ANL-6302

ANL-6302

MASTER

325/1-29-62
Argonne National Laboratory

DESIGN AND HAZARDS SUMMARY

REPORT BOILING REACTOR

EXPERIMENT V (BORAX V)

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or*
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.*

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

*Price \$3.00 . Available from the Office of Technical Services,
Department of Commerce, Washington 25, D.C.*

ANL-6302
Reactor Technology
(TID-4500, 16th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

DESIGN AND HAZARDS SUMMARY REPORT
BOILING REACTOR EXPERIMENT V (BORAX V)

May 1961

Operated by The University of Chicago
under
Contract W-31-109-eng-38

CONTRIBUTORS

R . E . Rice	Editor, Core Design - Mechanical, Reactor Control, Fuel Handling, Experimental Program, Hazards Evaluation
E . J . Brooks	In-Core Instrumentation, Process Control and Instrumentation
D . H . Brown	Fuel Handling, Core Design - Mechanical
J . D . Cerchione	Core Design - Mechanical, Process Piping, Reactor Control
D . W . Cissel	Experimental Program, Process Piping
R . N . Curran	Nuclear Instrumentation
R . A . Cushman	Core Design - Heat and Hydraulics, Hazards Evaluation, Summary
J . I . Hagen	Core Design - Reactor Physics, Hazards Evaluation
F . S . Kirn	Biological Shielding
M. Novick	Management
R . R . Smith	Reactor Vessel, Biological Shielding
A . W . Solbrig	Core Design - Reactor Physics, Hazards Evaluation
F . W . Thalgott	Experimental Program, Hazards Evaluation
R . W . Thiel	Reactor Vessel, Process Piping
M . B . Trillhaase	Reactor Control, Buildings
W . R . Wallin	Reactor Control, Electrical Systems, Process Control and Instrumentation, Reactor Operation

TABLE OF CONTENTS

	<u>Page</u>
CONTRIBUTORS	2
TABLE OF CONTENTS	3
LIST OF FIGURES	7
LIST OF TABLES	10
 I. INTRODUCTION	 13
II. SUMMARY	14
III. DESCRIPTION OF BORAX V FACILITY	24
A. Reactor Facility	24
1. General Features of Reactor	24
2. Boiling Core Design - Mechanical	26
3. Superheating Core Design - Mechanical	33
4. Neutron Source	42
5. Experimental Components	43
6. Core Design - Heat and Hydraulics	43
7. Core Design - Reactor Physics	58
8. Reactor Vessel	73
9. Reactor Control and Instrumentation	82
10. In-core Instrumentation: Development	98
11. Fuel and Radioactive Material Handling	103
12. Biological Shielding	108
B. Process Piping Systems	113
1. Superheater-Vent, Flood and Drain Systems	113
2. Batch Feed System	115
3. Boron-addition System	117
4. Forced-convection System	118
5. Steam Systems	118
6. Air-ejector Exhaust System	122
7. Condensate and Feedwater System	122
8. Auxiliary Water System	124
9. Makeup-water System	125
10. Raw-water Systems	126
11. Other Systems	126
C. Process Instrumentation and Control Systems	128
1. General Safety Considerations	128
2. Superheater Startup and Shutdown System	132
3. Boron-addition System	133

TABLE OF CONTENTS

	<u>Page</u>
4. Forced-convection System	133
5. Steam-pressure Control System	133
6. Desuperheater and Turbine Steam Systems	134
7. Feedwater Control System	134
8. Reactor Preheat System	135
9. Makeup-water System	135
10. Reactor-water-demineralizer Pressure-control System	136
11. Process Temperature-indicating System	136
12. Superheater Fuel Element Temperatures	136
13. Reactor-vessel and Shield-wall Temperature- recording System	137
14. Miscellaneous Control Building Meters	137
15. Control-rod Seal-water Flowmeters and Valves	137
16. Local Pressure Gages	137
17. Local Temperature Indicators	138
18. Television Monitor	139
19. Instrument Compressed-air Systems	139
D. Buildings and Electrical Service Systems	139
1. Reactor Building	139
2. Control Building	141
3. Turbine Building and Cooling Tower	141
4. Field Chemistry Laboratory	142
5. Electrical Service Systems	142
6. Communications and Alarms	146
IV. MANAGEMENT	147
V. REACTOR OPERATIONS	149
A. Loading	149
B. Open-vessel Operation	149
C. At-pressure Operation	150
1. Natural-convection Boiling with Superheater	150
2. Natural-convection Boiling	152
3. Forced-convection Boiling	152
4. Forced-convection Boiling with Superheater	153
D. Water Purification System Operation	153

TABLE OF CONTENTS

	<u>Page</u>
VI. EXPERIMENTAL PROGRAM	154
A. Startup and Initial Operation.	154
B. Superheating Experiments	154
C. Boiling Core Experiments	155
1. Static Measurements.	155
2. Dynamic Measurements.	155
3. Engineering Tests	155
D. Water Chemistry Experiments	155
VII. HAZARDS EVALUATION.	157
A. Inherent Safety of Boiling Reactors.	157
B. Possible Hazards	158
1. Hazards of Superheater Operation.	158
2. Core Melting	161
3. Cold-water Accident	162
4. Power and Air Failure	164
5. Pump Failure	164
6. Instrument Failure.	165
7. Control Rod Failure.	165
8. Change in Power Demand.	167
9. Too Much or Too Little Water in Reactor.	168
10. Leaks from External Systems.	169
11. Fuel-cladding Failure	169
12. Startup Accident.	171
13. Excessive Pressure	171
14. Improper Charging of Fuel.	171
15. Earthquake	172
16. Fire.	172
17. Human Error or Maloperation	173
18. Metal-Water Reactions	173
19. 17-4 PH Stainless Steel Components	180
C. Maximum Accident	181
D. Site	190

TABLE OF CONTENTS

	<u>Page</u>
E. Estimation of Radiation Hazards to the Surrounding Population.	194
1. Vessel Rupture	194
2. Dosage Calculations	195
F. Hydrology, Seismology and Meteorology.	200
1. Hydrology	200
2. Seismology	201
3. Meteorology	202
ACKNOWLEDGEMENTS	205
BIBLIOGRAPHY.	206

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	BORAX V Facility	15
2.	Reactor with Central Superheater	17
3.	Boiling Fuel Assembly	27
4.	Boiling Fuel Rod	28
5.	Sections - Boiling Core Structure	29
6.	Elevation - Boiling Core Structure	31
7.	Natural Circulation Belleville Spring	32
8.	Superheater Fuel Assembly	34
9.	Mockup Superheater Fuel Assembly.	35
10.	Sections - Central Superheater Core Structure	36
11.	Elevation - Central Superheater Core Structure.	37
12.	Sections - Peripheral Superheater Core Structure	39
13.	Elevation - Peripheral Superheater Core Structure	40
14.	Representative Local-to-Average Axial Flux.	45
15.	Equivalent Diameter as a Function of Rod Pitch.	49
16.	Water Gap Flux Peaking	50
17.	Typical Hydrodynamic Parametric Study Results	51
18.	Superheater Axial Temperature Profile - 40 Mw	55
19.	Heat Loss to Moderator as a Function of Reactor Power	56
20.	Superheater Axial Temperature Profile - 1 Mw	57
21.	k_{eff} vs Boiler Loading for Core With Central Superheater . . .	60
22.	Percent Power in Central Superheater vs Boiling Fuel Rod Loadings.	60
23.	Loading for Core With 19% of Power Developed in Central Superheater.	61
24.	Plan of Thermal Flux - Central Superheater	63
25.	Plan of Thermal Flux - Peripheral Superheater.	64
26.	Vertical Section of Thermal Flux - Central Superheater	65
27.	Reactivity vs Temperature for the Boiling Core.	67

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
28.	Reactivity vs Temperature for the Core with Central Superheater.	67
29.	Reactivity vs Temperature for the Core with Peripheral Superheater.	67
30.	Central Superheater Flooding Worths.	70
31.	Superheater Loading Diagrams	70
32.	Radial Power Distribution for the Boiling Core	71
33.	Radial Power Distribution for the Core with Central Superheater.	72
34.	Radial Power Distribution for the Core with Peripheral Superheater.	72
35.	Reactor Pressure Vessel.	74
36.	Cruciform Control Rod	83
37.	Control-Rod-Drive Mechanism	85
38.	Carriage Nut Assembly	85
39.	Connector and Seal Assembly	86
40.	Control-Rod Scram Time.	87
41.	Recorder Panel and Control Console	88
42.	Nuclear Instrumentation	94
43.	Nuclear Instrumentation Operating Ranges	95
44.	Radiation Monitors	97
45.	Reactor Building Floor Plan and Elevations	109
46.	Radiation Heating of Shield Versus Distance into Concrete. . .	112
47.	Radiation Heating of Shield Along Surface of Reactor Pit Wall.	113
48.	Superheater Startup and Shutdown System	114
49.	Main Process and Instrumentation Systems.	116
50.	Control Building Floor Plan	141
51.	Electrical Single-line Diagram	143
52.	TREAT Transient Experiment with a Stainless Steel-clad, Oxide-core Pin (CEN-24).	175

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
53.	TREAT Transient Experiment with a Stainless Steel UO ₂ Cermet Sample (CEN-50).	176
54.	Sample of Stainless Steel-clad, Oxide-core Pin after Transient (CEN-24)	176
55.	Sample of Stainless Steel-UO ₂ Cermet after Transient (CEN-50)	177
56.	Doppler Shutdown Factor vs E*.	186
57.	Excursion Following Cold-water Accident ($\Delta T = 67^\circ K$) in Core with Central Superheater	187
58.	Site Plan.	191
59.	National Reactor Testing Station.	192
60.	Central Facilities 20-Foot-Level Wind Rose, 1950 through 1956.	204

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Design Characteristics of BORAX V	18-23
II.	Boiling Burnout Test Data	45
III.	Comparison of Boiling Fuel Rods	48
IV.	Operating Characteristics - Core With Central Superheater.	51-52
V.	Comparison of Superheater Fuel Elements	54
VI.	Comparison of Critical Calculations	61
VII.	Cold Reactivity Worths of Fuel Assemblies.	66
VIII.	Number Densities and Volume Fractions	68
IX.	Nuclear Constants	69
X.	Control-Rod Data and Values of Reactivity under Various Conditions.	71
XI.	Peak-to-average Values of Power and Thermal Flux . .	73
XII.	Reactor Vessel Design Criteria for BORAX V.	75-76
XIII.	Nozzle List for the Reactor Vessel	80
XIV.	Scram Conditions	89-90
XV.	Alarm Conditions	92-93
XVI.	Process-System Interlocks	129-132
XVII.	Equipment Connected to Emergency Power Supplies . . .	145
XVIII.	Location and Amount of Radioactivity in BORAX IV Fuel-cladding Rupture.	170
XIX.	Summary of In-pile Data on Stainless Steel-Water Reactions	175
XX.	Comparison of SS-Water Reaction Energy With Nuclear-excursion Energy.	179
XXI.	Cold-water Accidents	181
XXII.	Some Mechanical and Thermal Characteristics of BORAX V Cores.	184
XXIII.	Conversion Factors for Use in Excursion Calculations .	185
XXIV.	Some Results of Reactor Kinetics Calculations	188
XXV.	Populated Area Near BORAX V.	193

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
XXVI.	Heat Released to Core Water by Rupture of 50% of Boiling Fuel Rods.	195
XXVII.	Total Integrated Dose and Relative Ground Concentration Downwind for Unit Source	197
XXVIII.	External β Dose (rep) and γ Dose (Roentgens) from Airborne Fission Product Activity.	197
XXIX.	External Radiation Hazard Due to Ground Deposition . . .	198
XXX.	Fission Products Contributing to Ingestion Hazard.	199
XXXI.	Ingestion Dosage, Inversion Conditions.	200

DESIGN AND HAZARDS SUMMARY REPORT BOILING REACTOR EXPERIMENT V (BORAX V)

I. INTRODUCTION

In 1957 the idea for another boiling water reactor in the BORAX series was conceived by personnel of both the Idaho Division and Reactor Engineering Division of Argonne National Laboratory. In 1958 the Laboratory proposed that an extremely flexible boiling reactor experimental facility, BORAX V, be constructed and operated as a modification of, and addition to, the existing BORAX plant at the National Reactor Testing Station, Idaho.

The primary objectives of the proposed BORAX V program are to test various nuclear superheating concepts, and to advance the art of boiling water reactor design by performing experiments which will improve the understanding of factors limiting the stability of boiling reactors at high power densities.

A Preliminary Design and Hazards Report, BORAX V (ANL-6120) was issued in February, 1960. Since that time the reference designs of both the boiling and superheater fuel have been changed, and new, more extensive heat engineering, reactor physics and hazards evaluation studies have been carried out. The remainder of the reactor and plant design is essentially unchanged.

Consideration of the hazards associated with the operation of BORAX V has been based on the following conditions: For normal steady-state power and experimental operation, the reactor and plant are adequately shielded and ventilated to allow personnel to be safely stationed in the turbine building and on the main floor of the reactor building. The control building is located one-half mile distant from the reactor building. For special experiments which are deemed hazardous, such as open tank operation at power, defected fuel element tests, or excursions, personnel are withdrawn from the reactor area. The isolated location of BORAX V at the National Reactor Testing Station allows the initial testing of concepts, such as of nuclear superheat and of very high power densities, which might not be considered feasible in a populated area.

II. SUMMARY

Some of the plant utilized in the previous BORAX experiments is incorporated in the BORAX V system, but the reactor and its building, as well as the control building, located $\frac{1}{2}$ mile from the reactor, are new. A cutaway view of the BORAX V facility is shown in Fig. 1.

The reactor vessel is a cylinder with ellipsoidal heads, made of carbon steel clad internally with stainless steel. The inside diameter is $5\frac{1}{2}$ ft and the internal height is 16 ft. The core is centered about 4 ft from the bottom. The top portion of the vessel is used as a steam dome. Control rods are driven from below the vessel.

BORAX V has been designed as an extremely flexible system. At present, there are plans to provide three separate core configurations: a boiling core with a centrally located superheater, a boiling core with a peripherally located superheater, and a boiling core without superheater. In all cases it will be possible to operate with either natural or forced circulation of water through the core. The water serves as the moderator in both the boiling and superheater regions of the core and as coolant in the boiling region. The superheater is cooled by steam.

Each of the three cores is 24 in. high and has an effective diameter of 39 in. when containing the maximum of 60 four-inch-square fuel assemblies arranged in an 8 x 8 array with the corner assemblies missing. The central superheater contains 12 fuel assemblies in the central position, whereas the peripheral superheater contains 16 fuel assemblies, 4 in the middle of each outer row. Each square cell of 4 assemblies is surrounded by adjacent control-rod channels. Experiments will be run with the full complement of 60 fuel assemblies as well as with reduced numbers. The BORAX V reactor with a central superheater is shown in Fig. 2.

The boiling fuel assemblies of the reference design are composed of individually removable fuel rods (the number of rods per assembly may vary from 0 to 49) made from UO_2 of 5% enrichment with 0.015-in. Type 304 stainless steel cladding. The OD of the rods is 0.375 in. Two different enrichments are available, nominal and twice nominal.

The superheater fuel assembly of the reference design is made up of 5 plate-type elements with intervening water gaps. Each element is contained in a stainless steel tube which maintains a static steam-filled insulating gap between the element and the moderator water outside the channel. Each element contains 4 fuel plates separated by three 0.062-in. coolant channels. The fuel plates contain a 0.014-in. thick, highly enriched UO_2 -stainless steel cermet meat within a 0.008-in. stainless steel cladding. The two outside plates in each element contain nominally half as much UO_2 as the two inside plates, since they are cooled only on one surface.

Fig. 1

BORAX-V Facility

Water flows up through the boiling core and the riser above the core by either forced or natural circulation. The steam formed in the core continues up into the steam space while the water flows down through the annular downcomer area between the core and the reactor vessel. Feedwater introduced at the top of the downcomer subcools the recirculating water and helps to increase the natural-circulation rate. For the boiling core case, the steam leaves the pressure vessel through a 6-in. steam line. In the superheating cores the steam from the steam dome enters the inlets to the superheater fuel assemblies, makes one pass down through half the assemblies, mixes in the lower plenums, and makes a second pass up through the remainder of the assemblies. It then leaves the pressure vessel through flexible tubing manifolds and the superheated-steam outlet nozzles. Because BORAX V is using the turbine-generator-condenser system from the preceding BORAX plants, the 600-psig steam must be reduced in pressure and temperature before entering the turbine.

Nine control rods, driven from below, are made of Boral canned in stainless steel. The drive mechanisms are those formerly used on EBWR, modified for use in BORAX V so that the maximum rate of addition of reactivity is limited to 0.05%/sec. Any one of 15 scram signals causes the rods to be completely inserted within 0.25 sec by the pressure within the reactor vessel, augmented by springs and gravity. An additional method of reactivity control is provided by a boron-addition system which introduces a boric acid solution into the reactor vessel as a reserve shutdown mechanism.

BORAX V possesses the inherent shutdown mechanism common to all boiling water reactors. The additional voids formed in the moderator in the event of an excursion cause a decrease in power, though the actual void coefficient of reactivity depends on the core configuration employed.

The greatest departure in design from the previous BORAX reactors is found in the superheater. In effect, this introduces a gas-cooled zone into a normal boiling core and presents problems common to such reactors, e.g., decay-heat removal and effect of flooding the gas passages. The problem of decay-heat removal is solved by designing the superheater fuel element so it can be radiation-cooled a short time after the reactor shuts down. In fact, the reactor can operate at 3.5 Mw with no steam flow, yet the fuel elements do not melt. An alternative system of shutdown cooling by flooding the superheater is also provided. Numerous safety interlocks are provided to prevent accidental flooding of the superheater during operation.

The maximum accident is postulated as a cold-water accident which would occur if the forced-convection pump starts up while the reactor with central superheater is just critical at the operating temperature and the water in the forced-convection piping system is at room temperature. In such a case, cold water would enter the core and cause a reactivity increase

Fig. 2

Reactor with Central Superheater

of 6.7% in 0.4 sec, or a rate of increase of 16.7%/sec. This rate can be compared to the 20%/sec reactivity insertion which caused the destruction of BORAX I. Under these conditions it is possible that the reactor would be destroyed. Elaborate safety systems insure that such an accident does not occur.

In the event a radioactive cloud is released after extended 20-Mw operation, the 300-r exclusion radius is less than one-half mile (the distance to the control building) except under extremely pessimistic conditions of inversion with low wind speed.

Table I summarizes the pertinent design characteristics of BORAX V.

Table I

DESIGN CHARACTERISTICS OF BORAX V

A. Reactor Description

1. Cores

Types: Central Superheater, Peripheral Superheater, Boiling

Geometry: (Flexible) Right Pseudocylinders; Square, Rectangular, and Octagonal Prisms, etc.

Material: Core Structure and Control Rod Shrouds	Alcoa X-8001
	Al-Ni Alloy
Moderator and Reflector	H ₂ O
Coolant	H ₂ O and Steam
Active Core Height, in.	24
Maximum Equivalent Cylindrical Diameter, in.	39
Maximum Number of Fuel Assemblies	60

Fuel Loading (60 Assemblies/Core), kg of U²³⁵

Core Type	Boiler	Superheater	Total
Boiling	47.4	-	47.4
Central Superheater	38.1	5.2	43.3
Peripheral Superheater	34.7	10.9	45.6

Table I (Cont'd.)

Core Composition (60 Assemblies/Core), Vol. % at 20 Mw

Function	Material	Boiler	Central Superheater	Peripheral Superheater
Moderation	H ₂ O	46.8	40.8	40.8
Fuel	U ²³⁵ O ₂	1.14	0.57	0.89
Structure	Al-X 8001	22.2	17.5	17.5
	St. Steel	4.2	17.4	17.1
Diluents	Void	3.9	23.7	23.7
	U ²³⁸ O ₂			
a. Boiling Core		21.6	-	-
b. Central Superheater Core		21.7	0.04	-
c. Peripheral Superheater Core		21.6	-	0.07
Poison	Boron- Stainless Steel Rods			
a. Boiling Core		0.12	-	-
b. Peripheral Superheater Core		0.13	-	-
2. <u>Boiling Fuel Assembly</u>				
Dimensions: Square X-8001 Al Tube, in.			3 ⁷ / ₈ by 3 ⁷ / ₈ OD by ¹ / ₁₆ w	
Number of Fuel Rods			49	
Rod Spacing, Square Pitch, in.			0.5	
3. <u>Boiling Fuel Rod</u>				
Fuel Material			5% enriched UO ₂	
UO ₂ Rod Diameter, in.			0.343	
Cladding Material			Type 304 SS	
Cladding Thickness, Nominal, in.			0.015	
Fuel Rod, OD, Nominal, in.			0.375	

Table I (Cont'd.)

4. <u>Superheater Fuel Assembly</u>	
Dimensions, Nominal, in.	3.875 x 3.633
Number of Elements per Assembly	5
Number of Plates per Element	4
Fully Loaded Plates	2
Half-loaded Plates	2
Meat Dimension, in.	24 x 3.495 x 0.014
Meat Composition	Cermet, highly enriched UO ₂ in Type 304 SS matrix
<u>Plate Type</u>	<u>UO₂ Content (w/o of meat)</u>
Fully Loaded Central (FCE)	20.8
Half Loaded Central (HCE)	11.4
Fully Loaded Peripheral (FPE)	31.8
Half Loaded Peripheral (HPE)	17.7
Cladding Material	Type 304L SS
Cladding Thickness, in.	0.008
Distance Between Plates, in.	0.062
5. <u>Heat Transfer and Flow Characteristics</u>	
Reactor Reference Design Power	
Core with Central Superheater, kw	20,000
Boiling Region, kw	16,600
Superheater, kw	3,400
Steam Flow, lb/hr	49,800
Steam Conditions Leaving Reactor,	
Superheated, °F	850
psig	587
Average Power Density in Boiler (Based on Region Vol.), kw/l	
	95
Average Power Density in Superheater (Based on Region Vol.), kw/l	
	80
Power Ratios, Peak to Average:	
Axial - Boiler	1.33
Radial - Boiler	1.45
Local - Boiler	1.3
Axial - Superheater	1.3
Radial - Superheater	1.2

Table I (Cont'd.)

Engineering Hot-Channel Factors, Superheater		
Factor for Enthalpy Rise		1.15
Factor for Film-temperature Drop		1.36
Heat Flux, Btu/(hr)(ft ²):		
Boiler, max		310,000
av		123,000
Superheater, max		100,000
av		55,000
Temp, Superheater Fuel Element		
Center, (max) °F		1,125
Temp, Superheater Fuel Element		
Surface, (max) °F		1,110
Temp, Boiling Fuel Rod		
Centerline, (max) °F		3,100
Temp, Boiling Fuel Rod		
Surface, (max) °F		515
Pressure Drop through Superheater Core, psi:		
1st Pass		3
2nd Pass		4
6. <u>Control Rods</u>		
Cruciform Blades, in.		14 x 14 x 0.312 thick
"T" Blades, in.		14 x 7 x 0.312 thick
Poison Material		Boral (50 w/o B ₄ C)
Poison Length, in.		24
Stroke, in.:		
Boiling Core		29
Superheating Cores		24
Material:		
Cladding		Type 304 SS
Follower		X-8001 Al
Control Rod Drive Location		
		Bottom of Reactor Vessel

Table I (Cont'd.)

Number of Control Rods:		
Cruciform		5
"T"-shaped		4
Shutdown Reactivity, Central Superheater Core, (Cold, Flooded Superheater, All Rods In), $\% \Delta k/k$		-18.0
7. <u>Control Rod Drives</u>		
Type: EBWR with Linear Seal, Lead Screw, and Magnetic Latch		
Scram Time, Effective Stroke, sec		0.25
Max Rate of Reactivity Insertion, $\%/sec$		0.05
8. <u>Reactor Vessel</u>		
Diameter, Inside, in.		66
Height, Inside, ft.		16
Total Wall Thickness, in.		$2\frac{1}{16}$
Cladding Thickness, in.		$\frac{3}{16}$
Vessel Material		ASTM-A-212-B, Fine Grain Steel
Cladding Material		Type 304 SS
Operating Pressure, psig		600
Design Pressure, psig		700
Design Temperature, °F		650
Neutron Shield Material (Top Head Only)		1% Boron-SS
Neutron Shield Thickness, in.		1
B. <u>Performance</u>		
1. Reactor		
Nominal Full Power, Thermal Mw		20
Maximum Plant Capacity, Thermal Mw		40
Average Thermal Neutron Flux at 20 Mw, $n/(cm^2)(sec)$		2×10^{13}
Operating Pressure Range, psig		0-600

Table I (Cont'd.)

