	—			34.4	40	63.8	3.9	—	67.7			1.07	
1338	292	265	105	100	200	128	225	123	31	54			38	40	65.6	8.7	10.3	84.9			1.07	
1348	320	280	105	98	200	125	280	127	44	79			38	40	65.3	14.4	14.7	94.4			—	
1404	240	240	80	80	160	108	310	112	45.5	63			36.7	36.7	52.9	13.9	15.2	82.0			—	
1430	245	230	90	90	180	105	200	119	45.5	63.5			36.7	36.7	59.6	5.8	15.2	80.6			—	
1510	297	250	93	97	196	110	260	122	45	71			37.7	37.7	63.7	12.0	15.0	90.7			—	

EBWR POWER LOG (Cont'd.)

Date and Time	Reboiler Plant Primary Flow, lb/hr x 10 ³		Reboiler Intermediate Flow, lb/hr x 10 ³			Reboiler Plant Temperatures, °F			Turbine Plant Flow, lb/hr x 10 ³		Linear Flux		Control Rod Position, in.		Reactor Power, Mw				Power Meter, Mw	Noise Meter, %	H ₃ BO ₃ gm/gal	Water Level, ft.-in.
	Feed-Water	Steam	No. 1 Reboiler Steam	No. 2 Reboiler Steam	Feed-Water	T ₁₉	T ₃ or T ₁₄	Sub-Cooler Outlet	Feed-Water	Steam	Band	Mean	Nos. 1-8	No. 9	Reboiler Int'r. Side	Dearator Sub-Cooler	Turbine	Total				
11/12/62																						
0910	129	130	62	60	121	135	180	118	50	54	10×10^{-5}		48	48	39	2.3	16.7	58	—	—	3.41	—
1000	129	128	62	62	122	135	180	118	50	54	7.0	-7.6	48	48	39.3	2.3	16.7	58.3	—	—	3.59	—
1100	126	129	62	60	120	135	190	117	49.5	54	6.9	-7.5	48	48	38.8	2.3	16.5	57.6	58	2.1	3.59	—
1200	138	140	70	63	134	140	185	119	49	54	7.4	-8.0	48	48	43	2.7	16.3	62	62	—	2.54	—
1235	173	165	78	75	150	160	215	118	49	62	3×10^{-4}		48	48	47.3	4.9	16.3	68.5	69.5	3.1*	—	—
1300	168	182	81	79	157	160	210	119	49	61.5	2.65	-2.9	48	48	49.5	4.5	16.3	70.3	71	—	—	—
1410	227	230	93	94	185	135	225	117	33	47	2.7	-3.0	48	48	59.8	7.2	11.0	78.0	78	3.9	2.04	—
1440	244	245	95	97	190	125	210	118	30	46	2.9	-3.4	48	48	62	6.6	10.0	78.6	79	4.2	—	—
1505	258	240	96	95	186	120	235	120	30	46	2.9	-3.5	48	48	61	8.9	10.0	79.9	81	5.2	1.70*	—
1535	264	245	100	99	191	120	230	120	40.5	60	3.1	-3.6	48	48	62.6	8.7	13.5	84.8	83	3.8	1.65*	—
1600	335	280	98	102	200	125	260	126	39	72	3.3	-3.8	48	48	64.8	13.2	13.0	91.0	90	4.5	—	—
1605	340	—	—	—	194	130	275	126	50	87	—	—	—	—	—	—	—	—	—	—	—	—
1607	360	—	—	—	200	130	275	130	50	92	—	—	—	—	64.8	15.3	16.7	96.8	—	—	—	—
1610	310	—	—	—	200	130	280	123	50	88	—	—	—	—	—	—	—	—	—	—	—	—
1625	320	270	97	100	190	120	255	118	47	78	—	—	—	—	62.2	13.0	15.7	90.9	90	5.6	—	—
11/13/62																						
1030	119	118	60	57	115	150	195	118	49.5	54	10×10^{-5}		35.5	40	36.6	2.7	16.5	55.8	51.5	1.8	2.2	—
1200	197	198	89	91	177	160	215	119	32.5	44	0.63	-0.68	48	48	56.7	5.5	10.8	73.0	74.5	3.6	2.2	—
1225	237	233	94	94	186	125	220	120	34.0	55.5	3×10^{-4}		48	48	60.5	7.0	11.3	78.8	81.5	4.9	—	—
1240	276	250	92	95	187	125	250	122	45.0	75.5	0.31	-0.36	48	48	60.8	10.3	15	86.1	84.5	4.0	—	—
1300	282	250	92	93	183	120	250	119	59.5	80.5	0.32	-0.35	48	48	59.9	16.8	19.8	90.5	85.5	3.9	—	—
1315	300	260	93	93	182	125	250	122	59.0	85	0.32	-0.37	48	48	59.3	11.3	19.6	90.2	86.5	4.9	1.6	16-0
1405	160	165	78	76	150	100	160	110	56.5	65	0.28	-0.30	48	48	49.9	2.3	18.8	71.0	75.5	2.4	2.16	15-5
1435	157	160	78	72	148	100	155	118	56.5	60.0	0.275	-0.29	48	48	49.3	1.7	18.8	69.8	73.5	2.0	—	—
1455	140	140	64	60	124	90	145	118	56.5	61.5	0.265	-0.275	48	48	41.7	1.1	18.8	61.6	64.0	1.0	—	14-10
1525	124	125	61	57	114	100	145	118	56.5	61.0	0.26	-0.27	48	48	37.9	1.1	18.8	58.8	61.0	0.8	—	14-8
11/14/62																						
0845	150	150	72	68	137	150	195	117	62	68	10×10^{-5}		37	40	43.6	3.4	20.7	67.7	63.5	3.2	0.86	15-10
0910	180	185	87	83	168	155	200	118	33.5	37.5	0.8	-0.86	37	40	53.2	4.3	11.2	68.7	63	2.9	—	—
0950	160	175	78	76	152	100	165	114	60.5	70	3×10^{-4}		48	48	50.6	2.4	20.2	73.2	73	2.3	—	15-5
1030	161	165	78	72	150	110	165	119	60.5	66.5	0.3	-0.32	48	48	49.5	2.2	20.2	71.9	72	2.0	—	15-5

*These readings are considered questionable.

EBWR POWER LOG (Cont'd.)

Date and Time	Reboiler Plant Primary Flow, lb/hr x 10 ³		Reboiler Intermediate Flow, lb/hr x 10 ³			Reboiler Plant Temperatures, °F			Turbine Plant Flow, lb/hr x 10 ³		Linear Flux		Control Rod Position, in.		Reactor Power, M _w				Power Meter, M _w	Noise Meter, %	H ₃ BO ₃ gm/gal	Water Level, ft.-in.	
	Feed-Water	Steam	No. 1 Reboiler Steam	No. 2 Reboiler Steam	Feed-Water	T ₁₉	T ₃ or T ₁₄	Sub-Cooler Outlet	Steam	Feed-Water	Band	Mean	Nos. 1-8	No. 9	Reboiler Int'l. Side	Dearator Sub-Cooler	Turbine	Total					
11/15/62											10×10^{-5}												
0840	125	125	61	58	114	140	185	118	60	65	0.67 -0.72	0.695	31	40	36.6	2.5	20	59.1	50	1.6	0.105	15-10	
											3×10^{-4}												
0930	188	190	87	85	170	110	180	115	60.5	76.5	0.32 -0.36	0.34	48	48	56.1	3.6	20.1	79.8	79	2.3	-	15-3	
0947*	140	160	81	75	150	100	175	117	60.5	72.5	0.33 -0.34	0.335	48	48	50.0	2.4	20.1	72.5	76.5	1.8	-	14-6	
1000	178	170	85	80	160	100	155	118	60.5	66.0	0.335-0.345	0.34	48	48	53.3	1.9	20.1	75.3	77	2.0	-	15-3	
1015	207	190	84	79	160	110	192	121	60	70.5	0.33 -0.355	0.345	48	48	52.8	4.3	20	77.1	80	2.8	-	15-5	
1030	230	215	85	85	175	120	222	121	62.5	77	0.34 -0.36	0.35	48	48	55.6	6.8	20.8	83.2	82.5	3.2	-	15-7	
1050	250	240	93	92	182	115	215	119	63	91	0.35 -0.37	0.36	48	48	59.8	7.0	21.0	87.8	84	3.0	-	15-7	
1104	276	250	94	96	185	115	240	121	62.5	83	0.35 -0.40	0.37	48	48	60.7	9.6	20.8	91.1	86.5	4.2	-	15-7	
1120	285	240	90	91	178	110	250	123	80	100	0.36 -0.40	0.38	48	48	58.8	10.6	26.6	96.0	88	3.5	-	15-8	
1137	298	250	92	94	184	115	230	124	79.5	94	0.36 -0.40	0.38	48	48	60.5	9.3	26.5	96.3	89	4.6	-	15-8	
1200	317	250	91	94	183	115	260	129	80	109	0.36 -0.43	0.39	48	48	60.2	12.3	26.6	99.1	91	4.7	-	15-8	
1210	316	250	94	96	190	120	265	130	79.5	109	0.36 -0.46	0.40	48	48	62.2	12.6	26.5	101.3	90	3.9	-	15-8	
1213	316	250	93	95	185	120	268	130	80	108	0.33 -0.45	0.42	48	48	60.5	12.6	26.6	99.7	91.5	10+	-	15-8	
1200	317	260	92	94	185	110	260	130	80	109	-	-	48	48	61.0	12.3	26.6	99.9	-	-	-	-	
1212	317	270	94	96	190	120	263	130	80	109	-	-	48	48	62.2	12.6	26.6	101.4	-	-	-	-	
11/16/62											10×10^{-5}												
1400	124	125	61	59	115	125	170	118	62	69.5	0.65 -0.74	0.70	30	40	37.5	1.9	20.6	60.0	-	-	-	16-0	
1450	101	100	50	50	95	100	150	117	62	67	0.62 -0.64	0.63	-	-	31.6	1.0	20.6	53.2	-	-	-	15-3	
1540	111	110	55	54	103	110	160	118	62	68	0.68 -0.71	0.69	1-5 @ 32 6-8 @ 31.5	40	34.0	1.4	20.6	56.0	-	-	-	15-3	
1620	124	125	62	60	116	120	180	120	62	67	0.75 -0.79	0.77	34	40	37.9	2.2	20.6	60.7	-	-	-	15-3	
11/19/62											3×10^{-4}												
1120†	191	220	93	93	183	150	205	118	37.5	59	0.33 -0.36	0.345	48	48	58.2	4.9	12.5	75.6	-	-	-	15-3	
1142	197	210	95	93	188	120	190	116	37	41.5	0.33 -0.36	0.345	48	48	61.5	4.3	12.3	78.1	-	-	-	15-3	
11/20/62																							
1030	218	215	88	89	170	105	190	118	41.5	55.5	-	-	34	40	56.3	4.6	13.8	74.7	-	-	-	16-0	
1115	226	-	93	93	185	114	190	119	41.5	55	-	-	35	40	61.8	4.7	13.8	80.3	-	-	-	16-0	
1300	193	-	81	82	160	105	177	119	41.5	53.5	3×10^{-4}		32	40	53.0	3.3	13.8	70.1	-	-	-	16-1	
1330	195	-	96	93	185	119	179	118	40	48.5	0.25 -0.28	0.275	48	48	60.5	3.5	13.3	77.3	-	-	-	15-3	
1407	167	-	80	77	155	95	150	117	40	43.5	0.33 -0.35	0.34	48	30	51.9	1.6	13.3	66.8	-	-	-	15-3	
1450	200	-	95	93	184	114	174	118	40	48	0.285-0.32	0.30	48	48	60.0	3.3	13.3	76.6	-	-	-	15-3	
											10×10^{-5}												
1520	170	-	82	77	155	98	155	117	40	43.5	-	0.90	38	40	51.6	1.9	13.3	66.8	-	-	-	15-3	
1550	82	-	42	40	80	84	135	110	40	43.5	-	0.45	26.5	40	27.0	0.6	13.3	40.9	-	-	-	15-3	
1610	106	-	51	50	100	98	150	114	40	44	-	0.515	26.5	40	32.6	1.1	13.3	47.0	-	-	-	16-0	

*Flow was changing

†System not stabilized

EBWR POWER LOG (Cont'd)

Date and Time	Reboiler Plant Primary Flow lb/hr x 10 ³		Reboiler Intermediate Flow lb/hr x 10 ³			Reboiler Plant Temperatures °F			Turbine Plant Flow lb/hr x 10 ³		Linear Flux		Control Rod Position, in		Reactor Power, Mw				Power Meter Mw	Noise Meter, %	H ₃ BO ₃ gm/gal	Water Level, ft in	
	Feed Water	Steam	No 1 Reboiler Steam	No 2 Reboiler Steam	Feed Water	T ₁₉	T ₃ or T ₁₄	Sub Cooler Outlet	Steam	Feed Water	Band	Mean	Nos 1 8	No 9	Reboiler Int'r Side	Deaerator Sub Cooler	Turbine	Total					
11/26/62																							
1045	46	7	26	22	34	95	135	96	31	35	10 x 10 ⁻³ 0 27 0 29 0 28		24	24	11 4	0 5	10 9	22 8	-	-	-	15 6	
1125	45	42 5	35	30	55	113	115	101	31 5	36	0 34 0 36	0 35	25 5	25 5	18 1	0 17	10 5	28 8	-	-	-	15 6	
1215	60	60	34	30	53	108	150	103	31 5	36	0 35 0 36	0 355	26 5	26 5	17 5	0 8	10 5	28 8	-	-	-	15 3	
1230	86	-	44	38	76	132	165	113	31 5	36	-	0 445	28 5	28 5	24 6	1 3	10 5	36 4	-	-	-	15 3	
1250	82	-	43	38	70	135	174	119	31 5	35 5	-	0 43	28 5	28 5	22 5	1 3	10 5	34 3	-	-	-	15 0	
1410	100	-	55	48	96	140	180	117	31 5	37	-	0 58	32 5	32 5	30 8	1 8	10 5	43 1	-	-	-	14 7	
1515	98	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	
1620	130	-	61	58	120	140	190	118	32	35	-	0 63	32 5	32 5	-	-	-	-	-	-	-	15 3	
11/27/62																							
0845	128	126	63	60	120	135	190	118	63 5	67 5	0 68 0 77	-	32 5	32 5	38 7	2 6	21 4	62 7	-	-	-	15 10	
0930	95	100	52	47	94	110	160	116	64	67	-	0 65	32 5	32 5	-	-	-	53 5	-	-	-	15 0	
1054	148	150	74	72	145	145	200	118	63	67 5	-	0 32	47 0	47 0	-	-	-	70 9	-	-	-	14 10	
1254	84	80	42	40	72	100	150	114	32	37	-	0 40	25 5	25 5	23 9	0 9	11 3	36 1	-	-	-	15 11	
1315	82	80	39	37 5	71	105	155	110	32	35 5	-	0 40	25 5	25 5	-	-	-	35 8	-	-	-	15 11	
1345	56	55	35 5	34	59	95	140	100	14 5	17 5	-	0 29	23 0	23	-	-	-	25 6	2	8 5	-	16 5	
11/28/62																							
0930	215	210	83	82	165	165	240	118	32	58	-	0 82	33	33	-	-	-	73 7	62	34	-	16 2	
1010	260	225	90	90	174	165	260	118	32	58	-	0 275	33	33	-	-	-	79 5	62	36	-	16 3	
1035	265	245	90	89	176	163	255	118	32	53 5	-	0 275	33	33	-	-	-	76	-	-	-	-	
1130	215	200	96	87	173	153	205	120	31	59	-	0 28	36 5	36 5	-	-	-	73 5	-	-	-	15 7	
1220	226	215	98	92	172	175	230	118	31	41	-	0 275	36 5	36 5	-	-	-	73 86	64	40	-	15 7	
1250	220	220	94	90	188	180	230	118	31	44	-	0 29	37	37	-	-	-	78 8	66	47	-	15 7	
1437	64	48	32	29	50	130	175	109	0	0	10 x 10 ⁻⁵ - 0 21		22	22	-	-	0	16 8	0	0	-	16 0	
11/31/62																							
0845	65	50	36	32	55	120	160	105	13	16 5	-	0 27	21	21	18 0	1 05	4 3	23 3	-	-	-	16 0	
0950	166	-	76	76	153	150	195	120	60	67	3 x 10 ⁻⁴ - 0 34		48	48	48 7	3 6	20 0	72 3	-	-	-	14 3	
1000	160	170	76	76	155	114	170	114	60 5	65	-	0 34	48	48	50 9	2 6	20 0	73 5	-	-	-	14 3	
1045	220	220	90	90	182	120	186	119	62	74	-	0 34	48	48	59 6	4 3	20 7	84 6	-	-	-	15 3	
1053	220	-	92	92	190	120	200	119	62	78	-	0 34	48	48	62 2	5 2	20 7	88 1	-	-	-	15 3	
1130	220	-	92	92	188	123	200	118	63	78	-	-	48	48	61 4	5 2	20 7	87 3	-	-	-	15 3	

RADIATION LEVELS IN REBOILER BUILDING AND SUB-REACTOR ROOM

(Test Report No. 126, dated November 27, 1962, by E. L. Martinec)

REBOILER PLANT

Radiation surveys were made in the reboiler plant during various phases of operation and at various reactor power levels. A summary of these surveys is presented in Table 126-I in conjunction with Fig. 126-1.

The high radiation levels in the reboilers were considered to be mostly from the large amounts of moisture carryover at high power levels. The No. 2 primary reboiler system was measured to be at a higher radiation level than the No. 1 system and this is attributed to the configuration of the piping system to the reboilers. The steam line to the No. 1 reboiler comes off the main steam line on a lateral while steam to the No. 2 reboiler continues straight ahead. Most of the moisture continues straight ahead into the No. 2 reboiler rather than make the turn into the No. 1 reboiler. As the moisture contains a large amount of radioactive corrosion products and other contaminants, the radiation level in the No. 2 system is naturally higher.

The radiation level exterior to the concrete wall of the primary reboiler cell reached 15 mr/hr at 2 in. from the wall. The intermediate steam line measured background activity level. Water samples taken from the intermediate side of the primary reboilers indicated no radioactivity present in the water. These water samples were taken periodically over the entire operation of the reactor.

BELOW REACTOR

A high neutron level was discovered in the rod drive room below the reactor. The radiation was being emitted from the forced circulation nozzles penetrating the bottom of the reactor shield. Apparently there was some neutron scattering occurring in the annulus between the forced circulation nozzles and the shield penetration sleeve (see Fig. 126-2). The neutron level prohibited any work in this area during high power operation.

To attenuate the neutron streaming down the annulus it was decided to fill the annulus with water. Measurements of neutron activity were taken while filling the annulus and are listed in Table 126-II. The water shielding was sufficient to enable work to be carried on below the reactor.

Table 126-1

RADIATION SURVEYS OF REBOILER SYSTEM

Date	<u>7/10/62</u>	<u>7/23/62</u>	<u>7/23/62</u>	<u>7/23/62</u>	<u>7/23/62</u>	<u>7/27/62</u>	<u>7/27/62</u>	<u>9/7/62</u>	<u>10/16/62</u>	<u>10/18/62</u>	<u>10/18/62</u>	<u>11/13/62</u>
Reactor Power, Mw	10	5	10	20	29	25	40	47	40	70	75	90
Reboiler Plant Power, Mw	10	5	10	20	29	25	40	47	40	70	66	70
<u>Location</u>												
1. No. 1 Prim. Reb. Head End	30	14	18	35	82	—	—	25	475	1.1r	600	800
2. No. 1 Prim. Reb. Center Bottom	—	—	—	60	135	—	—	50	320	1.2r	1r	1.5r
3. No. 1 Prim. Reb. Back End	15	5	25	30	25	—	—	25	80	240	250	450
4. No. 2 Prim. Reb. Head End	—	—	—	—	—	110	165	220	400	1.1r	650	1.1r
5. No. 2 Prim. Reb. Center Bottom	—	—	—	—	—	160	190	350	300	1.3r	1r	3.1r
6. No. 2 Prim. Reb. Back End	—	—	—	—	—	70	85	70	90	300	230	450
7. No. 1 Reb. Drain Tank Line	—	—	—	105	270	—	—	50	1r	2.5r	2.3r	2.7r
8. No. 1 Reb. Drain Tank Bottom	—	—	—	90	130	—	—	20	190	1r	500	1r
9. No. 2 Reb. Drain Tank Line	—	—	—	—	—	500	800	900	1.25	2.5r	2r	4.5r
10. No. 2 Reb. Drain Tank Bottom	—	—	—	—	—	120	300	400	1r	1.3r	900	1.6r
11. No. 1 Prim. Drain Cooler Back End	—	—	—	—	—	—	—	—	140	600	200	395
12. No. 1 Prim. Drain Cooler Center	—	—	—	—	—	—	—	—	45	—	—	310
13. No. 1 Prim. Drain Cooler Head End	7.5	—	—	60	140	—	—	25	40	220	160	430
14. No. 2 Prim. Drain Cooler Back End	—	—	—	—	—	—	—	—	100	600	170	400
15. No. 2 Prim. Drain Cooler Center	—	—	—	—	—	—	—	—	50	—	—	375
16. No. 2 Prim. Drain Cooler Head End	—	—	—	—	—	150	160	100	40	220	190	500
17. Primary Steam Line	70	25	40	60	110	100	125	160	250	900	850	230
18. Intermediate Steam Line	0	0	0	0	0	0	0	0	0	0	0	0
19. West Doorway to Cell	5	1.5	5	7	16	2.5	12	4	45	180	170	200
20. East Doorway to Cell	2	1.0	2.5	4	18	16	25	10	22	85	85	250

Note: Radiation readings are given in mr/hr unless otherwise noted

Readings were taken at 2 in. from surface.