<u>2. Steam System</u>		
Operating Pressure Range, psig		100-600
Max Steam Temp, Saturated, °F		489
Max Steam Temp, Superheated, °F		850
Turbogenerator System Capacity at 350 psi and 4 in. Hg, Absolute		
Input Power, Thermal, Mw		20
Steam Flow, Saturated, lb/hr		60,000
Atmospheric Vent Capacity		
Power, Thermal, Mw		20
SteamFlow, lb/hr		60,000
Turbogenerator-Condenser Plus Atmospheric Vent Capacity		
Power, Thermal, Mw		40
Steam Flow, lb/hr		120,000
Electrical Generating Capacity, Mw		3.5
<u>3. Feedwater System</u>		
Number of Feed Pumps		2
Pump Capacity, Each, gpm		150
Pump Head, psig		700
Maximum Feedwater Flow, gpm		300
Feedwater Temperature, Nominal, °F		100
<u>4. Forced-convection System</u>		
Operating Pressure, psig		600
Water Temperature, °F		489
Pump Capacity at 14 ft. of Water NPSH, gpm		10,000
Pump Head, ft. of Water		193
Material		Type 304 SS
<u>5. Reactor Water Demineralizer</u>		
Maximum Temperature at Demineralizer Inlet, °F		120
Maximum System Pressure, psig		60
Maximum Flow Rate, gpm		20
<u>6. Makeup Water Demineralizer System</u>		
Capacity, gpm		70

III. DESCRIPTION OF BORAX V FACILITY

A. Reactor Facility

1. General Features of Reactor

The BORAX V reactor system is designed to obtain an extremely flexible experimental facility at a minimum cost.

An example of this experimental diversity is the design of the reactor vessel. In order that the top of the vessel be free for easy access and for changing of experiments and fuel, the nine control rod drives are mounted on nozzles in the bottom head. The four forced-convection inlet nozzles in the bottom head and two outlet nozzles in the vessel shell are so located that any core can be tested with either natural or forced convection by removing or installing the portable forced-convection baffle inside the vessel. A spare water-preheat line, the main saturated-steam outlet, four superheated steam outlets also used for removal of in-pile loop steam, two instrumentation outlets, water-level taps, poison-injection nozzle, and feedwater inlet are also located in the vessel shell. The vessel top head has five nozzles for fuel-handling operations with boiling fuel assemblies, three instrumentation nozzles, and three oscillator-rod-drive nozzles.

Removable feedwater sparger rings are located either at the top of the boiling core shroud or at the top of chimneys 5 ft above the core. Feedwater may also be injected into the forced-convection system.

The integral nuclear superheating, rather than the two-reactor, approach, has been selected for testing in BORAX V. For small reactors the integral superheater seems to show the most economic promise. However, for possible future tests of the nuclear superheating concept which embodies two cores within a single reactor vessel, a second set of instrumentation holes and cooling coils in the reactor shield has been provided adjacent to the top of the reactor vessel. In this concept a conventional boiling core controlled by drives through the bottom of the reactor vessel furnishes saturated steam to a superheating core in the upper part of the vessel, which is, in turn, controlled by drives mounted on the top head.

Three separate core structures are provided: one for a pure boiling core, one for a boiling core with an integral central superheater zone, and one for a boiling core with an integral peripheral superheater zone. These core structures are readily replaceable after removal of fuel and control rods. No bolting is required to hold a core structure in the reactor vessel. The structure rests on pads on the bottom head of the reactor vessel and is held down by gussets inside the removable top head of

the vessel. By means of the building crane, the core structures are withdrawn from the vessel as a unit and stored in the dry-storage pit. The crane has a remote control station which can be shielded during these operations.

Boiling fuel assemblies are interchangeable between all three core structures. In the boiling core, the number of fuel assemblies may be varied to change core diameter. For high-velocity, forced-convection tests a minimum number of assemblies may be used. The number of fuel rods per boiling fuel assembly may be varied and boiling fuel rods of two different enrichments are available. Longer cores could be tested by inserting longer fuel assemblies and control rods. A stroke of 48 in. is available in the control rod drives.

Neither time nor money was available in the BORAX V program for extensive fuel-element development. Therefore, the basic fuel elements for both the boiler and superheater are of conventional design although the fuel assemblies are unique.

Provisions in the steam system allow handling the exit steam from two in-pile loops, which may be used for testing alternate or advanced superheater fuel assemblies with independent cooling systems. The in-pile loops may be used with either superheating or boiling cores.

Each of the boiling and superheat cores has five cruciform and four "T"-shaped control rods, made of Boral and having an aluminum follower. Partially independent control of superheater power is achieved with the control rods located in the superheater.

Four major reasons governed the choice of the 600-psig operating pressure:

1. Operation at 600 psig is considered adequate for all the experiments anticipated;
2. With equipment designed for higher pressures, experimental flexibility is inhibited by the increased difficulty of rapid fuel and experimental changes;
3. At pressures higher than 600 psig, the costs of pressure vessels, piping, and other hardware increase rapidly;
4. Six-hundred-psig saturated-water conditions are considered near the upper limit of use for X-8001 aluminum, which is proposed for some core components.

2. Boiling Core Design - Mechanical

a. Boiling Fuel Assembly

The boiling fuel assembly, shown in Fig. 3, permits a variable loading of from 0 to 49 removable rods. The boiling fuel assembly box is made up of a bottom cylindrical-to-square-transition fitting and a 3.875-in. -square tube, made of Al X-8001. Inside the aluminum tube are riveted an upper grid, 4 in. deep, and a lower grid, $\frac{3}{4}$ in. deep, of Type 304 stainless steel. A fuel rod inserted through a hole in the upper grid is guided through the corresponding hole in the lower grid. The fuel rod is then pushed down to compress the rod spring and turned 90° to latch it in place and hold the rod in tension between the two grids.

A replaceable, experimental orifice plate is provided in the lower end-fitting of the boiling fuel assembly. Holes are located in the upper end of the square tube above the fuel rods to receive the boiling-fuel-assembly grapples. Burnable poison may be installed by means of replaceable boron-stainless steel rods.

b. Boiling Fuel Rod

The boiling fuel rod, illustrated in Fig. 4, contains partially enriched UO_2 , of about 91% theoretical density, in the form of cylindrical pellets of 0.343-in. diameter, for a length of 24 in. The UO_2 is clad by a tube of Type 304 stainless steel with a 0.015-in. wall thickness. The rods have a fission gas expansion space with an internal Inconel X spring to allow for pellet expansion and are helium-filled. The maximum radial clearance between pellet and clad is 0.002 in. The stainless steel end-fittings are welded on in a helium atmosphere and the lower end-fitting contains a latching pin. On the upper end-fitting is the compression spring and washer, and a special head for receiving the fuel rod manipulator.

c. Core Structure - Boiling

Figure 5 shows a plan view of the boiling core structure, which provides for a maximum of 60 fuel assemblies, grouped in cells of 4 or 3 assemblies by the control-rod-guide shroud. The shroud is made of $\frac{5}{32}$ in.-thick, X-8001 aluminum, 63 in. high, and is joined by X-8001 aluminum vertical stanchions to form $\frac{1}{2}$ in.-wide channels for five 14 x 14-in. cruciform control rods and four 14 x 7-in. "T"-shaped control rods. Tubes are attached to the outside of the shroud to hold the antimony-beryllium neutron source, a rotating oscillating rod, and loading counters. The shroud is bolted to, and the fuel assemblies rest on, a 4- $\frac{1}{4}$ -in.-thick core support plate of Type 304 stainless steel.

Fig. 3

Boiling Fuel Assembly

Fig. 4
Boiling Fuel Rod

NOTE: SEE FIG. 6 FOR ELEVATION

Fig. 5
Sections - Boiling Core Structure

Figure 6 shows an elevation of the boiling core structure. The core-support plate has 8 stainless steel, tubular legs which are guided by tapered dowels and supported by pads on the bottom head of the reactor vessel. To these legs is fastened a diffuser ring which distributes the water from the 4 inlet nozzles under forced-convection operation. The core support plate and aluminum shroud structure is held down by 8 stainless steel, tubular struts terminating at the top in a stainless steel ring flange. The ring flange in turn is held down through a $54\frac{3}{4}$ -in. diameter Belleville spring by 6 gussets welded to the upper head of the reactor vessel. The spring, made of 17-4 PH stainless steel, is heat treated for four hours at 1150°F to give a hardness of Rockwell C-33.

The Belleville spring provides for the differential expansion between core structure and reactor vessel. A drawing of the spring arrangement is shown in Fig. 7. The deflections which have been considered in the design of the Belleville spring are:

1. The elongation of the reactor vessel due to the internal pressure of 600 psi.
2. The elongation of the carbon steel vessel due to the increase in temperature from 70 to 500°F.
3. The shortening of the core structure due to the Belleville spring compressive load.
4. The elongation of the stainless steel core structure due to the change in temperature from 70 to 500°F.

For all three core structures when natural circulation is being used, a spring force of 20,000 lb has been selected to hold the structures firmly in place.

Each of 4 boiling fuel assemblies in a cell is held down and centered by a $7\frac{7}{8}$ -in.-square aluminum hold-down box, which in turn is clamped down by latches bolted to the top of the shroud. These latches may be operated from the top by long-handled socket wrenches. Additional riser height for natural circulation is provided by $7\frac{7}{8}$ -in.-square stainless steel chimneys which pilot into the hold-down boxes and are held in place by their own weight. The cells containing three assemblies have hold-down boxes and chimneys shaped to fit inside the cell.

NOTE: SEE FIG 5 FOR SECTION

Fig. 6

Elevation - Boiling Core Structure

Fig. 7

Natural Circulation Belleville Spring

For forced-convection tests a conical, stainless steel baffle is installed after removing the reactor vessel upper head, the core structure ring flange with Belleville spring, and the feedwater sparger. The forced-convection baffle seals against a machined ring in the lower head of the reactor vessel and against the core support plate by means of laminated piston rings. It can also seal, if necessary, against the rabbeted edge of the core support plate by means of a hollow metal O-ring gasket. The baffle is clamped in place by the core structure ring flange through 8 stainless steel, tubular struts and a ring flange of its own. When installing or removing the forced-convection baffle, the lower core structure remains in place. Also, for forced-convection tests, orifice plates are installed in the shroud at the top of each control-rod channel to regulate the amount of water bypassed for control-rod cooling and to reduce the hydraulic lifting force on the control rods. The maximum hydraulic force on the core support plate is 29,000 lb; across the forced-convection baffle is an additional 21,000 lb, making a total possible hydraulic load of about 50,000 lb. An additional gasket load of 10,000 lb is required to seat the

hollow metal "O"-ring seal, if used, between the forced-convection baffle and the core support plate. Thus, for forced-convection operation a second Belleville spring of 60,000 lb load is required.

A special, low-headroom lifting fixture which engages lifting pads on the top ring flange of the core structure permits handling of the core structure as a unit by means of the building crane.

3. Superheating Core Design - Mechanical

a. Superheater Fuel Assembly

The reference design for the superheater fuel assembly is shown in Figs. 8 and 9. The assembly consists of 5 flat-plate fuel elements, with light water moderator between and outside them, joined into a double-wall, static-steam-insulated lower nozzle and/or upper riser. On a second pass assembly the riser has a double wall to form a static-steam-insulating gap, while a first-pass assembly has a single tube riser. Near the top of each assembly is an 800-lb Inconel X hold-down spring and flange, and near the bottom is a seal flange.

The flat-plate fuel element is a rigid assembly of 4 fuel plates, 2 grooved side plates, and spacers brazed with Coast Metals-60 alloy. The fuel element is enclosed by a rectangular, Type 304 stainless steel tube which forms a static-steam-insulating annulus to reduce loss of heat from the superheated steam to the moderator. It is seal-welded to the insulating tube at the top only to allow downward expansion. A fuel plate is 0.030 in. thick and has a sandwich construction with a 0.014-in.-thick meat of highly enriched UO_2 dispersed in a stainless steel matrix and a 0.008-in.-thick cladding of Type 304L stainless steel. Spacing between plates is 0.062 in. The two outside fuel plates are loaded only about half as much as the inner three, because they are cooled on one side only.

The design features of the reference superheater fuel assembly are good radiation and conductive cooling under shutdown conditions, and a small flux depression across the fuel element. Fuel-plate spacing is maintained by 0.062-in. square wires which are placed to reduce deflections caused by thermal stresses, hydraulic loads, and the two-pass, pressure-drop load transmitted through the insulating tube. In order to reduce heat loss across the insulating gap, 0.030-in., scalloped, round-wire spacers are used and the edges of the side plates are also scalloped. Burnable poison strips could be spot-welded to the outside of the insulating tubes, if needed. The reference design does not have burnable poison.

Fig. 8

Superheater Fuel Assembly

Fig. 9

Mockup Superheater Fuel Assembly

b. Core Structure - Central Superheater

The core structure for the central superheater zone is shown in Figs. 2, 10, and 11. Provision is made for locating 12 superheater fuel assemblies in the center of the core surrounding the central control rod. The remainder of the fuel-assembly locations are filled with boiling fuel assemblies.

Fig. 10

Sections - Central Superheater Core Structure

NOTE: SEE FIG. 10 FOR SECTIONS

Fig. 11

Elevation - Central Superheater Core Structure

A superheating core structure has the same basic design as the structure for the boiling core. It has a similar, though shorter, X-8001 aluminum shroud with boiling fuel assembly hold-down latches, a stainless steel core-support plate, a similar leg and strut arrangement, and is also held down by means of the reactor vessel top-head and Belleville spring. However, the core-support plates for superheat cores have additional holes drilled in them around the superheater fuel assembly locations for circulation of moderator water through these assemblies. These holes are sized for natural circulation. When forced convection is used, orifice bushings, retained by the bottom sealing-flange on the superheater fuel assembly, are installed to prevent bypassing an excessive amount of water. The same removable forced-convection baffle may be used on any of the boiling or superheating core structures.

There are two parallel steam-flow patterns in each of which the saturated steam from the reactor vessel steam dome enters the top of the superheater fuel assembly. A demister-type steam dryer may be installed on the fuel assembly inlets, if required. During the first pass, steam flows down through three fuel assemblies, through the core-support plate into a double-walled, static-steam-insulated plenum chamber and a U-pipe arrangement which passes beneath the central control-rod follower. During the second pass, steam flows up through three fuel assemblies and risers, which are coupled with single-bolt couplings, into a manifold system made of flexible, stainless steel tubing and double-wall pipe, and then through the steam outlets located below the reactor vessel main flange.

Welded to the bottom of the two U-pipes beneath the core-support plate is a superheater drain and flood manifold system which leaves the reactor vessel via piping through one of the forced-convection system inlet nozzles. The 2-in. downcomer pipe for this system has a coupling which must be disconnected during a core structure-removal operation because of the limited crane lift height above the reactor building floor slab.

A guide grid bolted to the top ring-flange on the superheating core structure locates the upper end of the superheater fuel assemblies until the hold-down grid, resting on the superheater fuel-assembly hold-down springs, is placed over them. This plate is held down through the Belleville spring by the reactor vessel top head, which also compresses the superheater hold-down springs which apply the proper gasket load to the seal between the fuel assembly and the core-support plate. The hold-down spring and flexible hose in the exit-steam manifold permit differential expansion between the fuel assembly and the core structure.

c. Core Structure - Peripheral Superheater

The core structure for the peripheral superheater, as shown in Figs. 12 and 13, is similar in design to that of the central superheater.

Fig. 12

Sections - Peripheral Superheater Core Structure

SECTION B-B

NOTE: SEE FIG. 12 FOR SECTIONS

Fig. 13

Elevation - Peripheral Superheater Core Structure

The sixteen superheater fuel assemblies are located in the outer core edge, two on each side of the radial blade of a "T"-shaped control rod. The remainder of the fuel assembly locations are filled with boiling fuel assemblies. There are eight parallel steam-flow paths, each made up of a pair of adjacent superheater fuel assemblies with a double-walled connecting plenum below the core-support plate. Each of these eight plenums has welded to it a superheater drain and fill line leading to a manifold which is connected to a single outlet pipe from the reactor vessel. Flexible metal tubing and double-wall-pipe manifolds connect the eight second-pass fuel assembly risers to the four superheated steam nozzles in the reactor vessel shell.

The superheater fuel assemblies in the peripheral superheater are held in place in the same manner as in the central superheater.

d. Seals - Superheater Fuel Assembly

An efficient mechanical seal is needed to limit leakage of reactor water into the steam passages of the superheater fuel assemblies. This seal is located between the sealing flange on the superheater fuel assembly bottom nozzle and the core-support plate. A program was carried out to test the following standard, commercially available, metallic seals:

1. Metallic Static Seal manufactured by Cadillac Gage Co.;
2. Skinner Static Spring Seal, manufactured by the Hydrodyne Corporation; and
3. Hollow Metal O-Rings manufactured by the Advanced Products Co., the D. S. D. Manufacturing Co., and the United Aircraft Products Co.

The choice of seals was restricted to the mechanical type because the low compressive forces required allow use of reasonably sized superheater fuel assembly hold-down springs.

The seals were tested in a vessel at 600 psig and 489°F by means of a mocked-up superheater fuel assembly with a hold-down and alignment arrangement which simulated the reference design.

The silver-plated hollow stainless steel O-ring was selected as the most desirable seal for BORAX V, and testing of the other types was discontinued. This selection was based on the fact that the O-ring gave by far the best seal and was comparatively unaffected by the amount of pressure drop across the seal. In addition, the O-ring seals are considerably more economical.

For proper sealing and to prevent overstressing, the O-ring must be deflected a specified amount and must be backed up on either the ID or OD. The reference seal is a standard $2\frac{5}{8}$ -in. diameter, silver-plated, O-ring press fit inside its own integral retainer ring. This retainer ring satisfies both of the above requirements and eliminates the necessity of machining seal grooves into either the grid support plate or the fuel element flange, which could collect dirt and be difficult to clean. Several manufacturers supply the O-rings press fit inside a retainer ring.

Extensive testing on stainless steel, silver-plated O-rings, with pressure equalizing holes, press fit inside an integral retainer ring, was conducted in the seal-test vessel with excellent results. This type of seal is shown on a mockup superheat fuel assembly in Fig. 9. With a 20 to 30-psi pressure differential across the seal and a hold-down force of 600-800 lb, leakage rates averaged 1 to 4 cc/hr.

As a means of evaluation, these leakage rates may be compared with the following: If BORAX V is operating at 20 Mw with 16 superheater fuel assemblies in the core and a superheated steam exit temperature of 850°F, a leakage rate of 2.58 gal/hr-fuel assembly is required to drop the superheated steam temperature by 10°F.

In order to facilitate seal replacement and ease of insertion and removal in the reactor, three equally spaced stainless steel leaf springs are inserted on the superheater fuel assembly bottom nozzle to retain the O-ring seal. A special "sliding door" jig is used to allow remote removal and replacement of seals on fuel assembly nozzles in the water-storage pit.

4. Neutron Source

The Sb-Be neutron source for BORAX V is interchangeable between all three cores. The antimony is a rod, 0.750-in. diameter by 12-in. long, canned in stainless steel, and the beryllium is a hollow cylinder, 2.890-in. OD by 0.965-in. ID by 14 in. long, also canned in stainless steel. The antimony slug has a source strength of about 400 curies.

Provisions are made for handling each piece independently. The source is located in a source tube in the radial reflector immediately outside the core when the core is fully loaded. For initial buildup to critical mass or with a reduced size core, the source may be located in a dummy fuel assembly.

5. Experimental Components

The following core components are used for certain experiments in all three cores:

(1) Dummy fuel assemblies made of X-8001 aluminum and stainless steel are used to plug off unused fuel-assembly holes in the core support plate when testing small cores with forced convection. Special dummy fuel assemblies may contain the neutron source, oscillating rod, or loading counters.

(2) Void tubes, made of empty, watertight, $\frac{1}{4}$ -in. OD aluminum tubing, and water tubes which are void tubes perforated to allow water filling are used in boiling fuel assembly coolant channels to calibrate void coefficients of reactivity at zero power. Void and water rods similar to the void tubes but made to the same outside design as the boiling fuel rods can be used in place of fuel rods for at-power void coefficient calibration.

(3) Poison rods are provided either to control reactivity, to adjust power distribution, or to use as burnable poison. These rods have the same outside dimensions as boiling fuel rods and can be inserted in place of boiling fuel rods. Two types of poison rods are available: one is made of solid stainless steel, and the second, which is more effective, is made of $\frac{1}{4}$ -in. diameter 2 w/o boron-stainless steel clad with $\frac{1}{16}$ in. of stainless steel.

6. Core Design - Heat and Hydraulics

Introduction

From the standpoint of heat transfer and fluid flow, the design of the core for a boiling water reactor with an integral superheater is complicated by the thermal coupling which exists between the boiling and the superheating regions. For a given operating pressure and exit steam temperature, the fraction of the reactor power generated in the superheater remains virtually constant regardless of the total reactor power level. For BORAX V, which operates at 600 psig and an exit steam temperature of 850°F, 83% of the power is used to turn the feedwater into saturated steam, and 17% is used to superheat the saturated steam from 489°F to 850°F. In order to transfer 17% of the reactor power to the superheated steam, about 19% of the fissions must occur in the superheater region, because some of the fission energy ends up in the water moderator and because there are conductive and radiative heat losses from the superheater assembly. The problem is to design the two sections so that they operate properly as a unit while staying within the individual limitations upon heat flux, surface temperature, and pressure drop.

The core size, as well as the design of the fuel rods and boiling fuel elements, was based almost entirely on mechanical considerations. The principal effort for the boiling region was to determine its performance capability. The situation for the superheater region was quite different since, based on heat transfer considerations, a fair amount of flexibility in design existed. The elements finally selected could be fabricated without a fuel-development program.

Two basic changes have been made in the core for BORAX V since the work presented in ANL-6120. The boiling fuel assembly now consists of 49 rods, each 0.375 in. in diameter, rather than of 100 rods, each 0.260 in. in diameter. The superheater fuel assembly remains similar in concept, but each of the 5 elements per assembly now contains 4 fuel plates rather than 5, and each has 3 coolant channels, 0.062 in. thick, rather than 4 coolant channels, each 0.045 in. thick. In addition, the rows of disk spacers between fuel plates have been replaced by square wire rods running the entire length of the fuel plates. These basic changes will be discussed more fully later in this section.

a. Experimental Work

Flow characteristics and critical heat flux in the boiling region have been investigated. Convective heat transfer coefficients for steam in turbulent flow in round and rectangular flow passages have been determined experimentally. Tests have determined the extent of thermal distortion of the superheater fuel plates.

(1) Boiling Burnout Tests

Results of natural-circulation stability and boiling burnout tests, using parallel tubes with 0.332-in. ID to simulate flow outside of the fuel rods in the formerly planned 100-rod boiling assembly, were reported in ANL-6120. Additional natural circulation boiling burnout tests have now been run, by R. P. Anderson and E. A. Spleha of Argonne National Laboratory, with parallel tubes to simulate the performance of the 49-rod assembly. By means of two parallel tubes, 24 in. long and 0.562 in. ID (corresponding to the equivalent diameter of the fuel rods away from the side walls of a boiling fuel assembly when the rods are uniformly spaced within the assembly), heat fluxes up to 514,000 Btu/(hr)(ft²), the maximum obtainable from the power supply on the test loop, were obtained without the occurrence of instability or burnout. A further test was made on the same tubes after slightly flattening the middle 18 in. of the tubes to produce an equivalent diameter in this region of 0.474 in. (the equivalent diameter of the boiling fuel assembly as finally designed). Again a heat flux of 514,000 Btu/(hr)(ft²) was obtained without the occurrence of instability or burnout.

In order to overcome the limitation on the power supply, a different test loop and a new test section were used next. The earlier test section had a uniform wall thickness over the heated length and therefore produced a uniform heat flux. For this final test, in an effort to produce an axial power distribution similar to that occurring in the reactor (see Fig. 14), the two similar parallel tubes had a varying wall thickness over their lengths. The minimum wall thickness occurred 8 in. from the inlet end of the section, and the thickness increased uniformly from that point to each end. The consequent axial variation in heat flux had a peak-to-average heat flux ratio of 1.35. The results of the test are given in Table II. Instability was first observed during Run #11, and the amplitude of the oscillations increased with increasing power levels on successive runs until burnout occurred during Run #14, with a peak heat flux of 1.09×10^6 Btu/(hr)(ft²) and an average of 8.05×10^5 Btu/(hr)(ft²).