Doorway readings are general background levels.

Measurement locations are shown in Fig. 126-1.

Fig. 126-1. EBWR Reboiler System Building Layout, 100 Mw Location of Radiation Measurement Points

Fig. 126-2. Forced Circulation Nozzle Configuration

Table 126-II

NEUTRON SURVEY OF THE SOUTHWEST CIRCULATION STUB

	Bottom of Bellows	Top of Bellows	Vessel Bottom	Power Level (Mw)	Water Level
N_f	1320	1235	—	~12	—
N_s	—	—	—	—	—
N_f	398	1081	—	~12	in bellows
N_s	151	464	—	~12	—
N_f	75	202	459	~12	above bellows
N_s	24	97	38	~12	—
N_f	20	33	36.1	~12	1 ft into shell
N_s	8	12	16	~12	1 ft into shell
N_f	14	22	25	10	1 ft into shell
N_s	9	9	14	10	1 ft into shell
N_f	29	49	53	20	1 ft into shell
N_s	18	18	29	20	1 ft into shell

N_f = fast flux, n/cm²/sec

N_s = thermal flux, n/cm²/sec

Time interval for readings was 5 sec.

SHORT DURATION TEST OF DEMISTER MATERIAL

(Test Report No. 111, dated February 13, 1962, by V. M. Kolba)

Samples of coarse and fine demister material, supplied by the York Corporation, were tested for 168 hr in 600 psig saturated steam flowing at 0.66 lb/min. The fine screen disintegrated and there was noticeable deterioration of the coarse screen.

Because of entrainment of water in the steam @ 100 Mw, leading to poor quality, a demister was proposed to improve the steam quality. The York Corporation proposed the use of a Type 304 stainless steel mesh demister. Since Type 304 stainless steel has associated chloride and stress corrosion problems it was decided to test the material. The anticipated flow rate of the EBWR steam through the proposed demister is ~ 3.75 ft/sec. To the degree attainable the test duplicated EBWR steam conditions.

The flow loop used is shown in Fig. 111-1. Steam was obtained from the large corrosion test autoclave in Building D-208. A 6-in. section of

Fig. 111-1. Demister Material Test Flow Loop

coarse mesh followed by a 1 in. section of fine mesh were placed in a piece of 1 in. Schedule 40, Type 304 stainless steel pipe. The test section was inclined at a 30-degree angle so that water collected by the demister would drain back into the autoclave. The steam flow passed through the coarse mesh first and then through the fine mesh. The steam leaving the test section was condensed by passing it through a regenerative cooler and then a secondary cooler. The condensate then passed through a regulating valve and flow meter and then into a tank at atmospheric pressure. The condensate was then pumped through the regenerative cooler for preheat and then into the autoclave to be recycled. The secondary cooler was cooled by tap water. The water in the autoclave at the beginning of the test was demineralized, deaerated water which had a resistivity of ~ 4 megohms. A day by day check of the water conductivity was made during the test. At the end of the 168 hour test the resistivity of the water was > 1 megohm and the Cl^- concentration was < 0.1 ppm as determined by a Taylor Comparator and the Enslo Slide Chlorimeter.

The mesh in the test section was in stagnant steam for $\sim 1\text{-}1/2$ days during the startup of the autoclave and $\sim 1\text{-}1/2$ days during the cooling down of the autoclave. The mesh was exposed to flowing steam for 168 hours at a steam flow rate of 0.662 lb/min equivalent to a velocity of 1.38 ft/sec through the 1-in. Schedule 40 pipe. Assuming $> 90\%$ porosity of the mesh there was no appreciable velocity change induced by the mesh. This velocity is less than that expected in the EBWR demister, where the bulk steam velocity in the demister area is ~ 3.5 ft/sec, without allowance for the Δp and small change in velocity due to the demister mesh.

The coarse mesh, Fig. 111-2, appeared to be in generally acceptable condition though there was a moderate amount of reaction product on the inside mesh; see Fig. 111-3. The fine mesh, however, had completely disintegrated and came out of the test section in a pile of loose "fines" as shown in Fig. 111-2. The coarse mesh in Figs. 111-2 and -3 had expanded from its compacted dimensions in the test section.

One may speculate as to the reason for the failure of the mesh, and an obvious choice is stress corrosion of one form or another. Since the prior history of the mesh is not known it is difficult to attribute any specific cause to the failure. The Cl^- ion concentration is low < 0.1 ppm. However, the O_2 concentration during the test was not known. The test was started with de-aerated water but the water could have picked up oxygen at the atmospheric tank prior to its return to the autoclave.

Should one desire to use a stainless steel mesh for the demister it is recommended that additional tests be run with material for which prior history is available and a close control of the oxygen content of the water be maintained.

Fig. 111-2. Demister Material (After Test)

Fig. 111-3. Coarse Mesh Demister Material (After Test)

It should be noted that the order for demister material for the EBWR demister specified Inconel wire for the mesh, with a mesh density of $\sim 12 \text{ lb./ft}^3$ i.e. $\sim 97.5\%$ mesh porosity.

SECOND TEST OF DEMISTER MATERIAL

(Test Report No. 111A, dated July 12, 1962, by V. M. Kolba)

A sample of coarse Inconel demister material supplied by Otto York Company was tested for 415.5 hours in 600 psig saturated steam flowing at 0.66 lb/min. This material is the same as that to be supplied for the EBWR demister presently on order. No disintegration or major deterioration of this material was noted.

Because of the results of the tests reported in Reference 1, the decision was made to test the Inconel demister screen supplied by the York Company. The nominal composition of Inconel is:

Ni 78.00	Cu 0.25	Si 2.00	C 0.20
Cr 13.0	Fe 6.00	Mn 0.75	S 0.010

The flow loop used was the same as that shown and described in Reference 1 with the same tests made of the water, etc., as described in that report. Differences between the previous test and this test were that no fine mesh was in the test section and the conductivity of the water was $>$ than 8 megohms for the duration of the test.

The mesh in the test section was in stagnant steam for ~ 166 hours during the various startup and shutdown cycles of the test. The frequent starts and stops (Table 111A-I) of this test were in conjunction with the testing of Borax V control rods in the autoclave used as the steam supply for the demister tests. The mesh was exposed to flowing 600 psig saturated steam for ~ 415.5 hours at a flow rate of 0.66 lb/min equivalent to a velocity of $\sim 1.38 \text{ ft/sec}$ through the 1 in. schedule 40 pipe. Prior to the test the mesh weighed 25.8713 gm and after the test it weighed 25.8681 gm. The change in weight was -0.0032 gm. The weight change of 0.01% is very low, probably within the normal handling loss. The mesh was still ductile after the test with no embrittlement noted. Visually, the mesh appeared in good condition with only slight temperature discoloration noted.

Material of this type is being supplied for the EBWR demister on order.

Table 111A-I

DEMISTER TESTS

<u>Date</u>	<u>Time</u>	<u>Temperature, °F</u>	<u>Pressure, psi</u>	<u>Remarks</u>
4/31/62	12:00	400	—	Started steam flow
4/31/62	16:00	486	—	Shut down
5/1/62	08:30	486	—	Started steam flow
5/7/62	08:30	486	—	Shut down
5/7/62	16:30	~325	100	
5/28/62	13:00	78	—	Started
5/29/62	15:00	—	400	Started steam flow
5/30/62	—	—	400	
5/31/62	17:00	486	600	
5/31/62	21:00	486	—	Shut down
6/1/62	09:00	78	—	Started
6/1/62	15:00	—	200	
6/3/62	06:00	—	400	Started steam flow
6/4/62	13:00	486	600	
6/15/62	08:30	486	600	Shut down

REFERENCE

- (1) EBWR Test Report No. 111.

Section VII. APPENDICES

	<u>Title</u>	<u>Page</u>
A.	EBWR Control Rod - Fuel Element Location (Memo, dated March 7, 1962, by V. M. Kolba).	517
B.	Investigation of SA-212 Grade B Material Removed from the EBWR Pressure Vessel Wall (R.E. No. 763) (Memo, dated April 24, 1961, by T. L. Kettles)	518
C.	Summary Report - Investigations Into Causes for Catastrophic Pre-Operational Failures of the 80-Mw EBWR Primary Reboilers (Memo, dated June 29, 1962, by N. Balai, J. F. Matousek, C. R. Breden, and M. Odie)	536
D.	Summary of EBWR Test Procedures and Reports during the Period 1960-1962.	566

APPENDIX A EBWR CONTROL ROD - FUEL ELEMENT LOCATION

(Memo, dated March 7, 1962, by V. M. Kolba)

Periodically the question arises as to the relative axial location of the active fuel and the poison section of the control rods when in or full out position. Figure A-1 provides this information. Reference dimensions on the right hand side are based on the 705 ft-0 in. Working Point reference at the reactor vessel bottom.

Fig. A-1. Relative Location of Fuel Zone and Control Rod Poison Section

APPENDIX B

INVESTIGATION OF SA-212 GRADE B MATERIAL REMOVED FROM
THE EBWR PRESSURE VESSEL WALL

(R.E. No. 763)

(Memo dated April 24, 1961 by T. L. Kettles)

When the EBWR pressure vessel was modified for operation at 100 Mw(t) two plugs were removed from the wall for the installation of a larger steam outlet nozzle and an additional feedwater supply nozzle. Mechanical properties of the plug material were obtained for comparison with the properties of virgin material.

The comparison revealed small differences which were attributed to effects of fabrication. No part of these differences was attributed to effects of nuclear radiation.

TEST SPECIMENS

Twenty-seven tensile and 165 notched bar impact specimens were obtained to compare mechanical properties. They were machined from the steam plug, feedwater plug, and a piece of virgin plate, part of that used to form the lower EBWR vessel head, in accordance with cutting diagrams, Figs. B-1, 2, and 3.

The tensile specimens were 1/4 in. nominal diameter in accord with ASTM E8-57T and the impact specimens were standard size Charpy V-notch in accord with ASTM E23-56T. Dimensions of these specimens are shown on Fig. B-1.

TESTS

The tensile specimens were pulled to provide measures of ultimate tensile strength, yield strength, per cent elongation, per cent reduction in area, and fracture strength. Tensile test procedures are described, nomenclature is defined, and data are tabulated in Attachment A.

The impact specimens were broken to provide measures of transition between brittle and ductile fractures. Impact test procedures are described, nomenclature is described, and data are tabulated in Attachment B.

MILL TEST REPORT DATA

MATERIAL - CARBON STEEL SA-212, GRADE B
 HEAT NO. - LUKENS NO. 22789
 ORDER SIZE - 2-3/8" x 11 1/4" x 273"
 CHEMISTRY: C-.26 P-.018 Si-.18 Mn-.75 S-.029
 PHYSICALS: YIELD STRENGTH - 41,900 psi
 TENSILE STRENGTH - 74,300 psi
 ELONGATION (2 inches)-34%

*F.R.D. - FINAL ROLLING DIRECTION

.250" DIA. TENSILE SPECIMEN

CHARPY "V" NOTCH SPECIMEN

NOMENCLATURE

- L - SPECIMEN TAKEN LONGITUDINAL WITH VESSEL AXIS. ALSO IDENTIFIES DIRECTION OF "V" NOTCH FOR RADIAL CHARPY'S.
- T - SPECIMEN TAKEN TANGENTIAL TO VESSEL SHELL, ALSO IDENTIFIES DIRECTION OF "V" NOTCH FOR RADIAL CHARPY'S.
- R - SPECIMEN TAKEN RADIAL WITH VESSEL AXIS (THROUGH SHELL WALL). ALSO IDENTIFIES DIRECTION OF "V" NOTCH FOR SOME TANGENTIAL CHARPY'S.
- I - SPECIMEN LOCATED 1/4 WALL THICKNESS FROM THE INSIDE SURFACE OF THE VESSEL SHELL.
- M - SPECIMEN LOCATED IN THE MIDDLE OF THE VESSEL WALL THICKNESS.
- O - SPECIMEN LOCATED 1/4 WALL THICKNESS FROM THE OUTSIDE SURFACE OF THE VESSEL SHELL.

RE-1-34194-C

Fig. B-1. Location of Test Specimens from EBWR Vessel Feedwater Plug

RE-1-34195-C

Fig. B-2. Location of Test Specimens from EBWR Vessel Steam Plug

BLOCK CUTTING DIAGRAM

MILL TEST REPORT DATA

MATERIAL - CARBON STEEL; SA-212, GRADE B
 HEAT NO. - LUKENS NO. 22789 (SAME AS VESSEL SHELL)
 ORDER SIZE - 3-15/16" x 121" x 121"
 CHEMISTRY: SEE FIG.
 PHYSICALS: YIELD STRENGTH - 42,500 psi
 TENSILE STRENGTH - 76,000 psi
 ELONGATION (2 inches)-34%

* F.R.D. - THE FINAL ROLLING DIRECTION SHOWN WAS DETERMINED BY SURFACE MARKS ON THE METAL BLOCK. HOWEVER, THE TEST DATA REVEALS THAT THE DIRECTION MARKED (T) WAS MOST PROBABLY THE PRINCIPAL ROLLING DIRECTION

L-LONGITUDINAL CUTTING DIAGRAM

T-TRANSVERSE CUTTING DIAGRAM

NOMENCLATURE

L - SPECIMEN TAKEN ALONG F.R.D.*
 T - SPECIMEN TAKEN TRANSVERSE TO F.R.D.*
 R - SPECIMEN TAKEN THROUGH PLATE THICKNESS
 M - SPECIMEN LOCATED NEAR MIDDLE OF PLATE THICKNESS
 O - SPECIMEN LOCATED NEAR OUTER SURFACES OF PLATE

GENERAL NOTE

I - FOR SPECIMEN DETAILS SEE FIG. B-1

RE-1-34196-C

Fig. B-3. Location of Test Specimens from Original EBWR Vessel Head Material

Fig. B-4

EBWR Pressure Vessel Showing Locations Where Plugs Were Removed for Installation of a New Steam Outlet Nozzle and an Additional Feedwater Supply Nozzle

EFFECTS OF NUCLEAR RADIATION ON MATERIAL PROPERTIES

Comparison between Measured Mechanical Properties of Steam and Feedwater Plugs

The locations where the steam and feedwater plugs were removed from the vessel wall are shown on Fig. B-4. The feedwater plug is sufficiently closer to the core so that any measurable changes to its material properties caused by nuclear radiation would be reflected in a difference in mechanical properties from those of the steam plug material.

Average tensile properties of the two plugs, longitudinal (L) to the principal rolling direction, at quarter point (1/4) wall thickness locations are compared in Table B-I below.

Table B-I

AVERAGE TENSILE PROPERTIES

Plug (L-1/4)	No. of Specimens	Y.S.* 0.2% Offset (psi)	U.T.S.†	EL.‡ in 1-in.(%)	R.A. # (%)
Steam	4	45,300	77,700	34	56
Feedwater	4	45,100	77,800	31	56

* Yield Strength

† Ultimate Tensile Strength

‡ Elongation

Reduction in Area

The data are the same, within limits of normal material variation and experimental error.

Energy absorbed at the break, by Charpy V-notch specimens cut from the two plugs, versus the temperature at which they were broken are shown by Plot 1 on Fig. B-5. A single curve fits the data from both plugs.

Fig. B-5

Transition Curves (Plots 1 and 2)

Table B-II

MEASURED PROPERTIES OF VESSEL WALL VERSUS REPORTED PROPERTIES OF VIRGIN PLATE OF SAME THICKNESS

Source	No. of Specimens	Y.S.* 0.2% Offset (psi)	U.T.S.* (psi)	Elongation in 1-in. (%)	R.A.* (%)
Steam & Feed- water Plugs	8	45,200	77,800	32	56
Vessel Wall Virgin Plate	-	41,900	75,400	-	-

* See Table B-8.

Summary

Tensile and impact data revealed no differences in material properties of the steam and feedwater plugs. It is concluded that mechanical properties of the EBWR pressure vessel, at or above the new feedwater nozzle, have not been affected by nuclear radiation.

EFFECTS OF VESSEL FABRICATION ON MATERIAL PROPERTIES

Comparison of Measured Mechanical Properties of Vessel Wall Versus Documented Properties of Virgin Plate

Tensile properties of the virgin plate, fabricated into the vessel wall, were provided by the plate manufacturer. This data is compared in Table B-II below with the measured properties of the two plugs, longitudinal to the principal rolling direction, at quarter point wall thickness location.

Comparison of Measured Mechanical Properties of Vessel Wall Versus Measured Properties of Virgin Plate from Same Mill Heat

The EBWR vessel wall and lower head were fabricated from mill heat lukens No. 22789 of the following weight-percent chemical analysis:

W/O				
C	Mn	Si	P	S
0.26	0.75	0.18	0.018	0.029

The wall plates measured 2-7/16 in. x 114 in. x 273 in; and the lower head plate measured 3-15/16 in. x 121 in. x 121 in. The difference in thickness adds the effect of greater hot work to the effect of vessel wall fabrication in the comparison. Material tensile properties in Table B-III below are transverse (T) to the principal rolling direction, at quarter point (1/4) wall thickness location.

Table B-III

MEASURED PROPERTIES OF VESSEL WALL VERSUS THICKER VIRGIN PLATE

Source (T-1/4)	No. of Specimens	Y.S.* 0.2% Offset (psi)	U.T.S.* (psi)	Elongation in 1-in. (%)	R.A.* (%)
Steam and Feed- water Plugs	8	45,000	77,200	30	51
Lower Head Virgin Plate	2	36,300	72,100	32	54

* See Table B-I.

Energy absorbed at the break, by Charpy V-notch specimens from the lower head virgin plate, versus the temperature at which they were broken are shown by Plot 2 on Fig. B-5. In the figure, plot 1, of specimens from the vessel wall, has been superimposed for comparison.

Summary

The first comparison above reveals that fabrication of virgin plate into vessel wall configuration caused small increases in material strength and the second comparison supports that conclusion. The second comparison further reveals a small decrease in ductility, a 20°C reduction in fracture appearance transition temperature, and no change in ductility (15 ft-lb) transition temperature.

Examination of data in Attachments A and B support the following additional conclusions:

1. Vessel wall material near the outside surface was strengthened more than material near the inside surface (stretching versus compression).
2. Vessel wall capacity to absorb impact above the fracture appearance transition temperature is relatively high, intermediate, and low, tangential to the circumference, parallel to the vessel axis, and radial to the vessel axis, respectively.
3. The fracture appearance transition temperature of the vessel wall is isotropic within the accuracy of visual measurement.

GENERAL OBSERVATIONS REGARDING FRACTURE APPEARANCE AND NOTCH ORIENTATION OF IMPACT SPECIMENS

Fracture Appearance

Fracture appearance, above nil ductility, can be qualitatively correlated with absorbed energy by considering the fibrous nature of the material. Fibers developed along the principal rolling direction are more elastic and are entitled "primary," those developed transverse to the principal rolling direction are less elastic and are entitled "secondary."

Charpy specimens are broken as beams simply supported and centrally loaded. Longitudinal specimens contain primary fibers in an axial direction and absorb more energy at break than transverse specimens which contain secondary fibers in the axial direction. Radial specimens which have no axial fibers break with low energy absorption.

When longitudinal specimens are broken, above nil ductility temperatures, the secondary fibrous structure is revealed. When transverse specimens are broken the primary fibrous structure is revealed. When radial specimens are broken the fracture surface appears barklike with only a minor fibrous structure, but always primary. As the temperature at the break is increased fibrous definition becomes less pronounced for all types of specimens. Figures B-6 through 9 show these dependencies of fibrous orientation on original sample geometry and fibrous definition on temperature at break.

Notch Orientation

The majority of impact specimens broken for the subject investigation had notches oriented parallel to the original plates surface, analogous to the

way a pressure vessel ruptures. ASME specification SA-300, Steel Plates for Pressure Vessels for Service at Low Temperatures required the notches to be oriented perpendicular to the plate surface.

Plot 4 on Fig. B-10 shows a transition curve for the vessel wall based on data from longitudinal specimens with notches oriented parallel to the plate surface. Plot 3 on Fig. B-10 shows a curve based on data from similar specimens except that the notches were oriented perpendicular to the plate surface. Plot 3 has been superimposed on plot 4 for comparison.