Fig. 14

Representative Local-to-average Axial Flux

Table II

BOILING BURNOUT TEST DATA

Run No	Power, kw	Average Power Density, kw/l	Maximum Power Density, kw/l	Average Heat Flux $\times 10^{-3}$, Btu/(hr)(ft ²)	Maximum Heat Flux $\times 10^{-3}$, Btu/(hr)(ft ²)	Inlet Subcooling, °F	Exit Quality, %	Total Flow Rate, lb/sec	Inlet Water Velocity, ft/sec
1	30.4	226	306	212	287	24	1.4	0.620	5.28
2	30.4	226	306	212	287	17	1.4	0.6139	5.23
3	40.56	301	408	282	382	20	2.8	0.6203	5.28
4	50.02	371	503	350	447	20	4.0	0.6276	5.34
5	60.8	451	611	425	576	37	5.3	0.6230	5.31
6	72.1	536	726	505	684	33	10.3	0.5804	4.94
7	79.9	594	805	556	753	38	12.2	0.5417	4.62
8	90.1	669	906	630	854	49	14.9	0.5164	4.40
9	90.1	669	906	630	854	42.8	14.9	0.517	4.40
10	94.8	704	954	663	898	44.9	17.5	0.488	4.16
11	100.0	744	1008	697	944	46.7	20.2	0.455	3.87
12	105.8	786	1065	736	997	47.1	20.1	0.487	4.15
13	110.5	820	1111	772	1046	48.9	21.6	0.471	4.01
14	115.1	855	1161	805	1090	BURNOUT			

(2) Superheated Steam Heat Transfer Experiments

The following is quoted from the Summary and Conclusions of a report on the convective heat transfer coefficients for superheated steam:(1)

SUMMARY AND CONCLUSIONS

"Forced convection heat transfer to superheated steam in turbulent flow was experimentally investigated, utilizing two channel geometries: a round tube, 0.333 in. ID x 12.00 in., and a thin rectangular channel, 0.047 in. wide, aspect ratio, 26.6 : 1, and 12.00 in. in length. The experiments encompassed a Reynolds number range from 20,000 to 370,000, pressure range from 300 to 1500 psia, inlet superheat from 5 to 160°F, and film temperature differences from 30 to 550°F.

"Concerning the round channel experiments, the following conclusions may be drawn:

"1. Convection coefficients may be predicted within ± 10 per cent, using the following equations and the properties found in Tables 1 and 2:

$$\frac{hD_e}{k_f} = 0.0157 \left(\frac{GD_e}{\mu_f} \right)^{0.84} \left(\frac{c_p \mu}{k} \right)_f^{1/3} \left(\frac{L}{D_e} \right)^{-0.04}$$

$$\text{for } 6 < L/D_e < 60$$

and

$$\frac{hD_e}{k_f} = 0.0133 \left(\frac{GD_e}{\mu_f} \right)^{0.84} \left(\frac{c_p \mu}{k} \right)_f^{1/3}$$

$$\text{for } L/D_e > 60$$

"2. Particular attention should be paid to the type of unheated entrance section preceding the heated length of channel. The first of the above equations is applicable when this precedent section is 16 diameters in length. It may be safely used (with respect to limiting surface temperatures) for shorter unheated sections and should closely apply for a more fully developed velocity profile (unheated $L/D_e > 16$), because of the relatively weak influence of L/D_e .

"3. Making allowances for the unheated entrance section, the data substantiates the experiments of McAdams, Kennel, and Addoms⁽²⁾ within ± 5 per cent, even to a Reynolds number of 370,000.

"4. The Dittus-Boelter equation,

$$\frac{hD_e}{k_f} = 0.023 \left(\frac{GD_e}{\mu_f} \right)^{0.80} \left(\frac{c_p \mu}{k} \right)_f^{1/3},$$

when compared to the present study, yields values of the coefficient which are 11.5 per cent high in the lower Reynolds range (25,000) and 1.8 per cent high in the upper range (370,000). It is conjectured that this is due, in part, to the unsettled question of correct steam conductivity values.

"Concerning the thin rectangular channel experiments, the following observations are made:

"1. The vena contracta caused by the sharp-edged entrance condition influences [decreases] the heat transfer coefficient to a length of approximately 75 equivalent diameters, including the unheated entrance length of 13 diameters.

"2. There is no observable difference between the rectangular data and the round channel correlation in the L/D_e range from 60 to 121. Even if it is postulated that the flow is not fully established in this region, there is no evidence that any serious decrease in coefficient will occur beyond this range, even according to the data of Levy, Fuller, and Niemi,⁽³⁾ whose data run to an L/D_e of approximately 180."

(3) Thermal Distortion of Superheater Fuel Element

The effects of temperature gradients in the superheater fuel element have been investigated. A simulated superheater fuel element with stainless steel, rather than stainless steel plus UO_2 , fuel plates was heated to 550°F, and subsequently one side of the element was dipped into liquid nitrogen at -320°F. No measurable distortion resulted. Additional tests of thermal distortion have been conducted on a two-plate, simulated fuel element radiantly heated on the outer surfaces while air flowed through the enclosed coolant channel. Thermocouples were attached to the outer surfaces to measure the axial and transverse temperature gradients. Tests were run with film temperature drops and axial and transverse gradients simulating reactor conditions. The variation of the overall thickness (two plates plus coolant channel) of the assembly during heating was measured to determine the distortion. No significant distortions were observed.

b. Analytical Work

The design of the core was established analytically prior to the completion of the experimental work just reported. The experimental work was of importance, however, in that it indicated the maximum power available from the boiling region and the conservatism of the superheater design.

(1) Boiling Region

The change in the boiling fuel assembly from one-hundred 0.260-in. diameter rods to forty-nine 0.375-in. diameter rods, which was dictated by considerations of fuel handling and fuel fabrication, resulted in revisions in the heat transfer and hydraulic analysis of the boiling core. The flow area per assembly was unchanged, but the heat transfer surface was reduced. In spite of this reduction, the resultant increase in equivalent diameter of the coolant channels increased the burnout heat flux, so that the allowable total power production was also increased. The rods of two different diameters are compared in Table III.

Table III

COMPARISON OF BOILING FUEL RODS

Rod OD, in.	No. of Rods/ Ass'y	Total Rod Cross-sect Area, in. ²	Flow Area, in. ²	Heating Surface, ft ²	Frictional D _e , in	Burnout q" Btu/(hr)(ft ²)	Design Max. q" Btu/(hr)(ft ²)	Max Temp, °F
0.260	100	5.31	8.64	13.65	0.364	390,000 (based on D _e = 0.332 in.)	300,000	2000
0.375	49	5.41	8.59	9.60	0.474	1,090,000	>600,000	5000

The most recent test results on natural-circulation boiling burnout indicate that the maximum allowable heat flux can safely be set in excess of 600,000 Btu/(hr)(ft²) (corresponding to a calculated fuel rod center temperature of 5000°F).

A series of problems was run to determine the optimum riser height at which the increase in driving head, due to additional height, is offset by the increased frictional pressure drop. In all cases considered, as the riser height increased, the head gain exceeded the loss. The design height of 5 ft above the top of the core left sufficient space above the riser for steam separation.

The spacing of the rods within the boiling fuel assembly is a problem because the burnout heat flux for a natural-circulation boiling water reactor is an unknown function of the equivalent diameter. The increase in burnout heat flux is nonlinear with increase in equivalent diameter. If the rods are spaced uniformly (i.e., if each rod has associated with it $\frac{1}{49}$ of the flow area within the assembly), the edge and corner rods, due to the presence of the wall of the fuel assembly box, have lower equivalent diameters than the interior rods. Figure 15 shows the variations of the equivalent diameters of the edge, corner, and central rods as a function of rod spacing. A 0.500-in. rod spacing was selected for BORAX V, giving a resultant 0.47-in. equivalent diameter for rods in all three locations. This spacing tends to bunch the rods within the box and leaves an additional water gap around the outside of the 7 x 7 array. The additional water gap causes an increase in the peak-to-average flux within a fuel assembly, as shown in Fig. 16. The peak-to-average flux in the assembly goes from 1.17 with uniform spacing to 1.26 with the $\frac{1}{2}$ -in. pitch, but the combined effect of the improved burnout heat flux and more severe flux peaking is to allow an increase in the maximum reactor power.

Fig. 15

Equivalent Diameter as a Function
of Rod Pitch

Fig. 16

Water Gap Flux Peaking

A parametric study has been carried out for the boiling region of the reactor. A large number of problems were submitted for solution using the RECHOP (RE160) program, with the degree of subcooling, the pressure drop due to an orifice at the boiling assembly inlet, and the exit void volume fraction as parameters. A typical plot of the results of a series of problems is shown in Fig. 17, in this case for 8°F subcooling and no orifice. For a given reactor power, radial flux plots can be utilized together with this type of curve to integrate the total flow and the steam flow across the core. For one subcooling (to be found by iteration) there is an energy balance between the incoming feedwater, the reactor power, and the outgoing steam. The void distribution can then be determined and compared with that which was used to obtain the flux plot originally used. Successive iterations may then be employed to obtain a better flux plot, etc., if necessary.

Fig. 17

Typical Hydrodynamic Parametric Study Results

Using this technique with flux plots derived from Fig. 26 (results of an R-Z calculation using PDQ on a core with a central superheater and distributed voids), the information in Table IV was obtained for BORAX V operating at 20 Mw with a central superheating core:

Table IV

OPERATING CHARACTERISTICS - CORE WITH CENTRAL SUPERHEATER

Power, Mw	20
Boiler	16.6
Superheater	3.4
Average Power Density, kw/l	
Boiler Region	95
Superheater	80
Peak-to-average Neutron Flux	
Boiler	
Axial	1.33
Radial	1.45
Local Peaking	1.30

Table IV (Cont'd.)

Superheater	
Axial	1.30
Combined Radial and Local Peaking	1.20
Void Volume Fraction - Boiler Region	
Average Core Exit	0.28
Maximum Core Exit	0.47
Average Distributed	0.09
Heat Flux, Btu/(hr)(ft ²)	
Boiler	
Average	123,000
Maximum	310,000
Superheater	
Average	55,000
Maximum	100,000
Fuel Element Temperature, °F	
Boiling Rod	
Max at Center	3100
Max at Surface	515
Superheater	
Max at Center	1125
Max at Surface	1110
Pressure Drop through Superheater Core, psi	
First Pass	3
Second Pass	4

For this case the maximum heat flux in the boiler region is well below the limit imposed by burnout considerations, so it will be possible to operate at significantly higher power levels. The maximum power level will be determined during the experimental program on the basis of type of core arrangement, flux profiles, fuel temperatures, and stability criteria. Operation at power levels as high as 40 Mw is contemplated.

A parametric study similar to that for natural circulation has also been carried out for forced-circulation flow using the RECHOP code. The driving head in this study was furnished by a long riser (varying from 5 to 125 ft) which acted as a pump while allowing the natural-circulation analysis to be performed.

(2) Superheater

Although the nominal design power of BORAX V is 20 Mw, the superheater has been designed to produce 850°F steam when the reactor is operating at 40 Mw. These conditions will not permit the maximum superheater fuel plate temperature to exceed the design value of 1200°F.

The chief reason for revising the superheater fuel assembly design shown in ANL-6120 was the fact that the calculated maximum surface temperature with the 5-plate element was only 1100°F rather than the design limit of 1200°F. This indicated that the design was over-conservative and failed to take maximum advantage of the properties of the fuel-element materials. The disk-type spacers between fuel plates have been replaced by square wire rods in order to lower the fabrication costs.

Because a superheater is essentially a gas-cooled reactor, it has a high fixed coolant temperature rise. Therefore, it is possible to decrease the amount of heat transfer surface without as great a corresponding increase in the fuel element surface temperature. In the revised design a reduction of 25 per cent in the heating surface accompanied by an increase in the channel thickness from 0.045 to 0.062 in. (which results in a reduced heat transfer film coefficient) caused only a 100°F rise in the maximum surface temperature. This, together with a reappraisal of hot channel factors, resulted in a maximum surface temperature of 1160°F with the 4-plate element. An additional beneficial result was a decrease in pressure drop through the two passes of the superheater. Also, the number of fuel plates to be fabricated was decreased 20 per cent, from 5 to 4 plates per element, with a consequent reduction in fabrication costs. The neutron economy has been improved because there has been a reduction in the amount of stainless steel and a slight increase in the moderator associated with each element. The ability to dissipate the shutdown heat by radiation and conduction has been increased because there is one less radiating surface to be considered. The differences between the two elements are summarized in Table V, which applies to the central superheater when the reactor is operating at 40 Mw.

Table V

COMPARISON OF SUPERHEATER FUEL ELEMENTS

	<u>5-Plate Element</u>	<u>4-Plate Element</u>
Maximum Surface Temperature, °F	1100	1160
Pressure Drop through 2-pass Core, psi	40.5	26
Thickness of Water between Elements, in.	0.335	0.367
Approximate % of Stainless Steel in Elemental Cell	28.1	24.4
Average Heat Flux, Btu/(hr)(ft ²)	75,700	110,000

Various superheater fuel element configurations were considered for use in single and multipass cores. A plate-type element was selected, with the steam to make two passes through the core.

The design of the superheater is indicated in Fig. 8. Each assembly contains five elements, with each element made up of four fuel plates enclosed in a stainless steel box. The four plates enclose three coolant channels, with the channel thickness maintained by the two side-plates and two 0.062-in. square wire spacers running axially through each coolant channel. The fuel plates are made of 0.014-in.-thick, highly enriched UO₂-stainless steel cermet meat clad with 0.008-in.-thick Type 304 stainless steel.

Figure 9 shows a mockup of a superheater fuel assembly. The outer plates in each element are nominally half-loaded since the coolant flows past their inside surfaces only. The heat loss to the moderator water between adjacent elements is reduced by a static-steam-insulating space between the 4-plate element and the surrounding box. The outer plate is used as one boundary of this insulating space in order to reduce the amount of stainless steel in the core and to facilitate the radiation of the shutdown heat from the element to the moderator water. In addition to serving as moderator, the water between elements is a sink for the shutdown after-heat. It is possible to remove the heat corresponding to the reactor operating at 0.8 Mw with no steam flow through the superheater without exceeding the superheater fuel element surface design temperature of 1200°F. About one-third of the shutdown heat is transferred by radiation and two-thirds by conduction.

Two types of superheaters are planned for BORAX V: a central superheater in which the twelve fuel assemblies surrounding the cruciform control rod through the center of the core are superheating

assemblies; and a peripheral superheater, in which the middle four assemblies on each of the four sides of the core are superheater assemblies. In the design of the superheater elements, peak-to-average axial and radial flux distributions in the superheating region of 1.3 and 1.2, respectively, were assumed. Also, although the nominal reactor power of BORAX V is 20 Mw, the superheating region was designed to heat the steam generated when the reactor is operating at 40 Mw. The major emphasis in the design was placed on the central superheater, since the smaller number of fuel assemblies for this case appeared to make it the most severe.

Figure 18 shows the calculated temperature profiles of steam and fuel element surfaces for the hot and average channels for the central superheater when the reactor is operating at 40 Mw. The peak temperature is 1160°F, safely below the design maximum of 1200°F. Engineering hot channel factors of 1.15 for the enthalpy rise and 1.36 for the film temperature drop were calculated and used to obtain the curve for the hot channel, together with the nuclear flux peaking factors listed in Table I. The convective heat transfer film coefficients for superheated steam were obtained by means of the correlation of McAdams, Kennel and Addoms:⁽²⁾

$$\left(\frac{h}{c_p G} \right) \left(\frac{C_p \mu}{k} \right)_f^{2/3} = \frac{0.0214 [1 + 2.3(D_e/L)]}{(D_e G / \mu_f)^{0.2}} .$$

Fig. 18

Superheater Axial Temperature Profile-40 Mw

The film coefficients were then multiplied by 0.83 to account for the maximum deviation reported in the original work and by 0.90 to account for the use of flat plates. The experimental work of Heineman on heat transfer coefficients, reported above, indicates that these coefficients are conservative, since his data correlate quite well with those of McAdams. Heineman found no difference between results obtained with flat plates and round tubes.

Since the superheater region of the core is essentially a gas-cooled reactor, corresponding hot channel problems are associated with a high temperature rise of the coolant. The mixing between passes reduces this increase, and the use of spacers between the fuel plates lowers the hot channel factors by maintaining the coolant channel dimensions.

The ability to lose shutdown heat by radiation and conduction under a loss-of-flow accident is a mixed blessing, since heat is lost from the assembly at all times. Figure 19 shows the ratio of heat transferred to the moderator to that transferred to the steam as a function of reactor power. At 20 Mw, the nominal design point, 3% of the superheater output is "lost" to the moderator. To insure that the coolant flowing through an exterior channel receives the same amount of heat from both bounding plates, 55% as much heat should be generated in an exterior plate as in an interior plate.

Fig. 19

Heat Loss to Moderator as a Function
of Reactor Power

In spite of the greater proportion of heat lost to the moderator at low-power operation, it is possible to operate BORAX V at as low as 1 Mw and still produce 850°F steam. (Since BORAX V is an experimental facility it will operate at off-design points, although perhaps not at extremes such as 1 Mw.) Figure 20 shows the different temperature distributions on interior and exterior plates and in interior and exterior coolant channels when operating at 1 Mw.

Fig. 20

Superheater Axial Temperature Profile-1 Mw

(3) Effect of Scale Formation

The presence of the aluminum shroud and fuel assembly boxes in an otherwise stainless steel system is expected to accentuate the problem of crud formation and deposition on heat transfer surfaces. The extent of the problem is not known at this time, because neither the deposition rate nor the effect of the cyclic operation on scale flaking can be determined. An attempt has been made to evaluate the effects of this scale, however.

Using a value of $0.44 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$ for the thermal conductivity of the scale (this is the value given for EBWR scale in ANL-6136, Water Chemistry and Fuel Element Scale in EBWR by Breden, Charak and Leyse) at the peak heat flux [$310,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$] in the boiler region, each mil of scale causes a temperature increase of 59°F in the fuel. Therefore, a 5-mil scale thickness would increase the maximum boiling fuel rod centerline temperature from 3100 to 3395°F at 20 Mw.

Uniform scale buildup will decrease the equivalent diameter of the coolant channels, with a slight deleterious effect on the burnout heat flux.

Scale buildup in the superheater fuel assemblies has also been considered. The degree of scale formation is one of the items to be determined in the operation of BORAX V. The effect on fuel plate temperature should not be serious, however, as all scale should appear in the first pass, where temperatures do not approach the design limitation. The effect on flow is of interest, although if the scale deposition is uniform within the superheater the flow restriction should also be uniform so that the result would be a uniform increase in pressure drop through all coolant channels.

(4) Control Rod Cooling

Heat generation in the control rods does not constitute a problem in BORAX V. Under normal operating conditions, the center of the $\frac{3}{16}$ -in. thick Boral is only about 2°F hotter than the edges, and in the unlikely event that the control rod is pushed against a shroud which forms one wall of the control rod channel, so that all heat must be removed from the other surface, the maximum temperature difference across the Boral is about 8°F. For this latter case there is an additional 12°F drop from the Boral to the water caused by the clad resistance and the contact resistance between the clad and the Boral. The pressure within the reactor vessel will maintain good contact between the clad and the Boral under operating conditions.

7. Core Design - Reactor Physics

a. Introduction

In designing BORAX V as a reactor with a major emphasis on mechanical versatility, engineering considerations have, for the most part, established the physical form of the core. Nuclear calculations were made to establish the feasibility of the design rather than to optimize a particular parameter.

In order to determine the fuel loadings for the reference design, two arbitrary conditions were imposed: (1) the reactor is to be just critical with 15% average steam voids in the moderator of the boiling section under hot, clean conditions, and (2) approximately 19% of the total power is to be produced in the superheating section. (Fifteen per cent of the boiler moderator in steam voids corresponds to a total power of 30 Mw.) After these requirements were satisfied with loadings for a just-critical reactor, the necessary additional loading for override of fission product poisons was determined.

Three core configurations (a central superheater with peripheral boiler, a central boiler with peripheral superheater, and a boiling core) are included in the reference design, and relative loading requirements are not the same for all cases. The boiling rods are loaded according to the requirements for the core with central superheater because this case represents the maximum enrichment required for the rods. The reactivity worth of these rods can be depressed in the other two configurations by substituting a number of judiciously placed poison rods for fuel rods.

The superheater loadings present a different problem. Since the reactivity worth in the peripheral location is much less than in the central location, the peripheral superheater assemblies require a considerably greater loading. As the design is not mechanically amenable to changing the loading, two separate sets of superheater assemblies are supplied.

b. General Procedure and Methods

Initially, a survey series of two-group, one-iteration problems were run on the UNIVAC with the one-dimensional code, RE-6.⁽⁴⁾ Constants for these problems were obtained by use of the MUFT-4⁽⁵⁾ and SOFOCATE⁽⁶⁾ codes. Homogenized number densities, corrected for disadvantage factors, and resonance self-shielding factors were obtained at chosen points for the input. Constants at intervening points were found by interpolation.

The disadvantage factors for the boiling rods were calculated by modification of a method suggested by Copic and Thie. The method used involves (1) calculation of diffusion-theory flux ratios for the cylinder; (2) transformation of the cylinder to an "equivalent" slab; (3) calculation of P_3 and diffusion-theory flux ratios for this slab; and (4) estimation of P_3 flux ratios for the original cylinder. This procedure was used because the P_3 and SNG¹¹ codes for cylindrical geometry and Amouyal's method⁽³⁹⁾ gave physically unreasonable results in this case. The self-shielding factors were obtained from the results of Dresner⁽⁷⁾ and of Hellstrand.⁽⁸⁾ An error of 10% in this factor means about 1% error in k_{eff} .

The results of the preliminary problem matrix for the central superheater are shown in Fig. 21, in which k_{eff} vs. $g U^{235}/\text{boiling rod}$ is shown, and in Fig. 22, in which the fraction of power developed in superheater vs. boiler loading is given. In each case the superheater loading in grams of $U^{235}/\text{assembly}$ appears as a parameter. From the curves of Fig. 21, four loading pairs giving a critical reactor are found. These pairs, along with the pairs, from Fig. 22, for 19% of total power in the superheater, are plotted in Fig. 23 to give two curves which intersect at the desired point. Curves for the peripheral superheater are similar

in form, although lines of constant superheater loading in the plot corresponding to Fig. 21 are barely separated, indicating the relatively low worth of loading in this position.

Fig. 21

K_{eff} vs. Boiler Loading for Core with Central Superheater

Fig. 22

Per Cent Power in Central Superheater vs. Boiling Fuel Rod Loadings

Fig. 23

Loading for Core with 19% of Power
Developed in Central Superheater

Rather large values of the superheater loading in the peripheral case were predicted by the one-dimensional, RE-6 code in order to meet the power fraction requirement. In order to check the assumption of cylindrical symmetry, the results of an X-Y PDQ⁽⁹⁾ problem were analyzed, and it was found that the RE-6 source integrals for the peripheral superheater needed to be increased by 2.4% to agree with the PDQ results. Similar X-Y problems were run for the boiling core and the core with a central superheater. The values of k_{eff} and the superheater power fraction obtained from these and four-group RE-6 problems are compared in Table VI with the results predicted by the graphical analysis described above.

Table VI

COMPARISON OF CRITICAL CALCULATIONS

	Boiling Core	Core with Central Superheater	Core with Peripheral Superheater
<u>Graphical</u>			
Boiler Loading, g U ²³⁵ /Rod	11.1	13.85	11.0
Superheater Loading, g U ²³⁵ /Assembly	-	400	705
<u>Four-group, RE-6 Check</u>			
k_{eff}	0.9990	1.0007	0.994 ^a
Per Cent of Power in Superheater	-	19.3	18.6 ^a
<u>PDQ X-Y Problems, 3 Groups</u>			
k_{eff}	0.9943	0.9964	0.9970
Per Cent of Power in Superheater	-	19.4	21.0

^aRun as three-group problem.

Thermal flux plots for the superheating configurations as obtained from the PDQ problems appear in Figs. 24 and 25.

c. Axial and Radial Distribution of Steam Voids

In the foregoing analysis the at-power condition was simulated by the assumption of a uniform distribution of steam voids. Thermal and fast constants were all obtained for the reactor at operating temperature (489°F) with a uniform void distribution.

An examination of the effect of a nonuniform void distribution was made by representing regions of different void content (average is approximately 15%) on a PDQ R-Z problem (two groups). The resulting thermal flux plot is given in Fig. 26 for the central superheater case. The axial reflectors contain a considerable amount of stainless steel which is also represented. The risers and lower nozzles from the superheater assemblies contain long, open steam passages and are simulated by a definite void fraction based on the geometrical volume involved. Some thermalization of the fast flux above the stainless region is evident. Relative fission densities, referred to the superheater region as unity, are listed. From this problem, k_{eff} is 1.0163 and the superheater power fraction is 19.65%.

d. The Final or "Reference" Loadings

The critical loading of the boiling fuel rods obtained for the central superheater configuration has been increased to allow for override of equilibrium xenon and samarium. A similar consideration has been given to the two superheater loadings. As previously stated, the plan is to use only the one loading for the boiling rods and to substitute poison in the form of dummy fuel rods so as to adjust the fuel worth for the particular application. Calculations were made by a thermal utilization method⁽¹⁰⁾ to obtain an estimate of the worth of the poison rods. Both stainless steel rods and boron-stainless steel rods are provided. Under power-producing conditions, i.e., approximately 15% average steam voids, the peripheral superheating core will require about 3 boron rods per eight boiling fuel assemblies and the boiling core, one rod per 2 assemblies to adjust the excess reactivity to a proper value.

Burnup is only a minor consideration in the BORAX V superheating reactors since a relatively short period of operation is contemplated. There is an adequate fuel reserve for operation as a boiling core when the reactor is used on advanced boiling-water studies.