Fig. B-6. Appearance of Charpy V-Notch Impact Specimen Fractures (Above Transition Temperature) Correlated with Specimen and Notch Orientation

Fig. B-7. Longitudinal Specimens, Radial and Transverse Notches, Vessel and Base Material, Quarter and Mid-Thickness Locations

Fig. B-8. Transverse Specimens, Longitudinal Notch, Vessel and Base Material, Quarter and Mid-Thickness Locations

Fig. B-9. Radial Specimens, Longitudinal and Transverse Notches, Vessel and Base Material

Both sets of data reveal the same fracture appearance and ductility (15 ft-lb) transition temperatures. The only difference is that specimens notched parallel absorb greater energy in the higher temperature range than do the specimens notched perpendicular to the plate surface. The reason may lie in the relation of fiber direction at break.

* SPECIMEN LOCATIONS - QUARTER POINTS OF WALL THICKNESS
SPECIMEN DIRECTION - PARALLEL TO P.R.D.
SPECIMEN NOTCH ORIENTATION - TRANSVERSE TO P.R.D.
(PLOT 4) AND PERPENDICULAR TO WALL (PLOT 3)
FRACTURE APPEARANCE - TRANS. TEMP. 40/45 °C

Fig. B-10

Transition Curves for EBWR Vessel Wall (Plots 3 and 4)

Both sets of specimens are broken across axial primary fibers. The specimens absorbing higher energies break normal to secondary fibers in the fracture plane while those absorbing lower energies break end-on to the secondary fibers.

Attachment A

TENSILE TEST PROCEDURES

The 1/4-in. nominal diameter specimens were pulled through the yield at a load rate of 1000 lb/min on a Baldwin Universal Testing Machine (No. 68150) of 60,000 lb capacity. Maximum load was read from the dial, and the load at the break was approximated from the dial at the sound of the break.

The load-stain relationship was traced to 0.006-in. strain by use of a Baldwin Type P3M 47 averaging extensionmeter at a 500/1 magnification ratio. A typical trace is shown on Fig. B-11.

TENSILE TEST NOMENCLATURE

The mechanical properties, ultimate tensile strength, yield strength

at 0.2% offset, per cent elongation in 1-in., and percent reduction in area were calculated in accordance with ASTM specification E6-58aT. Fracture strength was calculated by dividing the fracture area into the load approximated from the dial at the sound of the break.

The terms (and abbreviations) employed to identify the specimens (as used in Table B-IV) are defined below:

Steam Plug (S) - The plug removed from the vessel wall for installation of a new steam outlet nozzle.

Feedwater Plug (F) - The plug removed from the vessel wall for installation of an additional feedwater nozzle.

Baseplate (B) - The EBWR lower head virgin plate material.

Fig. B-11

Typical Load-Strain Trace for
EBWR Specimens Showing Mechanics
Employed to Obtain Data

Longitudinal (L) - Parallel to the principal rolling direction. For the vessel wall this direction is circumferential.

Transverse (T) - Perpendicular to the principal rolling direction. For the vessel wall this direction is parallel to the vessel axis.

Radial (R) - Perpendicular to the plate surface. For the vessel wall this direction is radial to the vessel axis.

Inside (I) - One-quarter the wall thickness from the inside vessel surface.

Outside (O) - One-quarter the wall thickness from the outside vessel surface.

Quarter-Point (1/4) - One-quarter the virgin plate thickness from either surface, or the average of inside and outside vessel wall data.

Middle-Point (M) - The middle of vessel wall or virgin plate thickness.

Table B-IV

TENSILE TEST DATA

Specimen Identification	Y.S.* 0.2% Offset (1000 psi)	U.T.S.* (1000 psi)	Elongation in 1-in. (%)	R.A.* (%)	Fracture Strength (1000 psi)
(S) (L) (I)	42.6	76.6	34	57	140.2
	44.6	77.4	34	56	139.8
(F) (L) (I)	44.1	77.1	30	57	140.8
	44.2	77.0	31	57	141.2
(S) (L) (O)	46.6	77.9	31	54	134.3
	47.5	79.0	35	55	140.9
(F) (L) (O)	46.4	78.8	33	54	136.0
	45.6	78.4	30	55	140.1

* See Table B-I.

Table B-IV (Cont'd)

Specimen Identification	Y.S.* 0.2% Offset (1000 psi)	U.T.S.* (1000 psi)	Elongation in 1-in. (%)	R.A.* (%)	Fracture Strength (1000 psi)
(S) (T) (I)	— 44.2	76.5 74.8	32 32	52 54	— —
(F) (T) (I)	44.2 45.0	77.6 77.4	27 30	48 49	124.0 130.6
(S) (T) (O)	47.2 46.9	78.3 78.0	27 30	50 50	— 125.4
(F) (T) (O)	43.7 45.1	76.2 78.5	29 28	50 48	129.6 129.7
(S) (L) (M)	39.8 40.5	73.1 74.1	31 30	44 48	107.9 115.4
(F) (L) (M)	39.3 38.7	73.2 73.0	31 26	48 45	116.7 113.4
(S) (T) (M)	— —	72.3 75.6	31 27	44 42	— —
(F) (T) (M)	39.4 —	72.3 71.5	25 27	36 36	102.3 98.7
(B) (T) (1/4)	36.1 36.4	71.5 72.7	34 29	55 53	— 126.1
(B) (T) (M)	35.6	71.3	34	53	126.3

* See Table B-I.

Attachment B

IMPACT TEST PROCEDURES

The Charpy V-notch specimens were broken on a Riehle Impact Machine (No. R-15921) set for an initial hammer energy of 110 ft-lb with a striking velocity of 13 ft/sec. Specimen temperatures were within the range from (-) 40 to 150°C, when broken, to provide data for plotting brittle to ductile energy transitions. The mediums employed to obtain the temperatures are listed below:

(-) 40 to 0°C	Dry ice (CO ₂) in acetone.
0°C	Ice in water
0 to 100°C	Water
100°C	Boiling water
100 to 150°C	Oil

IMPACT TEST NOMENCLATURE

Two criteria are used to rate the impact test data:

1. Fracture Appearance Transition Temperature - "the temperature where the fracture shows a 50% crystalline and a 50% fibrous appearance" (ASTM E 23-56T).
2. Ductility Transition Temperature - "the temperature corresponding to a specific energy value" (ASTM E 23-56T).

The energy value used for (1) above is 15 ft-lb for relation with ASME specification SA-300, Steel Plates for Pressure Vessels for Service at Low Temperatures. The subject data is not strictly comparable however, because V-notch specimens were used (in accord with the trend in this country) whereas the requirements of specification SA-300 are based on data from keyhole and U-notch specimens. Parker⁽¹⁾ has shown that V-notch specimens provide higher ductility transition temperatures than keyhole notch specimens.

The terms employed to identify the impact specimens in Table B-V following are as defined for the tensile specimens in Attachment A.

Table B-V

IMPACT TEST DATA

Specimen Identification	Temp. (°C)	Absorbed Energy (ft-lb)	Fracture Appearance (% Fibrous)
<u>Vessel Wall, L-1/4, Transverse Notch</u>			
S-O	-40	6	0
S-O	0	14	5
S-I	0	25	5
S-O	27	23	5
S-9	27	24	10
S-O	45	-	70
S-I	61	60	80
S-I	93	72	100
S-O	93	75	100
S-I	100	78	100
S-I	120	76	100
S-O	151	70	100
<u>Vessel Wall, L-1/4, Radial Notch</u>			
F-O	-40	3	0
F-9	0	10	5
F-I	0	13	5
F-I	0	23	5
S-I	0	20	5
S-I	25	40	20
F-I	25	19	20
F-I	25	27	10

Table B-V (Cont'd.)

Specimen Identification	Temp. (°C)	Absorbed Energy (ft-lb)	Fracture Appearance (% Fibrous)
F-O	25	25	5
F-O	45	37	70
F-O	60	43	60
F-I	61	44	60
S-O	62	47	60
S-O	93	59	99
F-I	94	57	100
F-O	94	42	95
F-O	95	64	100
F-I	100	60	99
F-I	120	59	100
F-O	150	59	100
<u>Vessel Wall, T-1/4, Longitudinal Notch</u>			
S-O	-40	5	0
F-O	-40	5	0
F-O	0	9	1
S-I	0	19	5
S-I	0	11	5
S-O	0	14	0
F-O	25	19	10
F-I	25	25	15
S-I	28	23	10
S-O	28	23	10
S-O	45	37	50
F-O	45	26	60
S-I	60	42	70
F-I	62	38	70
F-I	94	48	100
F-O	94	49	100
S-O	94	51	100
S-I	94	47	99
S-I	100	55	100
F-I	100	51	100
F-I	120	45	100
S-I	120	48	100
F-O	149	45	100
S-O	150	43	100
<u>Vessel Wall, L-M, Transverse Notch</u>			
S	0	18	5
	27	15	20
	45	31	60
	62	41	90
	94	45	100
	155	40	100

Table B-V (Cont'd.)

Specimen Identification	Temp. (°C)	Absorbed Energy (ft-lb)	Fracture Appearance (% Fibrous)
<u>Vessel Wall, L-M, Radial Notch</u>			
F	0	14	5
	25	10	10
	45	32	60
	60	34	70
	61	36	60
	62	38	90
	94	37	100
	150	38	100
	150	37	100
	152	39	100
<u>Vessel Wall, T-M, Longitudinal Notch</u>			
S	0	9	5
F	0	8	5
F	25	23	30
S	28	19	10
S	45	27	60
F	45	29	70
F	61	35	99
S	61	34	95
S	94	37	100
F	94	38	100
F	150	38	100
S	152	32	100
<u>Vessel Wall, R, Longitudinal Notch</u>			
S	-40	3	0
F	-40	3	0
F	0	4	0
S	0	4	0
F	23	8	20
F	23	7	20
F*	23	6	10
F*	23	6	10
F*	23	7	10
S*	23	5	10
S	25	7	10
F	25	8	20
F	45	13	60
S	45	12	90
F	60	14	95
S	61	13	95
S	93	16	100
F	94	16	100

* - These specimens were broken at an initial hammer energy of 220 ft-lb with a striking velocity of 18 ft/sec.

Table B-V (Cont'd.)

Specimen Identification	Temp. (°C)	Absorbed Energy (ft-lb)	Fracture Appearance (% Fibrous)
F	100	16	100
S	100	15	100
S	120	16	100
F	120	16	100
F	150	15	100
S	150	16	100
<u>Vessel Wall, R, Transverse Notch</u>			
S	-40	3	0
F	-40	3	0
F	0	5	1
S	0	6	5
S	23	8	20
S	23	8	20
S*	23	6	5
S*	23	6	5
S*	23	6	5
F	23	10	60
F*	23	6	30
F*	23	6	40
F	25	9	50
S	25	9	20
S	45	13	80
F	45	12	60
F	60	14	95
S	60	14	95
S	93	15	99
F	94	15	100
F	100	--	100
S	100	15	99
S	120	15	100
F	120	14	100
F	150	14	100
S	158	16	100
<u>Lower Head Virgin Plate, L-1/4, Transverse Notch</u>			
B	-40	5	0
	0	8	0
	24	31	5
	61	54	50
	94	66	70
	100	81	100
	120	65	100
	150	72	100

* - These specimens were broken at an initial hammer energy of 220 ft-lb with a striking velocity of 18 ft/sec.

Table B-V (Cont'd.)

Specimen Identification	Temp. (°C)	Absorbed Energy (ft-lb)	Fracture Appearance (% Fibrous)
<u>Lower Head Virgin Plate, T-1/4, Longitudinal Notch</u>			
B	-40	4	0
	0	8	1
	24	23	5
	61	43	30
	94	52	70
	100	78	90
	120	66	100
	150	59	100
<u>Lower Head Virgin Plate, L-M, Transverse Notch</u>			
B	0	6	0
	24	17	5
	61	37	20
	80	54	70
	95	45	80
	150	50	100
<u>Lower Head Virgin Plate, T-M, Longitudinal Notch</u>			
B	0	9	0
	24	16	5
	61	49	50
	80	52	80
	94	50	90
	150	49	100
<u>Lower Head Virgin Plate, T-1/4, Radial Notch</u>			
B	24	23	5
	24	14	5
<u>Lower Head Virgin Plate, R, Transverse Notch</u>			
B	24	10	5

REFERENCE

- (1) E. R. Parker, "Brittle Behavior of Engineering Structures," John Wiley and Sons Inc., New York (1957), pp. 116 and 153.

APPENDIX C

SUMMARY REPORT - INVESTIGATIONS INTO CAUSES FOR CATASTROPHIC
PRE-OPERATIONAL FAILURES OF THE 80-Mw EBWR PRIMARY REBOILERS

(Memo, dated June 29, 1962, by N. Balai, J. F. Matousek,
C. R. Breden, and M. Odie)

INTRODUCTION

Condensed Pre-Operational Reboiler Chronology

The purpose of this report is to present the results of the mechanical, corrosion, and metallographic examinations of the seamless austenitic stainless steel reboiler tubing following the discovery of a leaking tube during the field hydrostatic tests of connected pressure piping. Leakage was wholly unexpected and startling because neither of the two units showed signs of leakage in the shop hydrostatic tests prior to cleaning and shipping. On removal of the inspection manhole plates, unexpected deposits of rust were found in the tube boxes, roughened tube boxes, and black magnetic debris were also found in "clean" equipment -- equipment which was passivated for two hours with 20% nitric acid at 180°F. Further hydrostatic tests, at lower pressure, from the shell side revealed the presence of numerous tube-to-tube sheet weld "weepers." Approximately 80% of the leaks were found in the upper half of the tube bundles.

Repairs attempted in the field by a fabricator's qualified welder failed; tube welds immediately adjacent to repair areas started to leak, i.e., in regions which were tight previously. Both units were then disconnected from the piping and returned to the fabricator's shop for repair. All shop repair attempts failed. New tube sheets and welded 304L tubing (substituted for the original 304 seamless tubing) were incorporated into the re-manufacture of the reboilers. The re-manufactured units passed all shop and field hydrostatic tests and were placed in service.

In the elapsed period, time for reboiler repair attempts and re-manufacture, various samples of unused tubing remnants and samples of tubing removed from both reboilers were examined for the causes of the failure in the two units. Several unusual tubing characteristics were found which explained the repair failures and the catastrophic failures of the reboiler units. The lessons learned from the results of the investigation were applied in the re-manufacturing operations.

Reboiler Performance Requirements and Design Criterion

The EBWR was designed originally to produce 20 Mw(t), about 60,000 lb/hr of steam at 600 psig, with light water. Experiments with the 48-in.

Fig. C-1. Flow Diagram of EBWR 100-Mw System - RE-8-31888-F

diameter core revealed stable reactor operations at the 61.7-Mw power level. Extrapolations indicated that a 60-in diameter core, which could be accommodated on the original grid plate inside the reactor vessel, would generate 100 Mw(t) of heat. To permit the high power reactor stability studies, the 20-Mw capacity plant was modified by the addition of auxiliary heat dissipation equipment, i.e., a reboiler plant to supplement the existing turbine plant. The reboiler plant was designed as a multiple barrier system to dump 80 Mw of thermal energy, in the form of high pressure steam, into one of two heat sinks operated at lower pressures and temperatures. The output of the reactor, steam at 600 psig, was condensed in two primary reboiler units. The output of the primary reboilers, steam at an intermediate pressure of 350 psig, was then directed to the two heat sinks:

- (1) secondary reboilers
- (2) air-cooled condensers

Figure C-1 illustrates the general mechanism for the dissipation of 100 Mw of reactor heat.

Each reboiler unit is a conventional type U-tube bundle heat exchanger designed for vapor generation. The construction of a reboiler unit is illustrated in Fig. C-2. The operating and design conditions for each unit are summarized in Table C-I.

Fig. C-2. Primary Reboiler - RE-6-32683-D

Table C-I

SUMMARY OF DESIGN OPERATING CONDITIONS FOR EACH PRIMARY REBOILER UNIT

Reboiler tube surface	4000 sq ft
Heat exchanged - maximum	111×10^6 Btu/hr (32.5 Mw)
Quantity of primary steam to tube side	150,000 lb/hr (max)
Tube side pressure - operating	560 psig
- design	800 psig
- test	1500 psig
Tube side temperature - operating (both steam inlet and condensate out)	482°F
- design	520°F
Quantity of intermediate steam leaving shell side	140,500 lb/hr (max)
Shell side pressure - operating	350 psig
- design	400 psig
- test	600 psig
Shell side temperature - operating (both water inlet and steam outlet)	435.7°F
- design	448°F
Steam quality leaving shell dry pipe	98% (approx.)
Fouling factor	0.0005
Shell side water capacity at normal operating conditions	1925 gal
Weights - dry	31,500 lb
- operating	47,500 lb
- flooded	59,500 lb

The reboiler design conforms to the design requirements of Section VIII of the ASME Code and with TEMA standards for Class A construction. The shell, tube sheet, and channel are welded to form an integral unit without flanged joints. As indicated in Table C-II, all materials of construction are carbon steel except in the primary-reactor fluid circuit. All surfaces in contact with the primary fluid are austenitic stainless steel. Access to the shell and channel end permit inspection of the tube bundle for leakage at the tube-seal tube sheet welds, corrosion or erosion of the tubes, and stress cracking of the visible tubes.

Table C-II

SUMMARY OF PRIMARY REBOILER MATERIALS OF CONSTRUCTION

Shell	ASTM-A-212 Gr B
Shell weld stubs	ASTM-A-105 Gr 1
Channel	ASTM-A-212 Gr B, with minimum 3/16-in.-thick arc-deposited AISI-304 cladding on inside surface
Channel weld stubs	ASTM-A-240, Type AISI 304
Tube Sheet	ASTM-A-212 Gr B, with minimum 3/16-in.-thick arc-deposited AISI-304 cladding on tube side

Table C-II (Cont'd.)

Channel partition rib and cover	ASTM-A-240, Type AISI 304
Tubes	ASTM-A-249, Type AISI 304L
Shell handholes and weld coupling	Steel, ASTM-A-105 Gr 1

The tube bundle is of a two-pass U-tube design. The tubes are rolled and seal welded to the tube sheet to eliminate leakage of the primary fluid into the intermediate system. An unusual feature of the design is the extreme length of straight tubing, approximately 50 ft long, required for the fabrication of the U-bends.

SUMMARY OF TESTS

Types of samples received for tests were:

Virgin Metal Tubing

Two pieces 12-1/2 in. long, with location in 50 ft length of tube unknown.

Virgin Metal Tubing

Two pieces of 12 in. long and one piece about 40 in. bent into an unequal legged 180 degree U-bend. The two short pieces were "close" to the 180 degree U-bend. Material, therefore, was from the middle of the only tube rejected during manufacture. (Tube was rejected for "wrinkling" on the inside of the 180 degree U-bend.)

Reboiler No. 1 Tubing

One half of total tube approximately 50 ft long from O.D. of tube bundle. Tube was slit longitudinally to facilitate inspection of tube bore. Tube cut into sections about 42 in. long for shipment to ANL; all pieces were identified for sequence in reconstitution of tube. Tube had no previous history of leakage at ANL or of repair procedures prior to its removal, but it was located immediately adjacent to a tube which developed a split about 1 in. long.

Reboiler No. 2 Tubing

Three pieces 9 to 10 in. long, located at top, middle, and bottom positions in the upper half of tube bundle, adjacent to the tube sheet, and one unusual piece about 10 in. long from an outer row which protruded from the tube sheet after shop salvaging of the scrap stainless tubing. This fourth piece was particularly noticed because of its obviously embrittled condition; it is separately identified as a "smashed" tube. All tube samples were immediately adjacent to the tube sheet and approximately 5-1/2 in. away from tube-to-tube sheet welds.

Visual Examinations

Visual examinations of tubing samples received revealed that all the material was marked as required by the ASTM Standard A-213, which requires: (1) manufacturer's identity, (2) heat number, and (3) grade designation. All tubing was manufactured by the same manufacturer; all samples were from a stainless steel Type 304 heat No. 84432.

Reboiler No. 1 Tubing

One half of longitudinally parted tube, approximately 50 ft long from O.D. of tube bundle was cut into sections about 42 in. long before shipment to ANL; all pieces were identified for sequence in reconstitution of tube. Tube had no previous history of leakage at ANL or of repair procedures prior to its removal, but it was located immediately adjacent to a tube which had split.

Micrometer measurements at 12-in. length intervals revealed that the metal wall thickness had been reduced, on the average, by 0.003 in., and was below the 0.035 minimum wall required to meet the ASME Code for Unfired Pressure Vessels, Section VIII, requirements. A dye-check examination of the bore side of the tube failed to reveal macroscopic defects. Alternating zones of roughened and smooth regions along length of tube and on I.D. of tube were found. The return bend section of the tube was smooth.

Reboiler No. 2 Tubing

The general appearance of the four tubes, from the upper portion of the tube bundle, returned for examination was far poorer than the tubing from the No. 1 Reboiler. This condition was demonstrated by rapping one tube against the others which revealed that the metal was "dead" acoustically, i.e., there was no metallic "ring." These samples were the ends of four different tubes immediately adjacent to the steel tube sheet.