As a result of the foregoing considerations, the loadings given in the first column of Table VII were determined. Because of manufacturing difficulties, the boiling rods actually contain 16.3 g U^{235} /rod.

Fig 24

Plan of Thermal Flux - Central Superheater

Fig. 25

Plan of Thermal Flux - Peripheral Superheater

Fig. 26

Vertical Section of Thermal Flux - Central Superheater

In the boiling core, this slight increase of fuel content implies a reactivity increase of about 0.1%. The number of assemblies which have been calculated to give a cold, clean critical configuration (in cylindrical geometry) are as follows: 11 boiling assemblies (49 fuel rods/assembly), 33 central superheater assemblies (only 16 are actually available), and 9 peripheral assemblies. The worths of an assembly at the center and of one at the edge of the full 60-assembly core have been calculated for the three cases on the basis of a cold core made just-critical by adding a fictitious thermal absorber. The assembly in question is clean, i.e., contains no added absorber. The values appear in Table VII.

Table VII

COLD REACTIVITY WORTHS OF FUEL ASSEMBLIES

Core Description	Worth of One Assembly at Core Center, $\% \Delta k/k$	Worth of One Assembly at Core Edge, $\% \Delta k/k$
Boiling, 16 2 g U^{235} /Fuel Rod	2.72 (B)	0.34 (B)
Central Superheater 430 g U^{235} /SH Assembly 16 2 g U^{235} /Boiling Fuel Rod	1.00 (SH)	0.63 (B)
Peripheral Superheater 680 g U^{235} /SH Assembly 16 2 g U^{235} /Boiling Fuel Rod	3.00 (B)	0.21 (SH)

Note Letters within parentheses designate type of assembly added (i.e., superheater or boiling)

e. Reactivity Worths of Steam Voids, Temperature and Superheater Flooding

A uniformly distributed thermal poison was added to the reference loading to make a just-critical problem under at-power (hot, 15% average steam voids) conditions. This artifice was applied to all three configurations in order to allow comparison of the effect of variables of interest. The calculations were performed with two-group constants by the RE-6 code. The relations existing between these variables (steam voids, temperature, and superheater flooding) are presented graphically in Figs. 27, 28 and 29. Since the experimental program (see Section VI) may include determination of void worth in the boiler at temperatures below boiling, calculated values of void worth under appropriate conditions were obtained for comparison purposes.

A tabulation of the volume fractions, number densities, and heterogeneous correction factors used as input for obtaining the nuclear cross sections used in the foregoing calculations appears in Table VIII. These are given for the hot reactor with 15% average steam voids and for the comparable cold non-voided reactor.

Fig. 28

Reactivity vs Temperature for the
Core with Central Superheater

REACTIVITY VS TEMPERATURE
BORAX V WITH CENTRAL SUPERHEATER

Fig. 29

Reactivity vs. Temperature for the
Core with Peripheral Superheater

Table VIII
NUMBER DENSITIES AND VOLUME FRACTIONS

	At Power, 526.5°K				Shutdown, 293°K			
	CSH**	PSH**	Boiler	Reflector	CSH**	PSH**	Boiler	Reflector
<u>Loading</u>								
Grams U ²³⁵ /SH Assembly	430	680	-	-	430	680	-	-
Grams U ²³⁵ /Boiling Rod	-	-	16.2	-	-	-	16.2	-
<u>Volume Fractions</u>								
Water	0.40756	0.40756	0.44213	1.00	0.40756	0.40756	0.50710	1.00
Aluminum	0.17484	0.17484	0.22215	-	0.17484	0.17484	0.22215	-
Stainless Steel	0.17419	0.17066	0.04159	-	0.17419	0.17066	0.04159	-
UO ₂	0.00608	0.00961	0.22916	-	0.00608	0.00961	0.22916	-
Steam Voids	0.23733	0.23733	0.06497	-	0.23733	0.23733	0.00000	-
<u>Disadvantage Factors</u>	1.2359	1.3246	1.1079	-	1.3045	1.43106	1.1868	-
<u>Number Densities*</u>								
Hydrogen	2.372(-2)	2.321(-2)	2.410(-2)	5.307(-2)	4.371(-2)	4.390(-2)	3.545(-2)	6.692(-2)
Oxygen	1.213(-2)	1.255(-2)	2.151(-2)	2.654(-2)	2.188(-2)	2.233(-2)	2.671(-2)	3.346(-2)
Iron	9.357(-3)	8.769(-3)	2.356(-3)	-	9.040(-3)	8.335(-3)	2.237(-3)	-
Nickel	1.557(-3)	1.463(-3)	4.413(-4)	-	1.507(-3)	1.395(-3)	4.232(-4)	-
Aluminum	1.134(-2)	1.162(-2)	1.362(-2)	-	1.156(-2)	1.193(-2)	1.385(-2)	-
Chromium	2.406(-3)	2.254(-3)	6.057(-4)	-	2.324(-3)	2.143(-3)	5.751(-4)	-
U ²³⁵	1.232(-4)	1.864(-4)	2.399(-4)	-	1.190(-4)	1.772(-4)	2.277(-4)	-
U ²³⁸	8.876(-6)	1.343(-5)	4.492(-3)	-	8.575(-6)	1.276(-5)	4.265(-3)	-
<u>Resonance Self-shielding Factors</u>								
Aluminum	1.000	1.000	1.000	-	1.000	1.000	1.000	-
U ²³⁵ and U ²³⁸	1.000	1.000	.812	-	1.000	1.000	0.669	-

*The number densities given have been corrected by the disadvantage factors for use in the SOFOCAT code. The numbers in parentheses indicate the powers of 10 by which the tabulated values are to be multiplied to give nuclei/(barn)(cm).

**Central superheater core (CSH) peripheral superheater core (PSH).

The numbers listed have been calculated on the basis of a cell volume of $\frac{1}{60}$ of an entire core composed of boiler or superheater material; i.e., a proportionate part of the aluminum shrouds, control rod followers, etc., is homogenized with the cell. The four-energy-group structure is as follows:

Group 1 - 10 Mev - 0.821 Mev

Group 2 - 0.821 Mev - 5.53 kev

Group 3 - 5.53 kev - 0.625 ev

Group 4 - Below 0.625 ev.

Fission in U²³⁸ occurs primarily in Group 1 and resonance capture in Group 3.

The constants obtained are tabulated in Tables VIII and IX. The latter gives the adjusted thermal absorption cross section (poison added) to reduce the excess reactivity to zero when at power. Ages and diffusion lengths are also listed.

Table IX
NUCLEAR CONSTANTS

Group	D, cm	Σ_a , cm ⁻¹	Σ_f , cm ⁻¹	$\nu\Sigma_f$, cm ⁻¹	D, cm	Σ_a , cm ⁻¹	Σ_f , cm ⁻¹	$\nu\Sigma_f$, cm ⁻¹		
CSH Hot Unflooded					CSH Cold Flooded					
1	3 4127	0 0006	0 0495	0 0004	2 4885	0 0011	0 0848	0 0004		
2	1 5569	0 0003	0 0488	0 0006	1 0330	0 0003	0 0966	0 0006		
3	0 9108	0 0076	0 0452	0 0084	0 6060	0 0082	0 0937	0 0087		
4	0 4791	0 0715	-	0 0922	0 2513	0 0990	-	0 1228		
PSH Hot Unflooded					PSH Cold, Flooded					
1	3 4113	0 0007	0 0495	0 0007	2 4882	0 0012	0 0848	0 0007		
2	1 5552	0 0005	0 0487	0 0009	1 0323	0 0005	0 0965	0 0009		
3	0 9128	0 0104	0 0439	0 0129	0 6068	0 0112	0 0921	0 0135		
4	0 4778	0 0899	-	0 1345	0 2585	0 1210	-	0 1755		
Boiler Hot 15% Voids					Boiler Cold, 0% Void					
1	2 7284	0 0035	0 0603	0 0062	2 3835	0 0037	0 0776	0 0063		
2	1 2708	0 0021	0 0530	0 0011	1 0557	0 0021	0 0765	0 0011		
3	0 9698	0 0198	0 0439	0 0135	0 7681	0 0193	0 0681	0 0127		
4	0 4848	0 0992	-	0 1694	0 3224	0 1259	-	0 2115		
Reflector Hot					Reflector Cold					
1	4 2792	0 0011	0 0844	0	3 4082	0 0014	0 1059	0		
2	1 2308	0 00001	0 1196	0	0 9759	0 00001	0 1509	0		
3	0 7182	0 0008	0 1202	0	0 5695	0 0010	0 1517	0		
4	0 2592	0 0115	-	0	0 1594	0 0193	-	0		
At Power 526 5°K					Shutdown 293°K					
$(\Sigma_a + \Sigma_p)$ cm ⁻¹ Thermal Group τ cm ² L^2 cm ²	BC	CSH		PSH		BC	CSH		PSH	
		SH	B	SH	B		SH	B		
	0 1104	0 0752	0 1029	0 0997	0 1090	0 1401	0 1039	0 1306	0 1340	0 1383
	70 5	100 1	70 5	99 4	70 5	44 2	38 4	44 2	38 3	44 2
	4 391	6 37	4 71	4 79	4 45	2 30	2 42	2 47	1 93	2 33

Although temperature, void, and superheater flooding worths are all of operational interest, they are also of interest from the standpoint of potential hazards. A more detailed investigation of the flooding worth is graphically represented in Fig. 30, where, for the central superheater configuration, the results of radial-axial PDQ calculations are compared with RE-6 results. The PDQ values were obtained by representing the superheater as flooding from the bottom up. Further consideration of superheater flooding as a potential hazard is given in Section VII, B, 1, b.

f. Control Rod Calculations

The control rod detail may be observed in Fig. 36 and the rod arrangements in Fig. 31. It should be noted that $\frac{3}{32}$ in. of water is present on each side of the control rod blade. The effect of this and the adjoining aluminum shrouds were examined by running SNG⁽¹¹⁾ slab problems.

Fig. 30

Central Superheater Flooding Worths

Fig. 31

Superheater Loading Diagrams

A somewhat improved extrapolation length was predicted from the results of these problems. Using this length, the usual logarithmic boundary condition was employed in the thermal group of three-group PDQ problems to check the control rod worths under various conditions. Table X is a compilation of the results of these studies. The validity of these calculations will be checked by actual measurement in the reactor.

Table X
CONTROL ROD DATA AND VALUES OF REACTIVITY UNDER VARIOUS CONDITIONS

Condition	Configuration		
	Central Superheater	Peripheral Superheater	All Boiler
Number of Fuel Rods/Boiling Assembly Replaced by Boron-SS Poison Rods	0	3/8	1/2
Hot, Clean, 15% Average Voids, All Control Rods Out, % $\Delta k/k$	3.2	2.7	2.5
Cold, All Rods Out, (Maximum Reactivity) % $\Delta k/k$	14.4 [*]	15.2	13.5
Shutdown, Cold, All Rods In, % $\Delta k/k$	-14.3	-13.3	-15.1
Central Rod Out, All Other Rods In, Cold, 0% Voids, % $\Delta k/k$	-3.0 [*]	2.9 [*]	1.6 ^{**}
Two Opposite Peripheral "T" Rods Out, All Others In, Cold, 0% Voids, % $\Delta k/k$	-6.3 [*]	-8.3	-12.6
Total Worth of Rods, Rods All Out to Rods All In, % $\Delta k/k$	28.7 [*]	28.5 [*]	28.6
Max Worth of Central Control Rod, % $\Delta k/k$ Per Inch	94	14	13

*Calculated with superheater flooded under cold conditions

**The excess reactivity shall be adjusted so that the reactor cannot be made critical under any condition by the withdrawal of a single control rod

g. Fine Flux Structure

Radial power distributions obtained from the RE-6 results are plotted in Figs. 32, 33, and 34 for the three cases. The average power in each core region is normalized to unity. The water-gap peaking between assemblies and within control rod channels was examined by means of a number of SNG problems. The peak-to-average flux ratios as obtained from this series, as well as the overall values obtained from the RE-6 and PDQ results, appear in Table XI.

Fig. 32

Radial Power Distribution for the Boiling Core

Fig. 33

Radial Power Distribution for the Core with Central Superheater

Fig. 34

Radial Power Distribution for the Core with Peripheral Superheater

Table XI
PEAK-TO-AVERAGE VALUES OF POWER AND THERMAL FLUX

Core Type	RE-6*		PDQ**	SNG***
	Power	Thermal Flux	Thermal Flux	Thermal Flux
Boiling	1 67	1 64	-	-
Central Superheater				
Boiler	1 48	1 44	1.93	-
Superheater	1 03	1 02	1 36	-
Peripheral Superheater				
Boiler	1 54	1 54	-	-
Superheater	1 48	1 33	-	-
Ratio of Thermal Flux Peak in Region to Average in Fuel Between Boiling Assemblies	-	-	-	1 56
Across AI Control Rod Follower	-	-	-	1 36

*Reactor hot and with 15% average boiler voids Radial direction only
**Values are based on R-Z problem with a simulated void distribution Radial x axial values
***Reactor hot with 0% voids

8. Reactor Vessel

a. General

The general form of the reactor pressure vessel, as shown in Figs. 2 and 35, is a right cylinder closed at each end by an ellipsoidal head. The top head of the vessel is joined to the vessel proper by two bolted and gasketed flanges. The inside surfaces of the vessel are clad with stainless steel. The structural portion is fine-grained carbon steel and the assembly of the several parts is by fusion welding. Thermal insulation is stainless steel wool within a stainless steel skin. Support is by means of a bottom-mounted skirt, and the vessel is installed in a ventilated reactor pit. In order to provide the versatility desired in an experimental facility, the vessel is equipped with numerous nozzles of various types, as shown in Fig. 35 and Table XIII (p. 80).

b. Design Criteria

Table XII shows the principal design and expected operating conditions.

With one notable exception, vessel design and fabrication are based on the ASME Boiler and Pressure Vessel Code, Section I, 1956 Edition, including addenda and interpretations. In the interests of conservatism, the Code-allowed working stress of the BORAX V vessel metal has been reduced by 3000 psi, an amount equal to one-half the calculated neutron and gamma-induced thermal stress. Compliance with State of Idaho requirements is automatic, since the State has adopted the ASME Code, without exception.

Fig. 35
Reactor Pressure Vessel

Table XII

REACTOR VESSEL DESIGN CRITERIA FOR BORAX V

Vessel Height, Inside, Exclusive of Cladding, ft	16
Vessel Diameter, Inside, Exclusive of Cladding, in.	66- $\frac{3}{8}$
Cladding Thickness, Nominal, in.	$\frac{3}{16}$
Design Pressure, psig	700
Design Temperature, °F*	650
Operating Pressure, psig	600
Operating Temperature, °F*	490
Hydrostatic Test Pressure, psig	1275
Neutron and Gamma-induced Thermal Stress in Vessel Shell, Max, psi	6000
Load Transferred to Top and Bottom Head by Core Hold-down System, lb	75,000
Load Transferred to Bottom Head Due to Weight of Fuel, Core Structure, etc., lb	20,000
Impact Load, Downward, on Any Single 3- or 6-in. Nozzle in Top Head, lb	5000
Impact Load, Downward, on 3 or More Nozzles in Top Head, lb	15,000
Impact Load, Downward, on Each Control Rod Nozzle, lb	3000
Force, Due to Piping, on Each Forced-convection Suction Nozzle, Max, lb	3600
Force, Due to Piping, on Each Forced-convection Inlet Nozzle, Max, lb	2800
Force, Due to Piping, on 6-in. Steam Nozzle, Max, lb	1250
Force, on Each of the Remaining Nozzles, lb	500
Moment, Due to Piping, on Each Forced-convection Suction Nozzle, Max, in.-lb	129,000
Moment, Due to Piping, on Each Forced-convection Inlet Nozzle, Max, in.-lb	27,400

---

*The steam outlets and two of the instrumentation nozzles are thermally sleeved and are designed to conduct superheated steam at 900°F. The feedwater, spare heater inlet, and poison-injection nozzle are thermally sleeved and are designed to conduct water at 60°F.

Table XII (Cont'd.)

Moment, Due to Piping, on 6-in. Steam Nozzle, Max, in.-lb	48,000
Moments on Remaining Nozzles	Negligible
Lifetime Integrated Fast Neutron Flux at Vessel Wall, Min, nvt	5×10^{18}
Calendar Lifetime, Due to Fast Neutron Exposure at Vessel Wall, Min, yr	5

c. Vessel Life

Pertinent to the design of this vessel, there appear to be three areas of general agreement on the subject of change of characteristics or damage in structural metals due to fast neutron irradiation. These are:

- (1) Fast neutron radiation does cause some undesirable changes in structural metals, the change of primary importance being elevation of the ductile-brittle transition temperature.
- (2) Appreciable irradiation-induced changes in A-212 steel do not occur below 5×10^{18} nvt.
- (3) Metal characteristics in heat-affected zones adjacent to welds may exhibit the highest rates of change.

There is no general agreement regarding rates of change with increasing irradiation, temperature effect on rate of change, the effects of neutrons at energies below 1 Mev, and the effects of rate of dose.

With the foregoing in mind, the following considerations and treatments were applied in this phase of the vessel design:

- (1) BORAX V is an experimental facility and thus has a relatively short useful life, estimated at about 5 years.
- (2) Design factors are conservative, but gross overdesign is avoided for reasons of economics. Total fast neutron exposure is therefore set at 5×10^{18} nvt. Based on a calculated 1.5×10^{11} fast neutrons/(cm²)(sec) at the vessel inner wall at 20-Mw thermal power, and with the anticipated operating schedules, this exposure should be acquired in about five calendar years.

- (3) In line with accepted practice, the vessel structural metal is made to fine-grain specifications in order to depress the transition temperature.
- (4) A system for monitoring vessel condition is provided by a series of coupons made from the vessel plate material and installed at the vessel inner wall in the zone of maximum flux. At appropriate times, coupon sets will be removed for testing. The coupons include base metal, weld metal, and heat-affected metal.
- (5) Coupon temperatures will be observed by thermocouples.

d. Shell, Heads and Flanges

The structural portions of the vessel shell and heads, except as noted in Section "g" below, are of carbon steel, A-212, Grade B Firebox Quality, made to A-300 fine-grain requirements with 6 to 8 McQuaid-Ehn grain size. Shell wall thickness is $1\frac{7}{8}$ in. and the heads $3\frac{1}{4}$ in., less cladding. The heads are 2:1 ellipsoidal and are seamless.

The inboard surfaces of the shell and heads are clad by metallurgical bond, according to A-264 requirements, with Type 304 stainless steel. No credit for cladding strength has been taken in the calculations about the vessel wall.

The structural portions of the vessel flanges are of carbon steel, A-105, Grade II, made to A-300 Class I requirements with 6 to 8 McQuaid-Ehn grain size. The upper flange is $8\frac{1}{4}$ in. thick and the lower $7\frac{13}{16}$ in. The greater thickness of the upper flange is due to the moments created by the core hold-downs.

There are 40 flange bolt assemblies, each consisting of 3-in., A.I.S.I. 4340, 175,000 psi U.T.S. steel, profiled and bored studs, with gauging pins and A-194, Grade 2H nuts. The required bolt prestress is calculated to be about 26,000 psi.

The flange bores are clad with Type 304 stainless steel, electrode-deposited by a method which limits the carbon content to approximate 0.08% maximum in the outer 0.100 in. of the cladding. The gasket area of the flanges is inlaid with Type 309-moly stainless steel.

The lower flange is faced to receive two concentrically located, spiral-wound stainless steel and Teflon gaskets. The inner gasket alone is the normal pressure seal; the outer gasket functions as a standby seal. Between the two is a leak-off groove. Leakage is piped away from the vessel to permit monitoring of inner gasket integrity.

e. Neutron Activation in Upper Head

Unless preventive measures are taken, the buildup of Fe^{59} in the upper vessel head, from the neutron-gamma reaction on Fe^{58} , leads to a personnel-exposure problem, when the head is removed or replaced, as a result of neutron streaming from the virtually dry superheater fuel assemblies.

Cross-section considerations make it clear that the major source of activation is the thermal neutron group. This is effectively eliminated by the use of a neutron shield consisting of a 1-in.-thick dished plate of Type 304 stainless steel with 1% boron. The major nozzle openings in the head are similarly shielded. The largest remaining effect will be due to the fast neutron component, which the neutron shield cannot effectively reduce. These fast neutrons, degrading through inelastic collisions, will lead to a certain amount of Fe^{59} in the head.

For purposes of calculation, all fast neutrons were assumed to originate from a point source at the geometrical center of the reactor. Consideration of geometry and attenuation factors gave a value of $3.7 \times 10^{10} \text{ n}/(\text{cm}^2)(\text{sec})$ for the effective fast neutron flux at the neutron shield. A ten-group energy analysis of the fast flux in iron, coupled with cross-section values, makes it possible to establish the specific neutron-absorption rate of Fe^{58} as a function of distance into the iron. This rate, $1.5 \times 10^{-3} \text{ absorption}/(\text{cm}^3)(\text{sec})$ per incident fast neutron, proves to be essentially constant with distance and, corrected for the ratio of activation to absorption and for isotopic abundance, yields a resultant $1.7 \times 10^{-6} \text{ activation}/(\text{cm}^3)(\text{sec})$ per fast neutron. The use of standard methods of calculating shielding and the incident fast flux indicated a saturation radiation level of 200 mr/hr at the surface of the vessel head.

f. Radiation Heating in Shell

To obtain the maximum condition, all calculations were based on 40-Mw operation and a core volume of 170 liters. The following values were used for the concentration of core constituents:

H_2O (Density $0.8\text{g}/\text{cm}^3$)	$1.12 \times 10^{22} \text{ atoms}/\text{cm}^3$
Al	$2.10 \times 10^{22} \text{ atoms}/\text{cm}^3$
U^{235}	$1.64 \times 10^{20} \text{ atoms}/\text{cm}^3$
U^{238}	$2.85 \times 10^{21} \text{ atoms}/\text{cm}^3$

Although this composition does not contain stainless steel, as do the reference cores (see Table I), it is actually a worse case because it allows more radiation to impinge on the vessel wall.

It was assumed that the source of the neutron and of prompt and delayed gamma radiations from the core could be treated as an infinite plane. Nine energy groups of gammas were considered, namely: 0.5, 1.0, 1.5, 2.0, 2.5, 3.0, 4.0, 6.0 and 8.0 Mev. Prompt gammas from neutron capture in the water moderator-coolant were also considered. The following results were obtained for: (1) a $2\frac{1}{16}$ -in. shell wall (including the $\frac{3}{16}$ -in. stainless steel cladding) with a shield, consisting of a 1-in. stainless steel plate with 1% boron, and (2) with no shield:

Radiation-induced Thermal Stress

<u>Mev/(cm³)(sec)</u>	<u>Watts/cm³</u>	<u>Δt, °F</u>	<u>Thermal Stress, psi</u>
<u>With Thermal Shield</u>			
Inner Edge, 3.74 x 10 ¹²	0.60	8.7	2520
Outer Edge, 0.63 x 10 ¹²	0.10		
<u>Without Thermal Shield</u>			
Inner Edge, 9.55 x 10 ¹²	1.53	19.2	5560 (say 6000)
Outer Edge, 1.26 x 10 ¹²	0.20		

These calculated radiation stresses, when added to the balance of the anticipated stresses, indicate that thermal shielding opposite the core is not justified. This is particularly true when one considers that the maximum radiation stress will be achieved infrequently. Furthermore, the allowable design working stress in the structural material has been reduced by 3000 psi.

g. Nozzles and Reinforcements

The vessel nozzles and nozzle extensions are A-312, Type 304 stainless steel with Schedule 80 wall thickness, except for the level column and blowdown nozzles, which are Schedule 160.

For the thermally sleeved nozzles, a "Y"-type, one-piece junction device was chosen for connecting the nozzle outboard ends to the sleeves in order to place the weldments away from the areas of high concentrations of thermal stress.

The nozzle reinforcements are Type 304 stainless steel forgings and centrifugal castings. Although this will certainly cause ring stresses at the point of juncture to the carbon steel vessel walls, calculations indicate these stresses will not be excessive, and certain distinct advantages will be achieved. The foremost among these has to do with

nozzle placement and alignment, which are stringently specified. This construction permits vessel stress-relieving to be completed after the welding-in of the reinforcements, but prior to installation of the nozzles. After relieving, the reinforcements are precision-bored to receive and position the nozzles. The specified placement tolerances are thus more readily obtained. The remaining welds are stainless-to-stainless, eliminating the need for later stress-relieving with its inherent warpage potential.

The nozzle complement is set forth in Table XIII.

Table XIII

NOZZLE LIST FOR THE REACTOR VESSEL

Function	Number	I.P.S. Size (in.)	Comment
Feedwater Inlet	1	3	Thermally Sleeved
Spare Heater Inlet	1	2	Thermally Sleeved
Poison Injection	1	$1\frac{1}{2}$	Thermally Sleeved
Level Column	2	1	-
Control Rod	9	4	Externally Shielded, Portable Internal Shield, Thermal Sleeve
Steam Outlet	1	6	Thermally Sleeved
Steam Outlet	4	3	Thermally Sleeved
Blowdown	1	1	-
Instrumentation	3	3	Internally Shielded
Instrumentation	2	3	Thermally Sleeved
Access	5	6	Internally Shielded
Oscillator	3	$1\frac{1}{2}$	Internally Shielded
Forced-convection Suction	2	14	Externally Shielded
Forced-convection Inlet	4	6	Externally Shielded
Flange Leak-Off	1	$\frac{3}{4}$	-
Total Nozzles	40		

h. Nozzle Shielding

In keeping with accepted practice, the several nozzles and nozzle extensions which penetrate the shield pit walls are equipped with either steel radiation sleeves or shields, or are offset to provide a stepped configuration. In addition, the 9 re-entrant control-rod nozzles are fitted with internal shielding.

i. Internal Attachments

The several reactor vessel internal supports, blocks, rings, etc., are Type 304 stainless steel and are attached to the vessel structural metal by welding. Finish-machining of these items will be done after stress-relieving.

j. Thermocouples

Eighteen thermocouple blocks, each containing 2 thermocouples, are attached to the outside surface of the vessel for temperature monitoring.

k. Vessel Supports and Guides

The vessel is supported from the bottom head by an A-7 steel skirt and four sole plates. The sole plates are attached by welding to 14-in.-wide flange beams cast into the bottom reinforced concrete support and shield slab.

The upper 8 in. of the skirt contains 20 vertical slits to allow for vessel thermal expansion.

Four equally spaced guide plates are provided at the vessel flanges. Matching guides are anchored to the reactor pit walls. These plates and guides permit thermal expansion to take place but resist lateral vessel movement.

l. Thermal Insulation

Vessel insulation consists of a 3-in. blanket of Type 430, fine stainless steel wool batts, applied at a density of 5 lb/ft³. These batts are supported by a series of A-107 steel rings shop-welded to the vessel and are enclosed by a skin of 20-gauge, Type 304 stainless steel, wired and bolted in place. Certain sections are removable for access to flange bolts, etc. Handhole covers are provided opposite each thermocouple block for access during preliminary testing. It is estimated that heat loss through this insulation will not exceed 25 kw.

The vessel nozzles and nozzle extensions inside the reactor pit are insulated to appropriate thicknesses and enclosed in 24-gauge galvanized iron jackets.

Insulation deterioration is expected to be at a minimum since the reactor pit is force-ventilated and the products of dissociation are thereby removed.

m. Stress-relieving

Stress-relieving was done, according to the ASME Code, at a maximum temperature of 1000°F in order to avoid sensitization of the stainless steel components.

n. Inspection and Testing

An extensive program of inspections and tests has been formulated to assure a degree of quality control of materials and workmanship commensurate with the intended use of this vessel.

9. Reactor Control and Instrumentation

a. Control Rods.

BORAX V has a total of 9 control rods. Five of them, one located at the center and 4 equispaced on a 12.5-in. radius, are cruciform shaped with 14-in. blades. The remaining 4 rods, located on a $17\frac{5}{8}$ -in. radius at the midpoint of the core edges, are "T"-shaped, each with a 14-in. and a 7-in. blade.

The design of a cruciform rod is shown in Fig. 36, and the "T"-shaped rods are similar in construction. The poison section of a control rod is made of $\frac{3}{16}$ -in.-thick Boral, which is 50 w/o B₄C in the meat, covered by a $\frac{1}{16}$ -in.-thick Type 304 stainless steel cladding spot-welded to stainless steel slugs through the Boral and seal-welded at the edges to give a total blade thickness of $\frac{5}{16}$ -in. The upper end of each blade contains a gas expansion space to reduce internal pressure buildup due to helium gas formation. The stainless canned Boral section of the rod is helium leak tested and autoclaved at reactor operating conditions to insure tightness. Each rod has bolted to it a $\frac{5}{16}$ -in.-thick X-8001 aluminum follower to reduce flux peaking in the control rod channels. Bolted to the bottom of each rod is a 3 x 3 x $\frac{5}{16}$ -in. cruciform stainless steel extension shaft which extends down through a reactor vessel control-drive nozzle and connects to a control-rod-drive-mechanism extension rod. Control rods may be removed from the reactor core structure, after disconnection from the control-drive mechanism, by means of the bail at the top of the rod.

Two sets of control rods are required. In the superheating cores, the control-rod stroke and follower is limited to 24 in. by the flood and drain manifolds at the bottom of the core structure. The control-rod stroke and follower length in the boiling core are 29 in., to give an unpoisoned upper reflector 5 in. thick for certain experiments.

Fig. 36

Cruciform Control Rod

b. Control-rod Drives

The control-rod drive mechanisms for BORAX V are those originally used in EBWR; these were made available because of modifications to EBWR. They have been changed to meet the stroke and speed requirements of BORAX V and to incorporate improvements and simplifications demonstrated to be desirable from EBWR operation.

Figures 37 and 38 show the drives after modification. The original EBWR stroke was 48 in. For BORAX V this is shortened to a 30-in. maximum. Adjustable limit switches then reduce this to either 29 in. for the boiling core, or to 24 in. for superheating core operation. Drive motors with speed-change gears are used on the 4 "T"-shaped outer rods and the center rod. Because of the greater worth of the center rod, its drive motor is geared to give the slowest speed. Two-speed motors are used with the other 4 cruciform intermediate rod drives, giving a slower speed when all 4 rods are operated as a group than when operated individually. In any case, the maximum rate of reactivity addition by means of control-rod withdrawal is limited to 0.05%/sec, based on about 15 sec from delayed to prompt critical.

For position indication, the gears driving the selsyn generator give a ratio of 2 revolutions of the selsyn to one of the lead screw, or 10 revolutions of the selsyn per in. of travel.

Three changes have been made as a result of EBWR operating experience. The external dashpot was found to be unnecessary and has been removed. The flushing arrangement has been changed to eliminate trouble from the setting of radioactive crud in the seal housing. A new flushing-water inlet between the guide bushing and seal is used, with the outlet remaining below the seal in the lantern ring section. For intermittent flushing during shutdown periods, a blowdown pipe is connected above the guide bushing. A baffle to cause a swirling motion in the flushing water is located in the seal housing to increase blowdown effectiveness. Another change is in the heat treatment of the chrome-plated 17-4 PH control-rod extension shaft. This shaft is heat treated at 1150°F for a 4 hr to give a hardness of Rockwell-C-33.

Beyond these changes, the original EBWR description in ANL-5607 applies. The mechanisms are located outside the reactor shield and below the reactor. A screw shaft and screw nut are used to translate rotating drive motion into the linear motion of the control rod. The control-rod shaft is connected to the screw nut by a triggering device in the form of a roller latch and 15-volt dc solenoid. Interruption of the dc power de-energizes the solenoid, thereby releasing the latch and scrambling the rod.

Fig. 37
Control Rod Drive
Mechanism

Fig. 38
Carriage Nut
Assembly

The control rod and drive are connected by the chrome-plated extension shaft which passes through the pressure-breakdown seal. Figure 39 shows the seal and the connecting joint.

Fig. 39

Connector and Seal Assembly

When a rod is raised, a 140-lb spring is compressed between the rod guide and latch carriage. At reactor pressures below 300 psi this spring is most effective, and maximum rod velocity is reached in 4 in. However, the main propulsion force is always reactor pressure. For a scram at 600 psi, the time for the rods to travel 24 in. (the point at which the dashpot begins to act) is estimated to be 0.23 sec. Figure 40 shows a plot of control rod displacement versus scram time.

c. Reactor Control, Electrical

The reactor is controlled from the control building located one-half mile from the reactor. The control console (see Fig. 41) is designed for two operators who control the reactor and the process systems. Process system control is described in Section III, C.

(1) Control System

The reactor is controlled by moving control rods in and out of the core. The control rods are driven by 3-phase, 60-cycle, 480-volt, reversing motors. The intermediate rods have two-speed reversing motors, while the center and outer rods have single-speed motors. A permissive interlock requires that all control rods be at the full "in" position before the control circuits can be energized. Likewise,

all scram signals must be reset. Operating the reactor-on key switch (key switch No. 1) then energizes the latch solenoids for each rod and allows operation of the control circuits. The motors are operated through conventional motor-starters with overload protection. The control circuits are operated at 120 volts ac.

Fig. 40

Control-Rod Scram Time

The control rods are raised and lowered by the operation of 3 lever switches at the control console. One lever switch controls the center rod only, a second operates the intermediate rods, and a third the outside rods. Interlocks provide that only one of the three lever switches at a time is effective in the "raise" position. A selector switch for the intermediate rods and one for the outside rods allows each rod to be operated individually or 4 rods to be operated as a group.

To keep the rod-motion time as short as possible, yet within safe limits of reactivity-addition rate, the intermediate rods, when operated as a group, move at one-fourth the speed they have when operated individually. Because the outside rods have a lower reactivity worth, operating them at the same speed, both for individual and group operation, permits reasonable rod-motion times and safe reactivity-addition rates. Change gears will allow a variation of control-rod speed depending on the control-rod worth for various loadings.

RECORDER PANEL DEVICE IDENTIFICATION

- RP 1
LR 3 MAKE UP WATER-STORAGE
LEVEL RECORDER
LR 2 FEEDWATER STORAGE LEVEL RECORDER
STARTUP SCALERS
- RP 2
TR 3 FEEDWATER TEMPERATURE RECORDER
TR-1 VESSEL & SHIELD TEMPERATURE
RECORDER
- RP 3
FI 3 LOW RATE FEEDWATER FLOW INDICATOR
TI 1 PROCESS TEMPERATURE INDICATOR
FR 3 FORCED CONVECTION FLOW RECORDER
PS 5, PS 7, & PS 8 POWER SUPPLIES
- RP 4
PR 2 REACTOR VESSEL PRESSURE
TR 2 REACTOR WATER OR STEAM
TEMPERATURE RECORDER
LC 1 REACTOR WATER-LEVEL CONTROLLER
PC 1 & PC 2 REACTOR PRESSURE CONTROLLERS
- RP 5
SCRAM ANNUNCIATOR
SUPERHEAT TEMPERATURE ANNUNCIATOR
FR 2 MAIN FEEDWATER FLOW RECORDER
- RP 6
ALARM ANNUNCIATOR
LR 1 REACTOR WATER-LEVEL RECORDER
V-1 1 THREE-ELEMENT LEVEL CONTROL
- RP 7
LOG FLUX RECORDER
FR 1 MAIN STEAM FLOW RECORDER
PS 1, PS 2, PS 3, PS 4 POWER SUPPLIES
POWER SUPPLY FOR V-1
- RP 8
TR 4 SUPERHEAT FUEL TEMPERATURE
RECORDER
PR 3 SUPERHEAT PRESSURE DROP RECORDER
RELAY POWER SUPPLY
- RP 9
FI 4 SUPERHEAT STEAM-VENT FLOW
INDICATOR
AREA RADIATION MONITORS
TRC 1 STEAM TO TURBINE TEMPERATURE
RECORDER
TC 1 DESUPERHEATER-TEMPERATURE
CONTROLLER
- RP 10
PR 1 STEAM TO TURBINE PRESSURE
RECORDER
FR 4 TURBINE BY-PASS STEAM FLOW RECORDER
HIC-56 SUPERHEATER-VENT EMERGENCY CONTROL
- RP 11
TV 16 TELEVISION MONITOR
TV 16 TELEVISION CAMERA CONTROL
FISSION BREAK MONITOR RECORDER
PERIOD-METER

CONTROL CONSOLE DEVICE IDENTIFICATION

- CC 1 REACTOR CONTROL
2-STARTUP COUNT RATE INDICATORS
1-PERIOD INDICATOR
2-HIGH-FLUX SHUTDOWN INDICATORS
1 LINEAR FLUX INDICATOR
4 INTERMEDIATE CONTROL ROD
POSITION INDICATORS & CONTROL
- CC-2 REACTOR CONTROL
LINEAR FLUX RECORDER
CONTROL POWER ON
RESET PUSH BUTTON & INDICATOR
LINEAR RANGE SWITCH
- CC 3 REACTOR CONTROL
SCRAM & ROD DROP CONTROL
PERIODY ROD DROP TIMER
1 CENTER CONTROL ROD POSITION
INDICATOR & CONTROL
4 OUTSIDE CONTROL ROD POSITION-
INDICATORS & CONTROL
1 PROCESS-TEMPERATURE INDICATION
SELECTOR
1 SUPERHEATER FUEL TEMPERATURE
RECORDER SELECTOR
1 REACTOR WATER OR STEAM-
TEMPERATURE RECORDER SELECTOR
4 SHUTDOWN BYPASS KEYS
ALARM ANNUNCIATOR ACKNOWLEDGE
& RESET
MICROPHONE
- CC 4 PROCESS WATER CONTROL
BORON DUMP CONTROL
FEEDWATER-SYSTEM CONTROL
PRE-HEAT SYSTEM CONTROL
REACTOR WATER DEMINERALIZER-SYSTEM CONTROL
- CC 5 PROCESS STARTUP & SHUTDOWN CONTROL
REACTOR VENT CONTROL
SUPERHEATER DRAIN & FLOOD SYSTEM
CONTROL
SUPERHEATER VENT CONTROL
FORCED CONVECTION SYSTEM CONTROL
- CC 6 PROCESS STEAM CONTROL
MAIN STEAM TO TURBINE CONTROL
MAIN STEAM TO ATMOSPHERE CONTROL
DE SUPERHEATER CONTROL
- CC-7 TURBINE CONTROL
TURBINE-GENERATOR SPEED & VOLTAGE
CONTROL
TURBINE GENERATOR LOAD CONTROL
EXPERIMENTAL STEAM CONTROL

Fig. 41

Recorder Panel and Control Console

All control rods are scrammed into the reactor automatically under any of 15 scram conditions. In addition, three rod-drop buttons are provided for testing of control-rod operation. One button operates the center rod only. The other 2 buttons operate with the intermediate and outside rod-selector switches to drop the rod or the particular group selected. Operation of a scram or drop button interrupts the latch-solenoid circuit, causing the rod to move to the full "in" position under the forces of gravity, reactor pressure, and the scram spring. The latch solenoids are 15-volt, dc-operated, and are supplied by a rectifier taking power from the 120-volt ac system.

(2) Position Indication

The control-rod position is indicated by a selsyn transmitter and receiver with a mechanical counter read-out for each rod. The system will provide for reading the rod position to the nearest 0.01 in. The transmitting selsyn is geared to the drive lead screw. Therefore, the control-rod position and the drive position are the same only when the rod is latched to the drive lead screw.

Three position-indicating lights are operated from relays which, in turn, are operated from limit switches. These lights indicate when the rod is in the full "in" position, the full "out" position, and when the rod is latched to the drive.

(3) Scram System

Table XIV shows a list of the conditions and their operating points which cause a scram. An annunciator is provided to give visual indication of a scram. Manual reset is required.

Table XIV

SCRAM CONDITIONS

1. High Flux No. 1	125% of operating power or power not to exceed burnout heat flux.
2. High Flux No. 2	125% of operating power or power not to exceed burnout heat flux.
3. Short Period	Less than 5 sec.
4. Low Reactor Water Level	Variable set, (5 to 15 ft) above bottom of reactor vessel
5. High Reactor Water Level	Variable set, (5 to 15 ft) above bottom of reactor vessel

Table XIV (Cont'd.)

6. Low Main-Steam Flow, lb/hr	2000
7. Low Forced-convection Flow	80% of operating flow
8. High Superheater Fuel Temperature, °F*	1300
9. High Condenser Pressure, psig	2
10. Superheater Flood Valves (Manual and Solenoid-Trip), Open	-
11. Low Air Pressure, psig	100
12. High Reactor Pressure, psig	640

The following conditions cause a scram, but do not operate an alarm annunciator:

13. Manual Scram (Control Console)
14. Manual Scram (Reactor Building)
15. Loss of Electrical Power

---

*This item is developmental and may not be in the reactor.

There are 5 key switches (numbers 2 to 6) in addition to the reactor-on key switch which can be used for bypassing scrams or interlocks during certain periods of operation.

Key switch No. 2 is provided for bypassing the short-period scram during normal operation after startup.

Key switch No. 3 is provided for bypassing the low steam-flow scram during startup and shutdown. When this scram is bypassed, the superheater-vent valve is opened automatically and remains open unless the scram circuit is reset. This interlock is provided to assure cooling of the superheater fuel elements. The automatic-manual selector switch permits the operator to adjust the amount of vent valve opening.

Key switch No. 4 is provided for bypassing the superheater-flood-valve-open scram. When the reactor is being heated with the superheater flooded, this scram must be inoperative.

Key switch No. 5 is provided for bypassing the low forced-convection-flow scram. When critical experiments or natural-circulation tests are being performed, it is necessary to operate without forced-convection flow. To test the effects of varying the forced-convection flow rate, the scram operating point can be lowered.

Three permissive interlocks are provided in the scram circuit. These interlocks, which must be satisfied in order to reset the scram circuit, are listed as follows:

- (a) All control rods must be in the down position before the scram circuit can be reset.
- (b) Water must be flowing in the auxiliary water system or the forced-convection system before the scram circuit can be reset.
- (c) A temperature-difference interlock prevents resetting the scram circuit when the difference between reactor water temperature and forced-convection water temperature exceeds 15°F.

When the forced-convection system piping is disconnected from the reactor vessel, the latter two interlocks are not needed and may be bypassed by key switch No. 6.

To prevent inadvertent flooding of the superheater, the superheater flood valve can only be opened by a switch which has a spring-loaded guard. This switch is effective only when the scram circuit is de-energized (i.e., control rods are unlatched to drop rods into the reactor).

(4) Alarm System

An alarm and annunciator system is provided in the control room to monitor the reactor and process system. All alarms initiate a flashing visual and an audible signal. Acknowledging the alarm silences the audible signal and retains a steady visual indication. Clearing of the alarm condition causes the visual indication to change from bright to dim; pressing the "acknowledge" button removes the visual signal. All alarm circuits are supplied by the emergency power system. A list of alarms is given in Table XV.

Table XV

ALARM CONDITIONS

1. High Reactor Water Level*	Variable set (2 in. below scram setting)
2. Low Reactor Water Level*	Variable set (2 in. above scram setting)
3. High Reactor Pressure, psig*	625
4. Low Reactor Pressure*	15% below operating
5. Low Condenser Vacuum, in. Hg	15
6. High Shield Temperature, °F*	200
7. Low Main-Steam Flow, lb/hr*	4000
8. High Steam-to-Turbine Pressure, psig*	360
9. High Steam-to-Turbine Temperature, °F*	595
10. Low Control-Air Pressure, psig	125
11. High Steam Temperature, °F*	875
12. High Reactor-Demineralizer Water Temp., °F	120
13. Low Boron Tank Level, in.	46
14. High Forced-Convection-Pump Bearing Temp., °F*	150
15. High Auxiliary-Water-Pump Bearing Temp., °F*	150
16. Low Makeup-Water-Storage Level, in.**	36
17. Low Feedwater Pressure, psig	650
18. High Feedwater-Storage-Tank Level, in.**	60
19. Low Feedwater-Storage-Tank Level, in.**	36
20. Low Turbine-Oil Pressure, psig	7
21. Low Turbine Gland-Water Level, in.	12
22. Condenser Circulating Pump, Off**	-
23. Condensate Pump No. 1, Off**	-
24. Condensate Pump No. 2, Off**	-
25. Cooling Tower Fan, Off**	-
26. High Air Activity, Turbine Building*	***

Table XV (Cont'd.)

27. High Air Activity, Reactor Building*	***
28. High Reading, Fission Break Monitor*	***
29. High Area Activity,*	***
30. High Reading Air Ejector Exhaust (gas monitor)*	***
31. High Reading Air Ejector Exhaust (particulate monitor)*	***
32. High Superheat-Fuel Temperature, °F*	1250
33. Low Forced-Convection Flow*	85% of operating level
34. Low Auxiliary-Water-Pump Flow, gpm*	100
35. Turbine Throttle, Tripped	-
36. Chamber High-Voltage, Off for Safety, Period and Linear Circuits**	-
37. High Condensate Demineralizer Temp., °F*	100
38. High Temperature Differential, Forced- Convection Water vs. Reactor Water, °F*	15
39. Low Boron Tank Temperature, °F*	170 Min. or 10 below operating temp.
40. Low Reactor Water Quality, Megohm*	0.5
41-42. Spares	

---

*These circuits do not alarm when electric power to the sensing instrument fails.

**These circuits require electric power to the sensing instrument and alarm on power failure. The other circuits do not require power to the sensing instrument and alarm on closing of a contact.

***Set points to be determined by operating experience.

Note: A red alarm light located on the control console adjacent to the superheater-vent-valve control is "on" when this vent valve is in manual control.

d. Nuclear Instrumentation and Radiation Monitoring

(1) Nuclear Operating Instrumentation

A diagram of the reactor operating nuclear instrumentation for BORAX V is shown in Fig. 42. Instrument holes in the shielding

wall between the access shaft and the reactor pit are shown in Fig. 45 (p.109). Cooling coils are attached to the outside of the instrument-hole liners. The operational ranges of the various nuclear instruments are shown in Fig. 43.

Fig. 42

Nuclear Instrumentation

Fig. 43

Nuclear Instrumentation Operating Ranges

The nuclear operating-instrument circuits are listed as follows:

(a) Safety Circuits (High-flux Scram) (Two)

The safety circuits are ANL Model CD104 Safety Trip Circuits. They have a range from 10^{-8} to 10^{-4} amp in decade steps, and there is a trip circuit which is adjustable over a decade range. Outputs are provided to the control console for remote indication, range selection, and trip-point adjustment.

(b) Period Meter (One)

The period meter is an ANL Model ICD-6. It has a log current range of 10^{-11} to 10^{-3} amp, a period range from ∞ to ± 5 sec, and a positive-period trip at 5 sec. Outputs are provided to the recorder panel for remote log-current indication and recording, and remote period indication.

(c) Startup Circuits (Two)

The startup circuits are BF_3 proportional-counter circuits consisting of two pulse amplifiers, two scalers, and a dual-channel log-count-rate-meter. The log-count-rate-meter has a range of 10 to 10^5 counts/sec. Startup detectors will be located to give a suitable count rate at source strength.

The startup circuits will be used with fission counters as loading detectors during fuel-loading operations.

(d) Linear Operating Instrument (One)

The linear operating instrument is an ANL Model IA 8 Linear dc amplifier. It has a range from 10^{-12} to 10^{-4} amp in decade steps. Outputs are provided to the control console for remote indication, recording, and range selection.

(2) Radiation Monitoring System

A diagram of the radiation monitoring system for BORAX V is shown in Fig. 44. The system gives warning of any abnormal release of radioactivity, including that from a fuel-element rupture. It is discussed in detail in the following:

(a) Area Monitor (One)

The area monitor is a six-channel, gamma-sensitive monitor, each channel having a range from 0.01 mr/hr to 10 r/hr. In the reactor building, the detectors are located on the main floor and in the equipment pit, access shaft, and subreactor room. In the turbine building, detectors are located in the basement and on the main floor. Outputs are provided from each channel for indication and alarm both locally and in the control building.

(b) Continuous Air Monitors (Two)

A continuous air monitor of the moving-filter type is located on the main floors of the reactor and turbine buildings. Each provides an output locally and to the control building for high air-particulate activity alarm.

(c) Fission Break Monitors

The fission break monitor is provided to monitor a condensed steam sample for high-power operation, and the reactor water for low-power operation. The detecting system consists of a scintillation detector, pulse amplifier, single-channel pulse-height analyzer, and log count-rate-meter. The water passes through a system of small ion exchange columns, one of which has the detector mounted on it. Output from the system provides indication, recording, and high-level alarm in the control building.

Fig. 44

Radiation Monitors

(d) Air-ejector Exhaust Monitor

During turbine operation an airborne particulate monitor of the moving-filter type monitors the particulate activity in the condenser air-ejector system. A gas chamber is used to monitor gaseous activity. The two detector-unit outputs supply separate recorders located in the turbine building and a single alarm in the control building.

10. In-core Instrumentation: Developmenta. In-core Measurement Desired

An intensive effort is being made to develop good in-core instrumentation for BORAX V. The instrumentation discussed here is developmental and experimental, and is not required for operation. To provide operating and proof-of-design types of information, it is intended to measure the following variables:

(1) Flow Rates

- (a) Boiling Assembly Entrance
- (b) Boiling Assembly Exit
- (c) Superheater Assembly

(2) Temperatures

- (a) Fuel Temperature of Boiling Rod Center
- (b) Water-Channel Temperatures in Boiling Assemblies
- (c) Superheater Fuel Plate Temperature
- (d) Superheated Steam Inlet and Exit Temperatures
- (e) Saturated Steam Temperatures in Steam Dome
- (f) Reactor-Water Subcooling
- (g) Reactor-Vessel-Specimen Temperatures
- (h) Reactor-Water Temperatures

(3) Steam Void Ratio at Boiling Assembly Exit

(4) Neutron Flux

- (a) Axial and Radial Distributions
- (b) Absolute Magnitudes

(5) Reactor Vessel Pressure

b. Use of In-core Data

(1) Operating Information and Protection

An attempt will be made to monitor selected superheater fuel-plate temperatures on a periodic basis. A system, which includes an alarm light for each point monitored, allows selection of any point for continuous recording and scram protection to provide the reactor operator with an additional aid during startup and shutdown.