Virgin Metal Tubing

All virgin metal tubing samples were heat treated to simulate the heat treatment received by reboiler tubing during manufacture.

Heat Treatment No. 1 - 1/2 hour @ 900°F required to anneal cold bent U-bends

Heat Treatment No. 2 - 17-1/2 hour @ 900°F required for stress relief of final reboiler carbon steel welds.

MILL TEST REPORT PROPERTIES

A213 seamless tubing 3/4 in. O.D. x 20 Ga. wall, Type 304 Stainless Steel

Tensile Strength - 92,100 psi
Yield Strength (0.2% Offset)
- 51,600 psi

Elongation - 53%
Hardness - 85 R_B

Chemical composition:

C - 0.063; Mn - 1.31; P - 0.021; S - 0.005; Si - 0.64;
Cr - 18.12; Ni - 9.54

HARDNESS (ROCKWELL B SCALE)

As received - 78.0

Heat Treated 1/2-hour @ 900°F - 76.8

Heat Treated 17-1/2 hour @ 900°F - 77.8

SST mandrel core - 103.5 (used to protect tubing against collapse)

MAGNETIC PROPERTIES

Permeability too low to be rolled on flat surface under influence of field of hand-held Alnico $\bar{V}$ magnet.

Corrosion Resistance in Nitric Acid

Determinations of corrosion resistance in two nitric acid concentrations were made in glass equipment arranged to meet the requirements of the ASTM specification A 262-55T. The procedure requires condenser equipment to maintain the acid test concentration.

All corrosion test samples were polished with succeeding finer grades of emery cloth to remove foreign surface materials, degreased in acetone, washed in demineralized water, and stored in a dessicator. Losses were first determined, by weighing on a chemical balance, and then converted to the more conventional corrosion rates expressed, inches penetration per month.

Analytical reagent grade nitric acid was diluted with demineralized water to 65% HNO₃ strength, established by titration, for the Huey test (boiling 65% HNO₃). The same acid grade was diluted to 20 w/o with demineralized water in the measurement of nitric acid attack during the acid cleaning procedure (i.e., a two-hour exposure to 20 w/o HNO₃ at 180°F had been specified).

All samples tested, with the exception of virgin metal, had been previously exposed to a 20% HNO₃ solution at 180°F for two hours at the fabricator's shop at least once. The resistance of thirteen samples to nitric acid environments is summarized in Table C-III. The appearance of the samples following exposure is illustrated by Figs. C-3, 4, and 5.

Virgin Metal Tubing

Neither the appearance of the samples, after the short and the long Huey tests, nor metal losses showed cause for alarm. Metal loss rates were acceptable. The appearance of the tubing samples is displayed in the first column of Figs. C-3 and 4.

Table C-III
CORROSION RATES IN NITRIC ACID TEST MEDIA
(Inches Penetration per Month)

Sample Type: Identifications (Figs. C-3, 4, and 5)	Heat Treatment (at ANL)	Rates in Media		Comments
		2 hr 20 w/o 180°F	5 x 48 hr 65% Boiling	
<u>Virgin Metal Tubing</u>				
(1.0)	As received	0.0001	0.00069	Relative Position along 50 Ft Length Unknown
(1.1)	As received plus 1/2 hr @ 900°F	0.00008	0.00071	Relative Position along 50 Ft Length Unknown
(1.2)	As received plus 17-1/2 hr @ 900°F	0.0005	0.00076	Relative Position Along 50 Ft Length Unknown
(3.0)	As received	N.D.*	0.00072	Middle of 50 Ft Tube Section
(3.1)	As received	N.D.	0.00071	Middle of 50 Ft Tube Section
(3.2)	As received	N.D.	0.00068	Middle of 50 Ft Tube Section
<u>Reboiler No. 1 Tubing</u>				
(2.0)	As received	0.0006	0.1169†	Top of U-Bend Rolled into Tube Sheet
(2.1)	As received	0.0004	0.1612†	Top of U-Bend 37 in. from Rolled End
(2.3)	As received	0.00008	0.0012	Bottom of U-Bend Rolled into Tube Sheet
(2.4)	As received	0.0006	0.0010	Bottom of U-Bend 36 in. from Rolled End
<u>Reboiler No. 2 Tubing</u>				
(4-1)	As received	N.D.	0.179‡	Top-Upper Half of Bundle
(4-2)	As received	N.D.	0.265‡	Middle-Upper Half of Bundle
(4-3)	As received	N.D.	0.210#	Bottom-Upper Half of Bundle

* Not determined.

† Discontinued after 48 hr.

‡ Discontinued after 25 hr.

Involuntary termination by expulsion of acid - 15 hr Estimated Exposure Period.

Fig. C-3. EBWR Primary Reboiler Tubing (Nitric Acid Tests on 3/4 in.--Type 304 Stainless Tubing)

VIRGIN METAL HUEY TEST (240 hr TOTAL)	REMOVED FROM NO. 1 REBOILER 2 hr @ 180°F - 20% HNO ₃ (I.D. OF TUBE)	REMOVED FROM NO. 1 REBOILER HUEY TEST (240 hr TOTAL) (I.D. OF TUBE)
(1.0) AS RECEIVED	(2.0) TOP TUBE - ROLLED END	(2.0) TOP TUBE - ROLLED END
(1.1) 1/2 hr AT 900°F	(2.1) TOP TUBE 37 in. FROM ROLLED END	(2.1) TOP TUBE 37 in. FROM ROLLED END
(1.2) 17 1/2 hr AT 900°F	(2.3) BOTTOM TUBE - ROLLED END	(2.3) BOTTOM TUBE - ROLLED END
	(2.4) BOTTOM TUBE 36 in. FROM ROLLED END	(2.4) BOTTOM TUBE 36 in. FROM ROLLED END

Fig. C-4. EBWR Primary Reboiler Tubing (Nitric Acid Tests on 3/4 in.--Type 304 Stainless Tubing)

(4.1) TOP OF BUNDLE

(4.2) MIDDLE OF BUNDLE

(4.3) BOTTOM OF BUNDLE

Fig. C-5. Reboiler Tubing (About 1 ft from Tube Sheet)

Reboiler No. 1 Tubing

Metal losses in the 2 hr - 20 w/o HNO_3 - 180°F corrosive environment, with one exception, were five times as great as that of virgin metal tubing. After exposure, the metal was noticeably darkened, i.e., the corrosive film was thicker. However, such comparisons of surface effects can be highly misleading when surfaces are roughened - as was the case here. The middle column of Figs. C-3 and 4 shows the metallic appearance of the inside and outside of the test samples.

Metal losses in the Huey test were prohibitively large for samples removed from the top portion of the U-bend. From the rate of attack at the end of the first exposure period, it is apparent that little, if any, of the 0.035-in. tube wall would have been left at the end of 240 hr. There was a substantial difference in the rate of attack on the top tube samples which were initially separated from each other by only 37 in. in the reboiler. The metal furthest away from the weld zone was attacked most. The third column in Figs. C-3 and 4 shows that both the inside and outside of the top tube sections were severely attacked.

The differences in the corrosion resistance between the metal samples separated approximately 40 ft apart in the reboiler are shown in the third column of Figs. C-3 and 4. Comparison of the corrosion rate for the metal in the bottom portion of the U-tube with the virgin metal tube shows that the U-tube metal was attacked 1-1/2 times faster than virgin metal tubing. Arithmetic proportioning shows that the metal in the top section of the tube was less corrosion resistant than virgin metal tubing by a factor of approximately 1000. The metal in the bottom tube section was adequately corrosion resistant for service in HNO_3 as well as water.

These quantitative nitric acid corrosion tests established the following:

1. That reboiler manufacturing procedures contributed substantially to the degradation of the tubing. (The leak-rate histories of the two reboilers during shop repair were very different. One unit was acknowledged as a total loss very early in the repair attempts by the fabricator.)
2. That nitric acid free of noxious agents would not attack tubing catastrophically in the 2 hr - 20 w/o HNO_3 - 180°F cleaning operation. Tubing which was destroyed in the Huey test series resisted the milder corrosive media without the excessive metal losses sustained by the tubing in the shop acid cleaning operation.

Metallographic Studies of Tubing

Concurrent with the execution of the short and the long Huey corrosion tests on tubing samples, tubing specimens, showing the metallic structure in the longitudinal and transverse sections, were mounted for metallographic examinations. In addition to metallographic examinations of as-received materials (virgin metal, and tubing from Reboiler Units No. 1 and No. 2), tubing from Reboiler Unit No. 1 was also examined following the second re-exposure to the 2 hr - 20 wt-% HNO_3 - 180°F at ANL. In all, 27 photomicrographs were prepared. Table C-IV identifies and summarizes the metallographic findings. All of the photomicrographic plates are reproduced here as Figs. C-6 through C-17.

Table C-IV

IDENTITY AND SUMMARY OF METALLOGRAPHIC FINDINGS ON REBOILER TUBING

Sample Type and Section		Heat Treatment (at ANL)	Micro No. and Magnification		Fig. No. C-	Significant Findings
<u>Virgin Metal</u>						
Unknown Position in 50 Ft. Tube Length	L	As received	30505	100X	6	{ Flow lines-stressed metal-spongy appearance caused by preferential etching of stressed metal. Large variation of grain size.
	T	As received	30506	100X		
	T	As received	30507	250X		
	L	1/2 hr 900° F	30508	100X	7	{ Same as No. 30505 except fewer flow lines. Lesser attack of grain structure seen in No. 30506 @ 100X.
	T	1/2 hr 900° F	30509	100X		
	L	17-1/2 hr 900° F	30510	100X		
	T	17-1/2 hr 900° F	30511	100X	8	{ Almost total elimination of strain lines. Refinement of entire wall metal structure.
<u>Virgin Metal</u> From Middle of Tube (on Tangent Section of U-Bend)						
T	As received	31455	200X	9	{ Typical structure for austenitic 18-8 steel.	
L	As received	31456	200X			
T	As received	31457	200X	10	{ Typical structure. I.D. Roughness. Inclusions.	
L	As received	31458	200X			
T	As received	31459	200X	11	{ Typical structure. Inclusions.	
L	As received	31460	200X			
<u>Reboiler No. 1 Tubing</u>						
Top of U-Tube Rolled End	L	As received	31070	200X	12a	{ Carbide networks around grains. Martensite within grains. Severe surface work effects. Confirmation of martensite
	L	As received	31453	1600X		
	T	As received	31071	200X	12b	{ Same as in No. 31070 @ 200X Confirmation of Martensite
	T	As received	31454	1600X		
Top of U-Tube-37 in. from A	L	As received	31072	200X	13	{ Light Carbides at grain boundaries. Intergranular corrosion General sensitization
	T	As received	31073	200X		

Table C-IV (Cont'd.)