(2) Design Information

Through adequate measurements of the previously listed variables, it should be possible to determine the following steady-state and transient information:

- (a) Power generation per fuel assembly and at various locations.
- (b) Flow profile within the core as a function of power, pressure, location, etc.
- (c) Boiling fuel-rod central temperature and temperature response to power transients.
- (d) Steam void ratio at boiling fuel assembly exit as a function of power, saturation pressure, location, etc.
- (e) Effective boiling length of boiling fuel assembly for various functions of power, pressure, location, etc.
- (f) Recirculation ratio as a function of power, pressure, riser height, core size, etc.
- (g) Subcooling of core inlet water for various conditions of reactor operation.
- (h) Steam temperature.
- (i) Neutron-flux distribution within the core for given steady-state power levels.
- (j) Absolute neutron flux at specific locations as a function of feedwater temperature, flow, core size, power, etc.

Overall instrument system performance is aimed at a minimum (3 decibels) sine-wave response up to 2 cps for flow, temperature, and void devices. Ionization chamber systems are designed to follow the maximum anticipated frequency of 10 cps.

c. Description of Proposed Instrumentation

It is intended to install the instrumentation in two major divisions: that which is attached to two removable boiling fuel assemblies and two removable superheater fuel assemblies, and that which is attached to core structures. The instrumentation in the removable fuel assemblies should allow determination of data at different core locations by relocating the instrumented assemblies. The fixed instrumentation within the vessel is for measurement of data at fixed locations.

All instrumentation is installed so electrical and instrument taps necessary for operation terminate at a junction box outside the reactor vessel. The junction box is attached to the fuel assembly by a hollow steel tube which contains individual pressure tubes and metal-sheathed electrical conductors. This connecting tube offers some support for handling of the fuel assembly, and the individual tubes and electrical leads are brazed into the connecting tube to form a pressure seal. The connecting tube is welded to the junction box, and the final instrument nozzle closure is by means of a special flange and packing ring which surrounds the junction box.

Connections from the junction box to external recording or indicating instruments are through Amphenol-type water-resistant connectors. This arrangement allows handling of the instrumented fuel assemblies in a manner similar to the handling of standard fuel assemblies, i.e., it allows removal or replacement of the vessel head over a junction box or moving the assembly in or out of the vessel with the head in place.

The following describes the instrumentation in terms of the above divisions and types of instruments to be used:

(1) Boiling Fuel Assembly Instrumentation

(a) Flow rates

Special devices are necessary to measure the flow rates through these assemblies. A modified type of a commercially available turbine-type flowmeter has been selected.

(b) Void Ratio at Exit

The entrance and exit flowmeter technique of exit void concentration determination has been selected for use in the reactor. Preliminary testing, using a turbine meter in an air-water loop with void calibration by the gamma-ray-attenuation technique has indicated this method should be workable.

(c) Fuel Temperature

To obtain temperatures of the radial center of selected boiling fuel rods, it is planned to install thermocouples down the axes of the rods to cover the calculated axial hot spot regions. The thermocouple design, now undergoing final test evaluation, is based on the use of tungsten vs. 74% tungsten-26% rhenium as the couple, with beryllium oxide as the insulating material. The sheath is 0.062-in. nominal diameter tantalum in the fuel area and 0.062 in. nominal diameter Type 304 stainless steel in the water region. Insulation in this latter region is high-purity aluminum oxide. The selected fuel rods are partially assembled with hollow fuel pellets to allow insertion of the thermocouples. The assembly of fuel tube, thermocouple, and fuel pellets is seal-welded in an inert atmosphere and then leak tested. Individual thermocouple fuel rods are being designed to be removable and interchangeable with standard fuel rods. Thermocouple fuel rods are not designed for re-use of the thermocouple after removal of the rod from the assembly.

(d) Water-channel Temperatures

By using a vacant fuel rod space for installation of a thermocouple "rake," the axial distribution of water temperature is measured at intervals of approximately 2 in. The thermocouples are 0.062-in. OD, stainless steel-sheathed chromel-alumel of matched characteristics, aged to provide high stability.

(2) Superheater Assembly Instrumentation

The techniques described previously for pressure-sealing, lead terminations, etc., apply also to the superheater fuel assembly instruments. Two instrumented superheater assemblies and one spare are provided for each superheating core to make the following measurements:

(a) Flow Rates

Superheater assembly steam flow is measured by a venturi tube and external pressure transducer.

(b) Fuel-plate Temperatures

Stainless steel-sheathed, chromel-alumel thermocouples with flattened tips are assembled into the edges of selected fuel plates in the region of calculated axial hot spots to measure these temperatures. Brazing these thermocouples into the plates insures rapid response. This attachment technique has been successfully demonstrated. The 0.040-in. OD thermocouples are led through the static-steam-insulating spaces to their points of installation.

(3) Fixed Instruments

(a) Superheated Steam Temperatures

Stainless steel-sheathed, chromel-alumel thermocouples, 0.040-in. OD, are installed in each end of the instrumented superheater assemblies to measure steam temperatures. Extension leads are brought out of the vessel along with the fuel-plate thermocouples.

(b) Saturated Steam Temperature

To obtain a measurement of reactor vessel steam temperature, 2 thermocouples are installed as fixed instruments in the vessel steam dome. The thermocouples are chromel-alumel with $\frac{1}{8}$ -in. OD stainless steel-sheathed construction. Leads are brought through a vessel nozzle below the flange.

(c) Reactor Water Thermocouple

Two thermocouples are required to measure the temperature of the reactor water. One thermocouple is connected to the selector switch for (TR-2) and is used to observe reactor temperature during reactor startup and during critical experiments. The second thermocouple is required for the temperature-difference interlock of the forced-convection system. A spare thermocouple is also provided at the same location. These thermocouples are located in the downcomer inside the reactor vessel, are positioned vertically at the core centerline, and are made of chromel-alumel with $\frac{1}{8}$ -in. OD stainless steel sheaths.

(d) Subcooling

To determine subcooling of the water in the reactor vessel, 2 thermopiles are installed in one assembly to sense the temperature differential between the upper vessel region and core inlet region.

For integrity and accuracy, these thermopiles are fabricated using multiple stainless steel-sheathed, chromel-alumel thermocouples, $\frac{1}{8}$ -in. OD, assembled into a stainless steel tube-type housing. Leads are brought out through a nozzle below the vessel closure flange and terminated in a conventional thermocouple connection head.

The subcooling temperatures will be used along with saturated-steam temperature, feedwater flow, and feedwater temperature to obtain downcomer flow rate by a heat-balance technique.

(e) Reactor-vessel-specimens Temperature

Thermocouples are installed in three capsules mounted on the inside of the reactor vessel shell to monitor temperatures on reactor-vessel-material specimens.

(f) Neutron Flux

For zero-power and low-power detailed flux mapping, flux wires are inserted in the coolant channels of fuel assemblies, irradiated, and removed for counting.

For measurements of high-power flux distribution, pressure-tight, stainless-steel thimbles are installed through the top head nozzles into boiling fuel-rod locations at selected radial points. The neutron flux is measured by both commercially available miniature ionization chambers positioned by counter drives and by flux wires inserted in the thimbles manually.

(g) Reactor Vessel Pressure

A strain gage-type transducer is proposed for measurement of pressure transients within the vessel.

11. Fuel and Radioactive Material Handling and Storage

a. Description of Major Tools and Fixtures

The fuel and radioactive material-handling system may be divided into three subsystems: (1) loading of the boiling fuel assemblies with the reactor vessel head installed; (2) loading of the boiling and super-heater fuel assemblies, miscellaneous components, and core structures with the vessel head removed; and (3) the storage of fuel and core structure components.

A description of the major tools and fixtures is as follows:

(1) The boiling fuel-assembly grappler, which can operate through any of the five fuel-handling nozzles in the top head, is the most complex tool of the fuel-handling system. The grappler assembly consists of a telescoping main column and an extendable arm with a solenoid-operated grappler fixed to it. A telescoping column is required because of the 15-ft lift-height limit of the reactor building crane.

The grappler is operated by a push-pull type of solenoid attached to a plunger which forces the fingers of the grappler out on the down stroke. The fingers engage the fuel assembly box through two

counter-bored holes in the top. Ears on the end of the fingers prevent the fuel assembly from slipping. In the event of power failure, the grapple cannot become disengaged from the fuel assembly, since the plunger is held in the "down" position by gravity and the fingers cannot be retracted. A deliberate act is needed to disengage the fuel assembly from the grapple once it has latched the fuel assembly. To release the grapple, the fuel assembly must be set down and the plunger pulled up.

The extendible arm is essentially a three-bar linkage that is driven by a ball screw and nut. The ball screw is driven by means of a telescoping shaft down through the center of the telescoping column. By means of the extendible arm, the grapple can engage any boiling fuel assembly up to a radius of ten in. from the centerline of a fuel-handling nozzle in the reactor vessel top head. The grapple head can be rotated independently of the extension arm by means of cables, and the maximum rotation needed for the grapple head is 45°. The entire tool is rotated by means of a motor and bearing mounted on the coffin.

(2) The coffin is a modified existing coffin that was used in earlier BORAX experiments. The coffin has a pair of gear-operated bottom doors, a natural uranium section that provides shielding around the fuel portion of the fuel assemblies, a stainless steel pipe riser, and a support for the mounting of the grapple, winch, bearing, motor, and the control for the boiling fuel grapple. A frame which rests on the upper ledge in the reactor pit supports the coffin over the fuel-handling nozzles. The frame is positioned by dowel pins in the ledges and has a carriage for the coffin which is also located by dowel pins.

(3) A grapple is also provided to load boiling assemblies with the reactor head removed. This consists of a grapple head, as described previously, attached to cables that are supported by a winch on the coffin. The grapple is lowered into the vessel and positioned by the coffin and a long rod to engage the boiling fuel assembly.

(4) The superheater fuel-assembly grapple operates in a similar manner to that described above. Fingers engage a horizontal pin in the riser of the fuel assembly, and a plunger attached to the solenoid unlatches the fingers on the down stroke and latches the fingers on the up stroke. The grapple is supported by a winch on the coffin and is positioned by means of long rods and the coffin.

(5) To provide adequate shielding when needed, an extension spool can be sealed to the reactor flange with the vessel head removed. This enables the reactor to be flooded to a height equal to that of the fuel-handling nozzles.

(6) A working floor is supported on the lead shields in the reactor pit around the reactor vessel. This floor is slightly above the elevation of the reactor flange and furnishes a working area adjacent to the fuel nozzles or extension spool.

(7) Miscellaneous tools, such as rods with hooks and socket wrenches, are used to handle components such as chimneys, hold-down boxes, hold-down latches, couplings, etc. A periscope and underwater lighting are provided to facilitate all fuel-handling operations.

Fixtures and tools are provided for reloading the three different core assemblies, flood and drain manifolds, etc., and for storing these components in the dry storage pit.

b. Storage Facilities

Storage facilities for fuel and other radioactive materials are provided as follows:

(1) The water-storage pit is used for storage of boiling fuel rods and assemblies and for underwater handling and servicing of other radioactive components, such as instrumented fuel assemblies, sources, and oscillating rods. The storage racks for boiling fuel assemblies and individual boiling fuel rods are lined with cadmium, and are so located and fastened down in the pit as to give a safe (noncritical) geometry. Underwater lights, fixtures and tools for underwater handling and servicing of other radioactive components, such as instrumented fuel assemblies, sources, and oscillating rods, are provided for the pit.

(2) A fuel and material-storage area is located in the old BORAX reactor vessel and pit. A cadmium-lined storage rack is located inside the old vessel for storage of superheating fuel assemblies, and large boxes are located in the remainder of the pit to store control rods, chimneys, hold-down boxes, and other radioactive components. The space between the tubes and boxes is filled with gravel and capped with concrete. Concrete shielding slabs are used to cover this area.

(3) A dry-storage pit with concrete slab covers is located in the southwest corner of the reactor building. This pit is used to store core structures, forced-convection baffle, feedwater sparger, control rods, hold-down boxes, chimneys, and other large, radioactive items. The radioactive items are transferred from the reactor pit to the dry-storage pit by use of the building crane, which can be operated from a remote control station located against the west wall of the reactor building. The remote control station can be shielded by temporary concrete slabs if necessary.

c. Handling of Boiling Fuel Assemblies

Handling of fuel assemblies in the boiling core is normally done through the five $5\frac{3}{4}$ in. ID nozzles in the top of the reactor head. The only exception to this is the removal of the instrumented boiling assemblies, which must be loaded with the vessel head removed.

Nonradioactive boiling fuel assemblies are handled in the same manner as the radioactive assemblies when the vessel head is on. When the vessel head is removed, the "head-off" grapples are used, and the coffin is unnecessary.

The reloading of radioactive boiling fuel assemblies with head off is accomplished by using the coffin with hoist, boiling fuel-assembly grapple, extension spool, underwater lights, periscope, long-handled socket wrenches, and long-handled hooks.

The reloading procedure for a boiling fuel assembly with the reactor vessel head in place is as follows:

(1) Depressurize the reactor vessel and flood to within 8 in. of the top of the nozzles. Remove the concrete shield slabs and nozzle blind flanges as required.

(2) Place the coffin-support frame in the reactor pit, locating the frame in the desired location with dowel pins in the ledge. Place the coffin in the support frame and lower the boiling fuel-assembly grapple through the nozzle. Insert the underwater lighting, periscope, and hand tools through the other nozzles.

(3) Using the long-handled hook, lift the chimney and set it aside on the other chimneys; operate the hold-down latches with the socket wrenches; and lift the hold-down box and set it aside on top of the chimneys.

(4) Lower the boiling fuel-assembly grapple to a height just above the shroud. Extend the grapple arm and rotate the entire tool until the grapple is above the boiling fuel assembly. Lower the grapple into the fuel box and energize the solenoid to engage the fuel assembly. Hoist the fuel assembly to a height above the shroud so that the fuel assembly will clear the shroud when the grapple arm is retracted. Retract the grapple arm until it is in line with the telescoping portion of the tool. Hoist the fuel assembly into the coffin and close the bottom coffin doors.

(5) Using the building crane, transport the coffin into the water-storage pit, open the coffin door, and lower the fuel assembly into the storage rack in the pit. Disengage the grapple and remove the coffin.

(6) Fuel rods from the boiling fuel assemblies may be stored individually in the water-storage pit. They are removed and handled under water by means of fuel-rod manipulators, using underwater lights and a working bridge spanning the water-storage pit.

(7) Boiling fuel assembly loading procedure is the reverse of the above.

d. Handling of Superheater Fuel Assemblies

Handling of the superheater fuel assemblies, instrumented and noninstrumented, is done with the vessel top head removed. Nonradioactive superheater fuel assemblies are set in place with the building crane. The reloading of radioactive superheater fuel assemblies is done using the coffin with hoist, superheater fuel-assembly grapples, underwater lights, extension spool, and miscellaneous long-handled tools.

The proposed reloading procedure for a superheater fuel assembly is as follows:

(1) Depressurize and flood the reactor vessel to a level just below the main vessel flange. Remove the shielding slabs, insulation over the head bolts, nuts, washers, and reactor vessel head. Place the extension spool on the reactor flange and flood the extension spool.

(2) Disconnect the one-bolt couplings at the top of the superheater fuel-assembly risers, and bend and clip the flexible manifolds out of the way against the wall. Lift off the hold-down plate with bearing rings and the Belleville spring.

(3) Using the building crane, center the coffin over a superheater fuel assembly. Lower the grapple, guided by a long rod, engage the superheater fuel assembly, hoist into the coffin, and close the coffin doors. With the building crane, transport the coffin and superheater fuel assembly to storage.

(4) Superheater fuel-assembly loading procedure is the reverse of the above.

e. Handling of Other Reactor Components

The reactor vessel head must also be removed to lift the following large reactor or experimental components from the vessel: chimneys, hold-down boxes, forced-convection baffle, feedwater sparger, core structures, control rods, oscillating rod, and certain in-core instrumentation. The Sb-Be source and some instrumentation can be handled through nozzles in the reactor vessel head.

12. Biological Shielding

a. Description

The biological shielding for the BORAX V reactor is shown in Fig. 45. The main floor of the reactor building is accessible during operation because of protection by a total thickness of 5 ft 2 in. of high-density (215 lb/ft^3) concrete slabs covering the reactor pit, 2 ft of ordinary concrete covering the access shaft, and 1-ft-thick concrete covers over the piping trenches, equipment pit, and dry-storage pit. The access shaft and subreactor room are only partially shielded and are not accessible during operation. The reactor pit is shielded horizontally by $4\frac{1}{2}$ ft of concrete on the access shaft side and by earth on the other three sides. Between the floor of the reactor pit and the ceiling of the subreactor room is a 3-ft-thick slab of high-density concrete. Penetrations into the reactor pit are offset or stepped as required. The piping trench between the reactor building and the turbine building is covered with 6 in. of concrete and 3 ft of earth.

b. Shielding Design

This section is divided into two parts: shielding requirements at power, and shielding requirements for access to reactor components after shutdown. The calculations for the first part are based on a 40-Mw reactor operating full time, with personnel being exposed to a maximum of radiation 7.5 mr/hr , the tolerance for a 40-hr week. With respect to shutdown conditions, the reactor is assumed to have operated long enough to be in near equilibrium with its fission products. Higher short-time exposures might be expected in performing work on the reactor after shutdown. In both cases extreme values are used; hence, the shielding is conservative. Three types of radiation are considered; fast neutron ($> 1 \text{ Mev}$), thermal neutron, and gamma ray. The gammas are further divided into prompt fission gamma, fission product gammas, and capture gammas formed by the absorption of thermal neutrons in the shielding medium.

Additional shielding provisions will be necessary if experiments are conducted with a separate superheating core in the upper end of the reactor vessel. In areas where high-density concrete is specified, the calculations were made on the basis of magnetite-type concrete (density of 215 lb/ft^3).

(1) Shielding at Power

(a) Top of Reactor Pit

The most severe shielding problem occurs when the reactor core is loaded with a central superheater.

Fig. 45

Reactor Building Floor Plan and Elevations

This permits open radiation channels through the water above the core. The top shield was sized for this condition.

Thermal Neutron Flux. A 1-in.-thick, 1% boron-steel plate inside the head reduces to a negligible amount the thermal neutron flux entering the shield. The effect of that neutron flux which has been thermalized in the shield can be neglected because it is small with respect to the flux of fast neutrons and gamma rays.

Fast Neutron Flux. The fast neutron flux is high because of the lack of moderator in the superheater channels. A fast neutron flux of $3.7 \times 10^{10} \text{ n}/(\text{cm}^2)(\text{sec})$ from a geometrically reduced plane source was considered incident upon the shielding. Sixty-two in. of high-density concrete reduces the fast neutron flux dose to less than 4 mrem/hr.

Gamma Flux. The prompt fission gammas were divided into 8 energy groups from 0.5 to 8 Mev. The prompt-gamma dose was then calculated by considering each group as a plane source at the reactor shield and summing the individual effects. The secondary gammas resulting from neutron captures in the shield were determined by assuming an exponential source of gammas which followed the fast neutron flux. The final shield thickness of 62 in. of high-density concrete was determined principally by this last form of radiation.

(b) Access Shaft and Trenches

In determining the thickness of the cover over the access shaft and the placement of ducts and trenches, the practice was to assure that a shielding thickness at least equivalent to the number of attenuation lengths through the top shield be maintained in any straight-line direction from the reactor core.

(c) Activation of the Steam

Figure 45 shows the location within the building of the trenches containing the 6-in.-diameter main-steam line and the smaller superheated steam lines. Assuming no fission product release, the major activity to be considered is 6.2-Mev gamma rays from the 7.6-sec N^{16} . A specific activity in the steam line of $5.5 \times 10^6 \text{ Mev}/(\text{sec})(\text{g})$ of water was calculated for this isotope, provided all the N^{16} is entrained in the steam. The radiation level was then calculated with the steam pipe considered as a line source. One foot of concrete reduces this radiation level to below tolerance.

The condensate returned from the turbine building does not represent a hazard due to the holdup in the feedwater system and the short half-life of N^{16} .

(2) Shielding for Shutdown

Three areas were considered with respect to accessibility after shutdown: (a) the access shaft, (b) the subreactor room, and (c) the top of the reactor vessel. The numerous penetrations into the access pit and the subreactor room complicate the shielding calculation; therefore, the determination of shielding for direct radiation leakage is somewhat of a good guess. Since access to the reactor after shutdown is usually limited to short exposure times, levels of radiation higher than normal can be tolerated. For shutdown calculations, the radiation from the fission products was divided into 7 energy groups of gammas from 0.4 to 3.0 Mev. A delay of 2 to 24 hours after shutdown is recommended before access is permitted to the reactor vessel components.

(a) Access Shaft

The shielding wall between the reactor and access shaft is $4\frac{1}{2}$ ft of ordinary concrete and is augmented by the foot of water surrounding the core. All pipe penetrations into the reactor pit are offset by at least $1\frac{1}{2}$ diameters of the duct. Special plugs are designed for the instrument holes to reduce leakage through the shield.

Entry into the access shaft is prohibited during power operation.

(b) Subreactor Room

The shielding above the subreactor room is 3 ft of high-density concrete. All pipe penetrations are offset by $1\frac{1}{2}$ pipe diameters, and none of these pipes are directly in line with the core. To eliminate streaming along the control-rod drive and forced-convection nozzles a stepped bushing of steel is inserted, as shown in Figs. 35 and 45. The greatest leakage is through the large offset steam-line penetration, but the steam line can be filled with water after shutdown.

(c) Access to Top of the Reactor

Access above the reactor with the shield slabs removed requires flooding of the superheater channels with water and raising the level of the water to a depth of 8 ft above the core. As shown in Fig. 45, the radiation which is scattered up the area between the pit walls and the reactor vessel is shielded by a ring of 6-in. thick lead shielding blocks, mounted just below the flange of the reactor vessel.

c. Heat Generation in the Concrete Biological Shield

An estimation of the heat generated in the concrete biological shield was based on a consideration of the following sources of radiation heating:

- (1) Core gammas (both prompt and delayed).
- (2) Secondary gammas from neutron absorption in water.
- (3) Secondary gammas from neutron capture in the reactor vessel shell and intervening thicknesses of concrete wherever applicable.
- (4) Fast neutron moderation in the concrete.

A summary of results for radiation heating as a function of distance into the concrete at core centerline is given in Fig. 46. A similar summary as a function of elevation and along the floor of the reactor pit is given in Fig. 47. If the reactor pit shield is uncooled, the maximum temperature of the concrete at a reactor power of 40 Mw has been calculated to reach about 520°F. Since concrete deteriorates and spalls at this temperature, a shield-cooling system is provided, as shown in Fig. 45 and described in Section III, B, 10.

Fig. 46

Radiation Heating of Shield Versus
Distance Into Concrete

Fig. 47

Radiation Heating of Shield Along
Surface of Reactor Pit Wall

B. Process Piping Systems

1. Superheater Vent, Flood, and Drain Systems

a. Superheater Vent System

The superheater vent system insures that a sufficient flow of coolant steam is maintained through the superheater during reactor startup and shutdown operation to prevent overheating. This system consists of flowmeter FI-4 and a manual or automatic air-operated, Type 304 stainless steel superheater-vent valve (HIC-5AV) connected to the main-steam piping, as shown in Fig. 48. This system in the "vent" condition permits steam to flow from the steam dome of the reactor vessel, through the superheater, into the main-steam piping and then to atmosphere.

Valve operation is normally elective and adjustable for flows of from 0 to 8000 lb/hr. In event of a scram, the valve is opened automatically to a preset flow position calculated to provide adequate immediate cooling.

This is an essential system, and the valves and controls are therefore connected to the emergency power system.

Fig. 48

Superheater Startup and Shutdown System

b. Superheater Flood System

The superheater flood system is made up of superheater flood valve (HC-5V) (solenoid trip and air reset with back-up manual cable trip), a manual gate valve which parallels valve HC-5V, two air-operated pressure-equalizing valves (HC-17AV and HC-18AV), superheater water-level indicator (LI-2), and a piped connection to one of the reactor vessel forced convection inlet lines, all as shown in Fig. 48.

This system permits flooding of the superheater for additional cooling during the shutdown cycle. Calculations indicate that flooding will be accomplished in about one minute.