Sample Type and Section		Heat Treatment (at ANL)	Micro No. and Magnification	Fig. No. C-	Significant Findings
Bottom of U-Tube-Rolled End	L	As received	31074 200X	14	Typical structure of 18-8 steels. Traces of martensite at I.D. surface
	T	As received	31075 200X		
Bottom of U-Tube-36 in. from C	L	As received	31205 200X	15	Typical 18-8 steel structure Typical 18-8 structure. Roughness at bore surface
	T	As received	31206 200X		
Top of U-Bend Rolled End	L	As received, polished and after 48 hr in boiling 65% HNO ₃	31694 100X	16	Carbide networks across tube wall. Severe inter-granular attack. Grain separation from tube walls Same as in No. 31694 sample except micro in transverse direction
Top of U-Bend 37 in. from A	L		31695 200X		
Reboiler No. 2 Tubing	L-T	As received	none	17	All Reboiler No. 2 tubing specimens examined. Two micros typical of all 4 tubes which show: (1) carbide networks (2) very severe inter-granular attack (3) grain expulsion from I.D. wall of tube (4) severe reduction of strength of tube by corrosive attack
"Smashed" Tube	L-T	As received	31879 200X		
	L-T		none		
	L-T	As received	31880 200X

100X

LONGITUDINAL SECTION

TRANSVERSE SECTION

250X

TRANSVERSE SECTION

Fig. C-6. Virgin Metal Tubing As Received

100X

LONGITUDINAL SECTION

TRANSVERSE SECTION

Fig. C-7. Virgin Metal Tubing (Heat Treatment 0.5 hr - 900°F)

100X

LONGITUDINAL SECTION

TRANSVERSE SECTION

Fig. C-8. Virgin Metal Tubing (Heat Treatment 17.5 hr - 900°F)

200X

LONGITUDINAL SECTION

TRANSVERSE SECTION

Fig. C-9. Virgin Metal Tubing (From Middle of Tube Length) As Received

200X

TRANSVERSE SECTION

LONGITUDINAL SECTION

C-10. Virgin Metal Tubing (From Middle of Tube Length) As Received

200X

TRANSVERSE SECTION

LONGITUDINAL SECTION

Fig. C-11. Virgin Metal Tubing (From Middle of Tube Length) As Received

200X

1600X

MAGNIFIED AREA OF TUBING TYPICAL OF STRUCTURE

Fig. C-12a. Reboiler No. 1 Tubing - Top of U-Tube - Rolled End - As Received Longitudinal Section

200X

1600X

MAGNIFIED AREA OF TUBING TYPICAL OF STRUCTURE

Fig. C-12b. Reboiler No. 1 Tubing - Top of U-Tube - Rolled End - As Received Transverse Section

200X

LONGITUDINAL SECTION

TRANSVERSE SECTION

Fig. C-13. Reboiler No. 1 Tubing - 37 in. from Rolled End - Top of U-Tube - As Received

200x

LONGITUDINAL SECTION

TRANSVERSE SECTION

Fig. C-14. Reboiler No. 1 Tubing - Bottom of U-Tube - Rolled End - As Received

TRANSVERSE SECTION

LONGITUDINAL SECTION

Fig. C-15. Reboiler No. 1 Tubing - 36 in. from Rolled End - Bottom of U-Tube - As Received

100X

ROLLED END

200X

37 in. FROM ROLLED END

Fig. C-16. Reboiler No. 1 Tubing - Top of U-Tube - after 48 hr Huey Test (Surfaces Previously Prepared by Manual Mechanical Polishing with Emery Cloth) Longitudinal Section

200X

"SMASHED TUBE"

3 TUBES OTHER THAN "SMASHED" TUBE

Fig. C-17. Reboiler No. 2 Tubing - As Received Typical Longitudinal and Transverse Structures

DIGEST OF METALLOGRAPHIC INVESTIGATIONS

Material Inhomogeneity

The photomicrographs revealed that the tubing material was non-homogeneous before and after its incorporation into the heat exchange U-tube bundles of the reboilers. Evidence of this was found in the examination of virgin metal tubing samples and of a complete tube removed from Reboiler No. 1. Specific revelations are as follows:

Virgin Metal Tubing

Stray samples of straight lengths of tubing, which most probably originated as excess length of tubing for the inner (and shorter) U-bends, revealed heavily strained metal, Fig. C-6. These ends were either improperly annealed, or, annealed tubing was later resized without subsequent solution annealing. The examination of tubing containing a U-bend, which had been rejected in manufacture for "wrinkling" on the inside (of a short radius U-tube) surface, showed that the mechanically unworked tangents of the U-bend were normal austenitic stainless steel structures, Figs. C-9, 10, and 11. The material in the U-bend was from the middle section of a 50 ft length of tubing.

Tubing heat treatments, required 900°F anneals for 1/2 hr and 17-1/2 hr respectively, were found to be beneficial for the tubing which displayed flow lines in the tubing structure, Figs. C-7 and 8. Unexpectedly, and surprisingly, the 17-1/2 hr anneal at 900°F very nearly restored the distorted structure of the tubing to that of completely solution-annealed material, Fig. C-8. None of the low temperature (900°F) anneals precipitated carbides in the grain boundaries, Figs. C-7 and 8.

Reboiler No. 1 Tubing

Four points along the length of a non-leaking reboiler tube, an outside U-bend which was removed, were examined to determine characteristic tubing structures. Two points, roughly three feet away from rolled and welded ends, were selected to separate out the effects of workmanship and welding heat. Since these two points were three feet away from the welding zone and geometrically behind the heavy steel tube sheet, welding heat could not influence the metallurgical structure. The tube was found endowed with two characteristic structures:

1. An austenitic structure outlined by a connected network of carbides in the upper portion of the U-tubes, micrograph Nos. 31070 and 31071 of Figs. C-12a and 12b, respectively, and Fig. C-13.
2. A normal structure for 18-8 austenitic stainless steel in the bottom portion of the U-tube, Figs. C-14 and 15.

The presence of precipitated carbides revealed that this end (upper portion) of the tube was held in the sensitizing range at some time in its total history. Since the bottom portion of the tube was characteristically austenitic, it is improbable that furnace heating, for stress relief of the reboiler steel welds, produced sensitization only of the upper portion of the U-bend.

The micrographs show additional metal structure differences in the rolled end sections of the same tube. These differences are ascribed to quality of workmanship. The bottom end of the U-tube shows only a trace of martensite, micrograph No. 31074, Fig. C-14, and very little evidence of severe cold rolling. The top end of the same U-tube shows a considerable conversion of austenite to martensite, Figs. C-12a and 12b, as well as evidence of severe cold deformation, i.e., note the deep notch rolled into the tube, micrograph No. 31070, Fig. C-12a.

Reboiler No. 2 Tubing

As revealed by the micrographs of the Reboiler No. 1 tubing, the tubing of the No. 2 unit shows the same connected carbide networks across the wall of the tube, Fig. C-17. Since these samples were removed from behind a heavy steel tube sheet also, there was no possibility of welding heat effects. As was the case in the No. 1 unit, these examples of poor quality tubing were also from the top portion of the U-bend.

Correlation of Corrosion Resistance by HNO_3 and Metallurgical Structure

Although metallurgical irregularities of the tubing structure were originally suspected and attributed to a sensitizing heat treatment, the later metallographic examinations, at ANL, of the tubing from the No. 1 unit did not verify this suspicion. The metallographic studies conducted on a tube removed from the reboiler, which had no prior history of leakage, showed that the upper half of the U-tube had been sensitized but the lower half of the same U-tube was fully austenitic. It is obviously impossible to sensitize only half of a U-tube in a tube bundle inside a thick steel shell by a furnace heat treatment. The metallographic examination actually pinpointed the original cause of tubing-failures, i.e., poor quality tubing furnished to the fabricator by the tube maker. The metallography also revealed pyramiding causes for failure by manufacturing procedures.

It is interesting to note how each of the investigational procedures supported the results of the other. Although both the corrosion tests and the metallography were quantitative individually, each by itself permits conjecture. When the results of both were combined, the uncertainty was eliminated. Both of the nitric acid corrosion tests failed to show the presence of strained metal in virgin metal tubing which was revealed metallographically. These nitric acid tests were, however, significant, i.e., tube metal which is fully austenitic and free of precipitated carbides was resistant to both the short 2 hr - 20 w/o HNO_3 - 180°F and the 240 hr boiling HNO_3 environments. Austenitic strained metal tubing is not acceptable because of the real danger of stress corrosion cracking in service.

Tubing removed from the No. 1 Reboiler unit showed a higher average corrosion rate in the short time 2 hr - 20 w/o HNO_3 - 180°F environment. In the Huey test series, the tubing from the unit was preferentially attacked; the top of the U-bend was virtually destroyed in a short time whereas the bottom of the same tube withstood this corrosive environment successfully for the full test period. The obvious conclusion was inadequate solution heat treatment of the tubing at the mill. First, however, heat of welding effects had to be eliminated. This was accomplished, as previously described, by the examination of tubing material far removed from the weld area. The metallography of the four tube specimens from the No. 1 unit revealed the differences, i.e., the presence and absence of precipitated carbides. The correlation between basic metal structure and corrosion resistance was exact: in the absence of carbide networks, the metal resisted corrosion; in the presence of carbides, the metal was attacked. The relationship between basic metallurgical structure and corrosion resistance was related in exactly the same way for tubing samples from the No. 2 Reboiler.

None of the laboratory nitric acid test environments resulted in the severity of attack revealed by the as-received tubing from Reboiler No. 1, Figs. C-16 and C-17. The differences in rate of attack between the Huey test and the fabricator's 2 hr - 20 w/o HNO_3 - 180°F cleaning treatment can be explained by the presence of contaminating agents in the nitric acid purchased. None of the cleaning acid employed was available for analysis. Verbal discussions over the quality and type of acid used elicited the information that technical grade acid was used.

Conclusions

From the study of the metallurgical structures and of the behavior of tubing samples in two nitric acid environments, it was concluded that three causes contributed to the failures of the tubing:

1. Poor quality tubing furnished to the fabricator of the reboilers.
2. Degradation of tubing by manufacturing procedures.
3. Use of nitric acid containing corrosion accelerating agents.

APPENDIX D

SUMMARY OF EBWR TEST PROCEDURES AND REPORTS
DURING THE PERIOD 1960 - 1962

Fig. D-1. Summary of Test Procedures and Reports (1960)

Fig. D-2. Summary of Test Procedures and Reports (1961)

Fig. D-3. Summary of Test Procedures and Reports (1962)

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