In the "flood" condition, the superheater vent valve is closed and the flood valve and two pressure-equalizing valves are open. The flood valve permits water to flow by gravity from the boiling section of the reactor vessel, through the forced-convection line, into the superheater. The superheater second-pass pressure-equalizing valve (HC-17AV) assures that pressure in the superheater second pass is the same as that in the boiler during flooding. Superheater level-indicator pressure-equalizing valve (HC-18AV) equalizes the pressure across level indicator LI-2, permitting a true superheater water-level indication when the flood valve is reclosed. The manual gate flood valve serves the same

purpose as the superheater flood valve in case of failure of the flood valve and is manipulated from the operating floor by means of an extension rod. In order that inadvertent flooding of the superheater during operation may be avoided, the manual cable operator for the solenoid-trip valve and the reach rod operator for the manual flood valve are secured with frangible padlocks. The material of construction is Type 304 stainless steel.

Since the superheater flood system is essential, the valves and controls are connected to the emergency power system.

c. Superheater Drain System

The superheater drain system consists of an air-operated superheater drain valve (HC-6AV), a superheater drain-flow indicator (FI-5), the previously mentioned pressure-equalizing valves and level indicator LI-2, and a piped connection to the turbine condenser, all as indicated on Figs. 48 and 49.

When the reactor has been brought to pressure, the flooded superheater must be drained prior to the start of steam flow. In the "drain" condition, the flood valve is closed, and the drain valve and two pressure-equalizing valves are open. The drain valve permits water to flow out of the superheater, at about 120 gpm, to the turbine condenser where it is cooled. The superheater pressure-equalizing valve supplies boiler steam at the exit of the superheater second pass, thus assuring proper blowing-down of the second pass. Drainage flow is indicated by flowmeter FI-5, and complete drainage is first evidenced by a change in indicated flow rate and may be confirmed by closing the superheater drain valve, and then observing the superheater level indicator (LI-2).

The material of construction is Type 304 stainless steel, except for the piping between the superheater drain valve and the condenser, which is chrome/moly steel ASTM A-335, Grade B-11.

2. Batch Feed System

The superheater requires a continued flow of steam after reactor shutdown for as long as the decay heat remains above 0.4 Mw. As a result, an assured means of making up evaporation losses is required in order to maintain an adequate reactor water level.

The batch emergency feedwater system is provided to fulfill this need in the event of power or feedpump failure. The system consists of a 350-gal polyvinyl chloride plastic-lined, steel batch-storage tank, and a 150-gal Type 304 stainless steel, batch-feed pressure tank, piped and valved as shown in Fig. 48. Piping and valves are Type 304 stainless steel. Both tanks are normally filled with demineralized water.

DEVICE IDENTIFICATION LEGEND
FOR
MAIN PROCESS FLOW AND INSTRUMENTATION

F1-3	LOW RATE-FEEDWATER FLOW INDICATOR	Pd-35	LOW FORCED-CONVECTION FLOW SCRAM	FC-2 = HC-10V	FORCED-CONVECTION PUMP-DISCHARGE VALVE
F1-4	SUPERHEATER-VENT FLOW INDICATOR	Pd-45	LOW STEAM FLOW SCRAM	FC-1 = HC-11V	FORCED CONVECTION PUMP SUCTION VALVE
F1-5	SUPERHEATER DRAIN FLOW INDICATOR	LA-18	LOW BORON TANK LEVEL ALARM	FW-14 = HC-14V	LOW RATE FEEDWATER CONTROL VALVE
F1-6	REACTOR CIRCULATING PUMP FLOW INDICATOR	LA-22	LOW DEMINERALIZED WATER-STORAGE LEVEL ALARM	MS-17 = HC-15AV	DESUPERHEATER ATOMIZING STEAM VALVE
FR-1	MAIN-STEAM FLOW RECORDER	LA-24	HIGH FEEDWATER STORAGE LEVEL ALARM	FW-33 = HC-16AV	SEAL WATER SUPPLY VALVE
FR-2	MAIN FEEDWATER-FLOW RECORDER	LA-25	LOW FEEDWATER-STORAGE-LEVEL ALARM	MS-5 = HC-17AV	SUPERHEATER SECOND PASS PRESSURE EQUALIZING VALVE
FR-3	FORCED CONVECTION PUMP FLOW RECORDER	PA-7	HIGH CONDENSER-PRESSURE SCRAM	MS-6 = HC-18AV	SUPERHEATER LEVEL INDICATOR PRESSURE EQUALIZING VALVE
FR-4	TURBINE BYPASS STEAM-FLOW RECORDER	PA-8	LOW CONDENSER-VACUUM ALARM	FW-38 = TRC-1V	DESUPERHEATER FEEDWATER-CONTROL VALVE
PI-4	REACTOR-VESSEL PRESSURE INDICATOR	PA-23	LOW FEEDWATER PRESSURE ALARM	PS-12,3,4	PRESSURE-RELIEF VALVES
PR-1	STEAM TO TURBINE PRESSURE RECORDER	TA-9	HIGH SHIELD WALL TEMPERATURE ALARM	AW-12 = FC-1AV	CIRCULATING PUMP SUCTION VALVE
PR-2	REACTOR VESSEL PRESSURE RECORDER	TA-14	HIGH STEAM TO TURBINE-TEMPERATURE ALARM	FW-39 = LC-2AV	FEEDWATER MAKE-UP CONTROL VALVE
TR-2	REACTOR WATER OR STEAM-TEMPERATURE RECORDER	PR-3	SUPERHEATER-PRESSURE-DROP RECORDER	FW-15 = LC-1V	MAIN FEEDWATER-CONTROL VALVE
TR-3	FEEDWATER-TEMPERATURE RECORDER	TA-17	HIGH ION EXCHANGE TEMPERATURE ALARM	MS-18 = PIC-1V	MAIN STEAM BACK PRESSURE CONTROL VALVE
TRC-1	STEAM TO TURBINE TEMPERATURE RECORDER	TA-20	HIGH FC PUMP BEARING TEMPERATURE ALARM	MS-14 = PIC-2V	VENT STEAM BACK PRESSURE CONTROL VALVE
LR-1	REACTOR WATER LEVEL RECORDER	TA-21	HIGH AUXILIARY PUMP BEARING TEMPERATURE ALARM	MS-10 = PIC-3V	EXPERIMENTAL STEAM BACK-PRESSURE CONTROL VALVE
LR-2	FEEDWATER-STORAGE-LEVEL RECORDER	MS-4 = HC-1AV	REACTOR-VENT VALVE	AW-2 = PIC-4AV	ION-EXCHANGE PRESSURE CONTROL VALVE
LR-3	DEMINERALIZED-WATER STORAGE LEVEL RECORDER	B-1 = HC-2AV	BORON TANK PRESSURE-EQUALIZING VALVE	PIC-5V	GLAND SEAL WATER PRESSURE CONTROL VALVE
LI-1	REACTOR-WATER-LEVEL INDICATOR	B-2 = HC-2BV	BORON-TO-REACTOR VALVE	MS-24 = PIC-9V	TURBINE BYPASS PRESSURE-CONTROL VALVE
LI-2	SUPERHEATER WATER-LEVEL INDICATOR	B-3 = HC-2CV	BORON SEAL VALVE	MS-25 = FC-10V	STEAM-TO CONDENSER DUMP VALVE
TV	TELEVISION MONITOR	B-4 = HC-2DV	BORON LEAKAGE VALVE	MS-13 = HC-5AV	SUPERHEATER VENT VALVE
LG-1	REACTOR WATER-LEVEL GLASS	FD-1 = HC-5V	SUPERHEATER FLOOD VALVE	PA-28,5	HIGH REACTOR PRESSURE SCRAM
PV-25	HIGH AND LOW REACTOR WATER LEVEL SCRAMS	FD-3 = HC-6AV	SUPERHEATER DRAIN VALVE	AE-9 = FC-2V	DILUTION AIR CONTROL VALVE
				RM	AIR EJECTOR EXHAUST MONITOR
				TI-1-1	PROCESS TEMPERATURE INDICATOR THERMOCOUPLES
				THRU	
				TI-1-11	

Fig. 49

Main Process and Instrumentation Systems

When water addition is to be made, reactor steam is manually admitted to the top of the 150-gal pressure tank, and the line from the tank drain to the reactor feed-water line is valved open. When the tank pressure is the same as reactor pressure, water flows by gravity into the reactor. After the tank has drained, it is manually isolated from the reactor, vented to atmosphere, refilled by gravity from the 350-gal storage tank, and the pressurizing and drain cycle is repeated. In this manner, makeup equivalent to about $2\frac{1}{2}$ ft of reactor water can be added in approximately one hour. If necessary, raw water can be added to the reactor vessel in this manner. Under extreme conditions, the contents of the boron-addition tank can also be fed into the reactor vessel.

3. Boron-addition System

The boron-addition system is a reserve means of poisoning the reactor to make it subcritical. It is a stainless steel system, consisting of a 270-gal pressure tank with two 12-kw immersion heaters and temperature controls, and an assembly of heated piping and valves, all arranged as shown in Fig. 49. The tank is normally charged with 130 kg H_3BO_3 , dissolved in 200 gal of demineralized water.

In the storage condition, the boron-tank pressure-equalizing trip valve (HC-2aV), boron-to-reactor valve (HC-2bV), and boron-seal valve (HC-2cV) are closed and the boron-leakage valve (HC-2dV) is open. In this condition, the tank is isolated from the reactor vessel and, with the boron-leakage valve open, chance leakage of boric acid into the reactor vessel is prevented. Each of these valves is a solenoid-trip type.

Poisoning is initiated at the control building by a single control which causes valve HC-2dV to trip closed and valves HC-2aV, HC-2bV, and HC-2cV to trip open, admitting reactor steam into the top of the tank and establishing a drain route from tank to reactor. After the tank has reached reactor pressure, the boric acid solution flows by gravity into the reactor vessel.

The boric acid tank will normally be maintained at elevated temperatures by the thermostatically controlled electric immersion heaters. One hundred and thirty kilograms of boric acid can be dissolved in 200 gal of water at 152°F . This amount of boric acid when added to the reactor will maintain each of the three reference loadings subcritical with all control rods out. The tank will be normally maintained at temperatures of 180°F or above continuously. An alarm is provided to continuously monitor this temperature. Samples will be taken from the tank weekly to chemically analyze the solution.

For special loadings higher concentrations of boric acid must be maintained in the storage tank so that when added to the large volume of reactor water there will be sufficient boric acid to shut the reactor down with all rods out.

Even though this is a slow-acting system, the valves and controls, because of their importance, are connected to the emergency power supply. In addition, each of the trip valves is equipped with an auxiliary manual trip. Since inadvertent poisoning creates a special problem, the electric control circuitry has been arranged to provide the best degree of protection against accidental dumping.

4. Forced-convection System

During experiments to investigate higher core power densities and coolant flow velocities, a forced-convection system will be utilized on BORAX V. This stainless steel system, shown on Figs. 45 and 49, includes a Worthington 10,000-gpm centrifugal pump, vertically mounted below the reactor in the subreactor room. The pump has a NPSH of 14 ft, a total discharge head of 193 ft, and is driven by a 450-hp squirrel-cage induction motor.

Forced-convection flow rates may be remotely regulated by means of motor-operated, forced-convection-pump discharge valve (HC-10V). A venturi-type flow-meter is provided, with remote indication in the reactor control room.

As shown on Fig. 35, the forced-convection outlet from the reactor vessel is through two 14-in. nozzles near the bottom of the down-comer area. These nozzles extend through the reactor pit floor slab into the subreactor room and are flanged to 14-in. piping which enters a common 16-in. pump-suction line. The pump discharge passes through a 14-in. line and is then equally distributed into the bottom of the reactor vessel through four 6-in., forced-convection, reactor inlet nozzles. A diffusion ring at this point breaks down the nozzle effects and equalizes flow distribution before the water enters the core.

The forced-convection baffle is installed in the reactor vessel during forced-convection experiments to provide the necessary division between vessel inlet and outlet flow paths.

5. Steam Systems

The BORAX V steam system, shown on Fig. 49, is unusually flexible. It is capable of removing saturated steam from the pressure vessel steam dome and superheated steam from nozzle outlets in both

the steam dome area and the bottom of the vessel. The steam produced can be utilized in the turbogenerator unit, bypassed to the condenser, and/or exhausted to the atmosphere, as necessary.

A steam main, completely entrenched from the reactor to the turbine building, is provided.

The reactor is operable at all pressures up to 600 psig. Normal turbine inlet steam conditions are 350 psig with a maximum allowable superheat of 150°F, or a temperature of 585°F. At reactor pressures between 350 and 600 psig, pressure upstream from the turbine must be reduced, and at steam temperatures above 585°F desuperheating is necessary. Below 250 psig, the turbine is inoperable, and steam flow must be bypassed directly to the condenser or to the atmosphere. Steam produced in excess of approximately 20 thermal Mw must be vented to atmosphere, since this is the maximum capacity of the existing turbine-generator-condenser system.

Materials of construction are equal to A-376, Type 304 stainless steel, A-335, grade P-11 chrome-moly steel and A-106, grade B carbon steel, depending on location and service conditions.

The major components of the BORAX V steam system are as follows:

(a) Main Steam Line

The main 6-in. steam line extends from the reactor pressure vessel through the pipe trench to the turbine building and turbogenerator unit, generally as shown in Fig. 49. A large expansion loop is located in the trench between buildings to allow for pipe expansion. Located along the main steam line are the following: two sets of dual safety valves, flow, pressure, and temperature-measuring instruments, a 3-ft removable test section, a calorimeter, sampling and sample injection lines, control valves, shut-off valves, and many branch lines to make a completely flexible steam system. During operation at saturation temperatures, steam flows from the top of the steam dome into the single, chimney-like steam collector (Fig. 35) and thence into the main steam line.

(b) Internal and External Superheat Piping

Both central and peripheral superheater fuel elements will be tested in BORAX V. For the two-pass reference design superheater, the steam produced is removed, as shown in Figs. 11 and 13, through stainless steel flexible tubing coupled to the tops of the individual fuel elements. The flexible tubes are fastened to lengths of solid, double-walled pipe

attached to the 3-in. superheated steam vessel outlet nozzles by quick-disconnect couplings. The four 3-in. superheated-steam outlet lines are joined in a common manifold connected to the main 6-in. steam line. Internal moisture separators are provided at the superheater fuel assembly inlets for the superheating cores.

Should single-pass superheat be produced, it can be removed through the reactor vessel forced-convection inlet nozzles via plenum chambers and piping attached to the core support plate. A 6-in. superheated-steam line then carries the steam from the subreactor room up through the reactor pit into the main 6-in. steam line in the pipe trench.

(c) Back-pressure Stations

The main-steam, back-pressure control valve (PIC-IV) located in the main-steam line is designed to operate as a remote, variable-set valve. It is required for flow of saturated or superheated steam at capacities up to 60,000 lb/hr at a set-point between 300 and 600 psig. This valve, sensitive to upstream pressure, opens only when reactor pressure is increased above the preset pressure. In this manner it serves as a reducing station and a reactor containment device.

At powers above 20 Mw, or when it is otherwise desirable that steam flow directly to the atmosphere, a vent line with vent-steam, back-pressure control valve (PIC-2V), upstream from the main-steam back-pressure control valve, is utilized. This valve and line are designed to permit flow of steam over a range from 0 to 60,000 lb/hr; it has a remotely adjustable, variable set-point between 100 and 600 psig, and it operates in the same manner as the main-steam, back-pressure control valve.

(d) Safety Valves and Vents

Inasmuch as the main-steam system is normally operated at two different pressures, two sets of dual-safety valves are required. The two valves for the reactor vessel and high-pressure steam line have settings of 650 and 670 psig, respectively, with a combined capacity of 120,000 lb/hr of saturated steam. They are exposed to full reactor pressure, and there are no isolating valves between them and the reactor vessel. The two safety valves for the low-pressure portion of the steam system have settings of 388 and 400 psig and also have a combined capacity of 120,000 lb/hr. All four discharge to the atmosphere through a 6-in. stack on the west side of the reactor building.

A 2-in. superheated-steam vent line with superheater-vent valve (HIC-5aV) is installed, as previously described, to insure that steam flow through the superheater continues on shutdown. An additional reactor-vent system allows purging and blowdown of reactor pressure via the reactor-vent valve (HC-1aV).

For independent, single-fuel assembly experiments, a separate flow path, flow measurement, and the experimental-steam back-pressure control valve (PIC-3V) are provided for steam flowing through two of the superheated steam outlet lines.

(e) Desuperheater Station

Turbine-temperature limitations require that a desuperheater (DE-1), as shown on Fig. 49, be installed in the main steam line to insure a maximum temperature of 585°F for the steam entering the turbine. A feedwater line is piped to the desuperheater station and steam is provided via the desuperheater atomizing-steam valve (HC-15aV) to improve feedwater atomization at low flows.

(f) Steam Bypass and Separation

The turbine-bypass, pressure-control valve (PIC-9V) can be set at the turbine building to control turbine throttle pressure. This valve bypasses steam directly to the condenser. Incremental changes in turbine load affect bypass flow rate, permitting reactor load to remain virtually constant.

A steam separator is located immediately upstream from the main turbine stop valve for quality improvement during saturated-steam operation.

(g) Turbine-generator-condenser Unit

The existing turbine-generator-condenser unit at the BORAX site is utilized, with a few minor modifications, as a heat sink for BORAX V.

The turbine-generator is a 1926 Westinghouse unit rated at 3750 kva. The main, water-cooled, surface condenser is located directly beneath and is supported from the turbine. A 6500-gpm Allis-Chalmers main-circulating pump is used to circulate cooling water from the condenser to the forced-draft, evaporative cooling tower. Return flow from the tower is by gravity. A parallel loop in the cooling-water system circulates water through a water rheostat which is used to apply electrical loads, as desired, to the main generator.

A set of two-stage air ejectors, operated from the main steam supply, is used to create the normal operating pressure of about 4 in. of Hg absolute in the main condenser.

Turbine gland-seal water is supplied from the gland-seal water-storage tank. Gland-seal water-control valve (PIC-5V) is installed in the main condensate-return line and maintains the necessary head to elevate condensate to the tank.

6. Air-ejector Exhaust System

A special air-ejector exhaust system is provided to remove the radioactive particles and noncondensable exhaust gases from the turbine-condenser air ejector and discharge them to the atmosphere. As shown in Fig. 49, this system contains auxiliary equipment with which to monitor, sample, remove moisture from, filter, measure flow rates of, and add dilution air to the exhaust gases.

The air-ejector condenser, as well as another heat exchanger and a demister, are provided to remove excessive water vapor from the noncondensable gases. The gases then flow through two filters: the first an absolute filter to remove particulates; the second, a charcoal filter, will remove any iodine fission products. Flow of the gases then passes through a rotameter to a 150-cfm blower. Also connected to the blower suction is a 4-in. dilution air line from the outside atmosphere. This line contains a dust filter, a flow orifice and a fail-safe, quick-closing valve which closes if the blower fails to insure continued expulsion of the air-ejector gases out of the elevated discharge stack. The entire system is designed with a minimum pressure drop to allow continued operation, minus the dilution air, in case of blower failure.

The 150-cfm blower serves a dual purpose. First, it maintains a slight vacuum on the filters, preventing out-leakage into the turbine building, and, second, it supplies about 140 cfm of dilution air to mix with the approximately 10 cfm of radioactive gases from the air ejector. This serves to dilute the short-lived radioactive gases which are discharged to the atmosphere and insures a noninflammable, hydrogen-air mixture in the exhaust piping. A stack gas activity monitor and several sampling points monitor the normal operating activity level as well as give an indication of any increases in activity level. Three system-low-point drains insure drainage of all condensate to the drain-collecting tank.

7. Condensate and Feedwater System

The condensate return and feedwater system is seen in Fig. 49. Two condensate pumps and one condensate booster pump remove condensate from the condenser hotwell and pump it back to the reactor building via a condensate-return line located in the pipe trench. After passing in series through a condensate filter, a full-flow, mixed-bed condensate demineralizer, and a filter to catch resin fines, the condensate enters a

stainless steel feedwater-storage tank, which is located on the upper level of the equipment pit. The demineralizer may be bypassed if desired. Two 150-gpm centrifugal feedwater pumps, located on the lower level of the equipment pit, and taking suction from the storage tank, discharge into a common main-feedwater line. The range of reactor power is normally from 0 to 20 Mw, with a maximum capacity of 40 Mw for short durations. This requirement, together with the needs of the reactor water demineralizer, desuperheater, and seal systems, establishes the need for two 150-gpm feed pumps.

A feedwater-regulating system utilizing automatic and/or manual control permits maintenance of reactor water at desired levels over a wide range of powers. The main-feedwater-control valve (LC-1V) is required to pass the design flow of 275 gpm. In parallel to the normal main feedwater control valve is a motor-operated, low-rate, feedwater-control valve (HC-14V) to provide precision manual control at low flow rates (from 0 to 30 gpm).

The feedwater line downstream from the regulating valves is piped so that there are three alternate flow paths for the water to enter the reactor vessel. When the proper valves are manipulated, feedwater may enter the vessel through the internal feedwater sparger ring, the forced-convection inlet piping, or the forced-convection outlet piping.

Feedwater is also supplied to the steam desuperheater station DE-1. The desuperheater, located in the pipe trench on the west side of the reactor building, is equipped with an automatic control valve (TRC-1V) which is capable of passing a maximum of 15 gpm of desuperheating feedwater. This results in a reduction of steam temperature from 850°F to 585°F, for a maximum of 60,000 lb/hr of 600-psig superheated steam.

The reactor seal-water system also originates at the feedwater regulating-valve header. This 1-in. line is piped into the subreactor room, supplying seal water to the forced-convection and auxiliary pump glands and the control-rod-drive seals. Flow to the control-rod seals first passes through a filter, through a seal-water supply valve (HC-16aV), and then through individual rod-seal, inlet and outlet valves and rotameters, thus allowing accurate control and monitoring of seal water for the control rod drives.

Reactor leakage from the control-rod seals is piped to the subreactor room sump, while the return from the forced-convection and auxiliary pump seals, approximately 10 gpm, is returned to the feedwater storage tank.

8. Auxiliary-Water System

The auxiliary-water system makes use of the 150-gpm auxiliary-water pump to circulate reactor water through the reactor preheat or demineralizer systems, as shown in Fig. 49 and described below.

a. Reactor Preheat System

The reactor preheat system is composed of a bank of immersion-type electric heaters with a total rated capacity of 144 kw, a 150-gpm auxiliary-water pump, and suitable piping, valves, and equipment to make a complete system, located in the subreactor room.

This system is used during startup operations or whenever it is desirable to heat the reactor water without nuclear heat. It is especially useful during initial plant startup since it allows plant checkout and hot-critical experiments to take place without undue irradiation problems.

A second function of the system is to heat uniformly and simultaneously the reactor vessel and forced-convection piping, when necessary, to prevent excessive stresses due to differential expansion and the possibility of a cold-water accident. With the forced-convection piping in place, the auxiliary-water-pump suction line is connected to the forced-convection-pump suction piping immediately upstream from the closed pump suction valve. The auxiliary-water-pump discharge passes through the heater bank and flows back into the forced-convection-pump suction line immediately downstream from the pump-suction valve. Reactor water then passes through the forced-convection pump and returns to the reactor vessel via the reactor forced-convection inlet piping. This arrangement insures flow and temperature equalization throughout the system.

When the forced-convection piping is not in place, it is necessary to make use of the spare, 2-in. preheat-water line and one of the reactor vessel forced-convection inlet nozzles as connecting points for the auxiliary-water-pump suction and discharge lines.

A third function of the auxiliary-water pump is to move reactor water through the reactor demineralizer system as described below.

b. Reactor Demineralizer System

The reactor demineralizer system is designed to clean reactor water continuously, with the reactor operating or shut down, at a range of flow rates from 0 to 20 gpm. This system is required to maintain reactor water quality, pH, and low chloride concentration within very close limits.

Another function of this system is removal, when necessary, of boric acid from reactor water after the acid concentration has been reduced by dilution.

With the reactor shut down or at very low pressures, it is necessary to use the auxiliary-water pump to force reactor water through the demineralizer system. Whenever reactor pressure is greater than about 50 psig, demineralizer flow is maintained by reactor pressure alone.

When the auxiliary-water pump is used, the demineralizer supply water flows from a branch line at the pump discharge to heat exchangers located on the upper level of the equipment pit. A globe valve at the heat exchanger inlet is used for throttling purposes to adjust the flow rate.

When reactor pressure is available to create demineralizer supply flow, water leaves the reactor through the 1-in. reactor-blowdown line and flows to the heat exchangers in the equipment pit. Flow rates through the system at from 0 to 20 gpm are adjustable through use of the demineralizer pressure-control valve (PIC-4aV), located at the inlet to the heat exchangers. A pressure breakdown orifice downstream from the control valve reduces supply pressure as necessary for system protection. Adjustment of the control valve is accomplished at a control panel located on the east side of the reactor building operating floor.

Demineralizer supply water flows in series through the first two-pass heat exchanger and the second four-pass heat exchanger where it is cooled to 120°F, which is the recommended maximum temperature for the ion-exchange resins. Water then flows directly to the filter and mixed-bed ion-exchange column in the water-storage pit. This discharge from the column passes through a filter and is then sent back to the reactor vessel if the auxiliary-water pump is in use, or to the feedwater-storage tank if reactor pressure is creating the demineralizer flow.

9. Makeup-water System

A 1½-in. raw-water supply line connects with a makeup-water system located on the main floor of the reactor building. Flow through this makeup system is rated at a maximum of 70 gpm.

The demineralized water is stored in a 5000-gal makeup-water-storage tank in the reactor building. From here, makeup water is fed by gravity automatically by control valve LC-2aV to maintain a proper operating level in the feedwater-storage tank (see Fig. 49).

When operating at 40 Mw, it is necessary to dispose of approximately 20 Mw by venting steam to the atmosphere because of the limitation of the steam-consuming equipment. This in turn creates a loss of about 120 gpm of feedwater, which must be made up from the makeup-water system. If this system is operated at a rate of 60 gpm, there remains 60 gpm to be made up from the 5000-gal storage tank. This is the factor which establishes the duration of 40-Mw operation at about $1\frac{1}{2}$ hr.

10. Raw-water Systems

A raw-water distribution header, located near the east wall of the reactor building, is supplied by a 4-in. supply line from the EBR-I well. Distribution lines from this header include: a $1\frac{1}{2}$ -in. line via a surge tank to the makeup-water system; a $\frac{3}{4}$ -in. line to provide cooling water to the sample test station; a 2-in. line to feed the reactor-shield cooling system and to supply gland-cooling water to the auxiliary water pump; a 3-in. cooling-water supply line to provide flow in series through the two demineralizer heat exchangers; and a $1\frac{1}{2}$ -in. line to supply water to the water-storage pit. In addition, the raw-water supply header contains a $1\frac{1}{2}$ -in. firehose connection and three $\frac{3}{4}$ -in. hose connections.

It is estimated that a raw-water supply of 175 gpm is sufficient to fulfill the total demand of the above systems. There are at least 200 gpm of raw water available for use at the BORAX site.

The shield-cooling system, mentioned above and shown in Figs. 45 and 49, requires a maximum flow of approximately 30 gpm. This system includes: a series of 1-in. pipes located about the horizontal reactor centerline and in contact with the inside steel surface of the reactor pit; a parallel system of $\frac{1}{2}$ -in. piping 12 in. from the pit inner wall, and $\frac{1}{4}$ -in. cooling coils on the instrument-hole liners. To provide for the possibility of investigating a separate superheating core in the steam dome area of the vessel, an auxiliary set of cooling coils is imbedded in the pit wall near the top of the reactor but is left unconnected until required.

At maximum rated power, the raw-water makeup required by the main-condenser cooling tower is approximately 112 gpm. For conservation purposes, the cooling-water return lines from the demineralizer heat exchangers, the shield-cooling coils, and the pump glands are all combined in a 4-in. line piped to the turbine building and discharged into the condenser cooling-water piping.

11. Other Systems

a. Sampling System

To permit sampling of the principal plant streams, stainless steel tubing lines have been provided between selected sampling

points on the piping systems and in the reactor vessel, and a sampling station located in the northwest corner of the reactor building. These lines permit sampling of:

- (1) reactor feedwater system,
- (2) reactor forced-convection system,
- (3) steam system,
- (4) influent and effluent of the full-flow condensate demineralizer,
- (5) influent and effluent of the reactor-water demineralizer,
- (6) plenum chamber below core in reactor vessel,
- (7) steam dome in reactor vessel.

The sampling station comprises a panel-mounted array of components used for analytical operations on the plant sample streams. These are enclosed in a hood to contain and remove radioactive water, solids and gases, thereby preventing contamination of the reactor building during sampling operations.

The operations which can be performed on selected sample streams at the sampling station include:

- (1) continuous measurement of pH,
- (2) continuous measurement of electrolytic conductivity,
- (3) continuous filtration to collect undissolved solids for analysis,
- (4) continuous ion-exchange to concentrate dissolved-solids for analysis,
- (5) continuous measurement of radiolytic, dissolved oxygen,
- (6) continuous monitoring for fission product activities,
- (7) periodic collection of "grab" samples of water for laboratory chemical and radiochemical analyses of solids and gases.

These operations are involved in both the routine monitoring and control of plant water chemistry and special testing to investigate chemistry variables in the plant.

b. Drains

All systems are equipped with drains which discharge into one of three local sump pits. Pumps deliver the sumpage to an existing leaching pond. High-temperature drains are suitably vented. Disposal of liquid radioactive waste is controlled by monitoring.

c. Water-storage Pit Cleanup System

A 10-gpm circulating pump, cellulose filter and small mixed-bed demineralizer are provided to keep the water in the water-storage pit and old reactor vessel transparent for viewing and at high purity to inhibit corrosion of stored aluminum core components.

C. Process Instrumentation and Control Systems

The equipment included for process control includes instruments, transmitters, and valves in the various systems which are described herein and shown in Figs. 48 and 49.

1. General Safety Considerations

a. Signal Transmission between Reactor and Control Buildings

All instrument signals are carried electrically between the reactor and control buildings. Proportional-type signals use a biased signal wherever possible (i.e., 4 to 20 ma proportional to 0 to 100% of process variable) so distinction can be made between an extreme value of a variable and a failed instrument.

b. Failure of Instruments

Design is based on the use of "fail-safe" circuits so that scrams are actuated on loss of power. Air-operated valves are installed. These close on loss of air to reduce the interconnection of systems and to seal the reactor from the atmosphere in case of accident.

Two valves are exceptions to this air-to-open action. Valves PIC-5V (gland-seal-water pressure-control valve) and HC-4aV (demineralizer-cooling-water valve) are designed to fail open on loss of air to prevent condenser flooding and resin damage, respectively.

c. Valve Position Indicators and Lamps

All valves which are critical in terms of safety to plant and personnel are equipped with limit switches and/or stem position transmitters. Position is noted in the control building by indicating lamps or

electrical meters calibrated to read in percentage of valve opening. Limit switches are also used in interlock circuits to prevent undesired operations, such as starting the forced-convection pump with its suction and/or discharge valves closed.

d. Process System Interlocks

Table XVI shows the electrical interlocks which are provided to give protection to personnel and equipment.

Table XVI

PROCESS-SYSTEM INTERLOCKS

A. Superheater Startup and Shutdown System

1. The superheater-drain valve (HC-6AV) cannot be opened unless the superheater-flood valve (HC-5V) is closed. This interlock prevents draining of the boiler through valve HC-6AV.
2. The pressure-equalizing valves (HC-17AV and HC-18AV) open and close with opening and closing of the superheater-flood valve (HC-5V). The equalizing valves open with the opening of the superheater-drain (HC-6AV) valve providing the back pressure control valves (PIC-IV and PIC-2V), the reactor vent valve (HC-1AV) and the superheater vent valve (HIC-5AV) are closed. The equalizing valves can be opened independently of the superheater-flood and drain valves providing the back-pressure-control valves (PIC-IV and PIC-2V), the reactor-vent valve (HC-1AV), and the superheater-vent valve (HIC-5AV) are closed.

The boiler and the superheater are separated mechanically within the reactor vessel. When the reactor is producing superheated steam, they must stay isolated. When the superheater is being flooded or drained, the top of the superheater fuel assemblies, the superheater-level-measuring line, and the pressure vessel steam dome must be at the same pressure (interconnected) in order to allow the proper flow of water. After the flood valve has been closed and the superheater has been drained, the superheater water level may be checked. If a true reading on the superheater-water-level indicator is to be obtained, the superheater-level-measuring line must be connected to the pressure vessel steam dome. This is accomplished by independent operation of the equalizing valves. The interlock to restrict independent opening of the equalizing valves is provided because opening the equalizing valves while the reactor is flowing steam would bypass the superheater.

Table XVI (Cont'd.)

3. The superheater-flood valve (HC-5V) can be tripped open only when the reactor is shut down (i.e., scram circuit de-energized to unlatch control rods). Also, the manual cable trip operator on the solenoid trip valve and the reach rod operator on the manual superheater-flood valve are secured with frangible padlocks to inhibit inadvertent flooding.

Introducing water to the superheater while the reactor is running could cause a severe reactivity addition as well as thermal shock of the fuel plates.

4. The superheater-vent valve opens a preset amount on any scram regardless of the automatic-manual selector switch position. The selector is kept in automatic during normal steam flow. An alarm light adjacent to the switch indicates that this selector is in manual. With the selector in automatic, bypassing the low steam flow scram by means of key switch No. 3 opens the superheater-vent valve the preset amount. On failure of normal electric power, 24-volt dc battery power and emergency air may be used, by means of the emergency vent-control switch, to cycle the valve from full open to full close.

These provisions assure flow of an adequate amount of cooling steam through the superheater in emergency conditions.

B. Boron-addition System

1. The boron-dump valves operate in sequence. Operation of the pushbutton initiates the closing of the boron-leakage valve (HC-2DV); closing of this valve initiates opening of the boron-tank pressure-equalizing valve (HC-2AV); opening of this valve initiates the opening of boron-seal valve (HC-2CV) and opening of this valve initiates the opening of the boron-to-reactor valve (HC-2BV). All valves are tripped by emergency 24-volt dc power and must be manually reset.

This system assures that leakage from the boron tank will be to waste and not to the reactor. When boron flows into the reactor, the leakage valve closes first to prevent dumping the boron to waste. The pressure-equalizing valve opens ahead of the seal and reactor valves to permit proper flow of water to the reactor.

Table XVI (Cont'd.)

C. Forced-convection System

1. The forced-convection pump-suction valve (HC-11V) must be full open and the discharge valve (HC-10V) must be opened a predetermined amount (for minimum flow) in order to run the forced-convection pump. Operation of this pump without flow could cause it to overheat in a short time.
2. Water must be flowing in the circulating system or the forced-convection system before the scram circuit can be reset and the reactor started up.
3. A temperature-difference interlock which prevents scram system reset and reactor startup when the temperature difference between the water in the reactor vessel and the water at the forced convection pump exceeds 15°F.
4. An interlock is provided to require that the forced-convection suction valve be closed in order to run the auxiliary water pump. This interlock prevents short circuiting of the auxiliary-water flow. A limit switch on the discharge valve prevents operating the auxiliary pump when the valve is open less than 10%.

Interlocks 2, 3 and 4 can be bypassed by means of key switch No. 6 when the forced-convection piping is disconnected.

The forced-convection system water and piping must be kept at the same temperature to prevent excessive mechanical stresses and the sudden introduction of cold water, and thus of reactivity, to the reactor.

5. The forced-convection pump cannot be started while the reactor is operating.

Rapid changes in forced-convection flow rate could cause large reactivity changes.

D. Auxiliary-water Systems

1. The demineralizer, pressure-control valve (PIC-4AV) closes and the auxiliary-water pump does not run if the temperature of the water going to the ion-exchange columns exceeds 120°F. The ion-exchange resin will be damaged if the temperature exceeds 120°F.

Table XVI (Cont'd.)

2. The electric preheaters are not energized unless the flow through the preheaters is greater than 100 gpm. Flow below 100 gpm would cause the electric heaters to burn out.
3. The auxiliary-water pump suction valves (FC-1AV) and the auxiliary-water pump are operated by the auxiliary-water pump motor starter.

The valves prevent having reactor pressure on the auxiliary-water-pump packings during normal operation and prevents having reactor pressure on the reactor-water-demineralizer system if the manual isolating valves are not properly closed.

4. The auxiliary-water-pump-bearing temperature must be below 150°F in order to run the pump.

Cooling water must be supplied to the bearings of this pump to maintain satisfactory temperatures on the pump packing.

2. Superheater Startup and Shutdown System

The valves in this system are operated from the control building through electrical connections, although in some cases local operators are provided for emergency valve operation. A combination of solenoid trip-type valves and air-operated valves is used.

a. Superheater-vent Valve

Valve (HIC-5AV), the superheater-vent valve, is interlocked to assure coolant steam flow through the superheater. During reactor shutdown and startup, key switch No. 3, low steam-flow-scrum interlock, is "off" and the vent valve automatic-manual selector switch is on "manual." This gives the operator manual control of valve opening. When main-steam flow exceeds 2000 lb/hr, key switch No. 3 is turned "on" and its key removed, and the selector switch is turned to automatic. The superheater-vent valve opens a preset amount on any scram. If the operator turns switch No. 3 "off" before turning the selector switch to "manual," the valve still automatically opens the preset amount.

The valve is air-operated and the remote electrical control signal is converted at the valve into an air signal. This valve is operated from the emergency systems.

The superheater-vent flow indicator (FI-4) associated with this system gives the reactor operator information needed to regulate flow. A temperature recorder also provides information on superheater fuel temperature at specific locations.

b. Superheater Water-level Indicator

This level indicator (LI-2) is used by the reactor operator to determine flooding level in the superheater during startup and shutdown operations of the reactor. It operates from a differential-pressure transmitter connected to a pipe-line external to the vessel.

c. Superheater Drain-flow Indicator

This flow indicator (FI-5) is used for superheater startup operations. Its purpose is to indicate flow of water and steam from the superheater during the draining phase of the startup operation.

3. Boron-addition System

To accomplish boric acid addition, valve operation is initiated from the control building. All valves are solenoid-operated from the 24-volt dc supply, and must be energized to complete the action. Valve-control circuits are interlocked for proper sequence of operation, and indicating lamps are provided for monitoring the operation of all valves involved. Boric acid is added by gravity feed.

A local pressure gage, local level-gage glass, a temperature indicator and low water-level alarm in the control building are provided for the boron tank. An electric heater and automatic thermostat control are provided to maintain the boron tank at elevated temperatures when required.

4. Forced-convection System

The forced-convection pump is started and stopped by push-buttons on the control console. The suction and discharge valves are electric-motor operated and are positioned by control switches on the control console.

Flow rate in this system is recorded. The signal is received from a differential-pressure transmitter connected to a flow-tube section.

5. Steam-pressure Control System

The steam-pressure control system regulates flow of reactor steam to maintain reactor vessel pressure constant at a preselected point.

Two air-operated valves with two separate controllers maintain set pressure during steam flow transients created by load changes or variation in reactor steam generation rate. Through the main-steam, back-pressure control valve (PIC-1V), reactor steam is routed to the steam turbine via the desuperheater. Through the vent-steam, back-pressure control valve (PIC-2V) the excess steam above turbine capacity is vented to atmosphere. This valve is also used at pressures below turbine operating range.

The following instruments are included to monitor this system:

- a. Reactor Vessel Pressure Recorder (PR-2)
- b. Superheater Pressure Drop Recorder (PR-3)
- c. Steam-to-turbine Pressure Recorder (PR-1)
- d. Reactor Water and Steam Temperature Recorder (TR-2)

6. Desuperheater and Turbine Steam Systems

Reactor steam, both saturated and superheated, is passed through a desuperheater which is automatically controlled to maintain steam conditions at the turbine inlet between saturation conditions at 250 psig and superheat conditions of 585°F at 350 psig.

The temperature recorder-controller sends a signal to the air-operated desuperheater-feedwater control valve (TRC-1V) which admits feedwater to the desuperheater.

A pressure-regulating valve (PIC-9V) maintains constant steam pressure at the turbine inlet, bypassing excess to the condenser through a pressure-controlled, air-operated valve. A flow recorder monitors this flow.

7. Feedwater Control System

Feedwater admission to the vessel is controlled by two valves, independently operated. The main-feedwater control valve (LC-1V) is capable of regulating feedwater rates over the range from 0 to 275 gpm, and is air-operated by an automatic or manual control signal output from the three-element control system. The second low-rate-feedwater control valve (HC-14V) passes a maximum of 30 gpm and is manually controlled through an electric motor to provide small feed rates to the vessel.

The automatic level-control system is of the three-element type similar to that used in conventional power plant boiler-level control. A reference signal, proportional to the desired water level, is compared

with a signal proportional to actual water level. The error developed is then added to the error signal, produced by comparing steam flow and feedwater flow signals. The sum of these signals is then used to actuate the main-feedwater control valve which corrects the reactor water level. Each of the three signals (feedwater flow, steam flow, and reactor water level) is capable of adjustment to provide the necessary gain and time-constant factor for proper reactor water-level control. The effect of steam flow and feed flow can be made negligible in the system. This mode of operation is termed a "single element" control since only the water-level signal is used to control actual water level. Requirements for the control system and components are set so that for all normally anticipated transients in steam-flow rate and power, the water level is maintained within ± 1 in. of the set-point. The set-point is adjustable from 5 to 15 ft above the bottom of the reactor vessel.

Recorders monitor steam flow rate, steam temperature, feedwater flow rate, and reactor vessel-water level. Manual compensation for temperature effects on steam mass-flow rates is made by electrical-compensating circuits which are to be set by means of a calibrated-dial potentiometer. Integrated feed and steam flow is indicated on each recorder.

Feedwater pumps are turned on and off at the control console. A flow indicator for measuring the 0 to 30-gpm rate from the manually operated low rate-feedwater valve (HC-14V) is provided. A separate, wide-range, reactor water-level indicator is provided for observing draining and flooding. The gage glass provides backup indication of reactor water level through the television system.

The water level in the feedwater-storage tank is maintained by on-off-type control of the makeup-water-addition valve (LC-2aV). This valve is opened or closed depending on the level in the feedwater-storage tank. Feedwater-storage level is recorded on recorder (LR-2) which is located in the control building.

8. Reactor Preheat System

The electric preheaters and auxiliary water pump are turned on and off from the control console by switches located in the reactor building. The heating rate can be set at 36, 72, 118, or 144 kw. A flow indicator in the reactor building allows monitoring of the flow rate in this system.

9. Makeup-water System

The raw-water demineralizer is manually controlled and water is fed manually to the makeup-storage tank until the high-level float control is reached. Then the raw-water supply valve to the demineralizer is

controlled automatically to maintain water level. Water-level recorder (LR-3) located in the control building records the level in this makeup-water-storage tank.

10. Reactor-water-demineralizer Pressure-control System

This system is designed to limit the pressure on the primary side of the reactor-demineralizer heat exchanger to its safe recommended value and is operated from the control console. The control system regulates by comparing the actual line pressure with a value set on the indicating controller and operating the pneumatic, demineralizer pressure-control valve (PIC-4aV) to vary flow to the heat exchanger. The flow rate is monitored by a local rotameter. A pressure-reducing orifice in the line limits the maximum downstream pressure at 20 gpm to 50 psig for a maximum upstream pressure of 600 psig.

11. Process Temperature-indicating System

This system consists of a potentiometer-type temperature indicator, associated thermocouples, a selector switch to monitor process temperatures as required, and has a capacity of 24 temperature points. Process-system temperatures measured are:

- a. steam-to-turbine
- b. forced-convection water from reactor
- c. forced-convection water to reactor
- d. preheater-water from reactor
- e. preheater-water to reactor
- f. shield-cooling inlet
- g. shield-cooling outlet
- h. feed pump inlet
- i. reactor-water-demineralizer heat-exchanger inlet
- j. reactor-water-demineralizer heat-exchanger outlet
- k. forced-convection-pump casing
- l. boron tank water.

12. Superheater Fuel Element Temperature

If available, two superheater fuel assemblies will be installed with thermocouples attached to them. A monitoring system measures

these temperatures once each 120 sec and sounds an alarm should any temperature exceed the design maximum. The temperature from any one of these thermocouples can be recorded by selecting the particular point with a selector switch. An independent, superheater fuel element, "hot-spot"-temperature scram and alarm circuit is also included.

13. Reactor-vessel and Shield-wall Temperature-recording System

This system consists of an automatically cycled multipoint temperature recorder and associated thermocouples. Thermocouples are positioned on the external wall of the reactor vessel and within the reactor pit shield wall and floor to obtain records of temperature gradients.

Reactor-vessel temperatures are measured at 18 locations which are all vertically coplanar, and paired to locate 9 installations diametrically opposite each other. A symmetrical grouping on the vessel is thus accomplished.

Shield-wall thermocouples are positioned at points within the wall calculated to be within the zone of maximum heat generation. An additional thermocouple, imbedded in the concrete shield wall, activates a high-temperature alarm by means of an alarming-pyrometer indicator in the event of excessive temperature. The pyrometer can be connected to any shield-wall thermocouple which exhibits the highest operating temperature.

14. Miscellaneous Control Building Meters

The makeup-water-tank and the feedwater-storage-tank level recorders are also located in the control building.

15. Control-rod Seal-water Flowmeters and Valves

The seal-water supply valve is operated from the control building. Inlet and discharge flow to each rod drive mechanism seal is measured by 18 "Rotameter"-type flowmeters, and is adjusted by valves associated with each flowmeter. Valves and flowmeters are located in the subreactor room.

16. Local Pressure Gages

The following pressure gages are installed locally in the reactor building:

- a. feedwater from demineralizer system
- b. water from electric preheaters

- c. reactor standpipe
- d. boron tank
- e. water to reactor (forced-convection line)
- f. water from reactor (forced-convection line)
- g. batch-emergency-feed tank
- h. forced-convection pump, P-1 (discharge)
- i. auxiliary-water pump, P-2 (discharge)
- j. auxiliary-water pump, P-2 (suction)
- k. feedwater pump, P-3 and P-4 (suction)
- l. feedwater pump, P-3 (discharge)
- m. feedwater pump, P-4 (discharge)
- n. emergency air-compressor
- o. emergency air header
- p. condensate filter
- q. condensate demineralizer filter
- r. air pressure.

17. Local Temperature Indicators

Temperature indicators are installed locally at the following locations in the reactor building:

- a. reactor-water-demineralizer heat exchanger cooling-water outlet
- b. batch-emergency-feed tank
- c. forced-convection pump suction
- d. electric heater outlet
- e. electric heater inlet
- f. demineralizer heat exchanger first-stage outlet
- g. cooling water supply line
- h. condensate demineralizer tank

18. Television Monitor

A remotely controlled television camera located in the access shaft allows observation of the reactor vessel water-level sight glass, reactor vessel pressure gage, and general conditions of the area and equipment. The viewing screen is located in the control building.

19. Instrument Compressed-air Systems

Normal air supply for operation of valves is furnished by two compressors and is monitored by low-pressure alarms. An emergency air compressor is electrically interlocked to start at a preset low pressure and, by means of suitable check valves in the system, furnishes air only to the superheater-vent valve, the experimental-steam back-pressure control valve, the reactor-vent valve, and the two pressure-equalizing valves. These 5 valves are required to isolate the reactor vessel, and to control temperatures and pressures in the reactor.

D. Buildings and Electrical Service Systems

1. Reactor Building

The BORAX V reactor building is an insulated, prefabricated metal building, 40 x 86 x 20 ft high (vertical wall height). Figure 45 shows details of the building, pits, trenches, and major equipment.

The equipment pit houses the feedwater pumps, reactor-water-demineralizer heat exchangers, condensate filters and demineralizer, feedwater-storage tank, and water-storage-pit filter system. A water-storage pit, a dry-storage pit, and the old BORAX reactor pit are used for fuel and radioactive material storage. The reactor vessel is mounted in a reactor pit and the forced-convection pump, auxiliary-water-pump, and electric preheaters are in the subreactor room. Access to the subreactor room is through an access shaft by means of a spiral stairway.

Conventional structural concrete is used throughout for all footings, foundations, walls, etc., except for the top and bottom shielding around the reactor, where a magnetite high-density concrete is used. Water-cooled coils imbedded in the pit walls surrounding the reactor vessel prevent overheating and spalling of the concrete.

All plant piping and valves are in shielded trenches below the floor line and are accessible for servicing. Steam lines lie in the steam pipe trench and run underground from the reactor building to the turbine building. Electrical and control wiring runs in floor trenches, or trays and conduit along the walls.

Access to the reactor building is provided for both personnel and freight by a 16 x 13-ft-high, rolling steel door and by a 3 x 7-ft hollow-metal door at each end, and one 3 x 7-ft door at the middle of the west wall. Eight 4 x 6-ft fiberglass-plastic windows on both the east and west walls admit daylight to the interior.

A heating and ventilating building of pumice-block construction, located next to the reactor building, as shown in Fig. 45, houses the furnace, pit ventilation blowers and filters. Two 3000-cfm blowers are installed for pit ventilation. For summer use, both are needed during reactor operation to supply sufficient air to keep the subreactor room temperature below 120°F. When cooler ambient temperatures prevail, as during the winter months, one blower will suffice. Inlet air is drawn in through the hot air furnace and warm air expelled into the building through a distribution duct along the west wall. The ventilation blower draws the building air into the east pipe trench, down the access shaft, up through the reactor pit, out the steam pipe trench, through a filter bank, and then discharges it up the stack. A small amount of air is also drawn along the west pipe trench.

The steam pipe trench between buildings is ventilated to maintain the temperature below 120°F. Air is drawn in at the turbine building, passes the full length of the trench, and is exhausted at the reactor building end of the trench by a blower. Resin in the condensate demineralizer and insulation on instrument wires are thereby protected from excessive temperatures, and the air line in the trench is protected against excessive pressure due to overheating.

The reactor building 10-ton bridge crane is suspended from the building frame. The crane serves almost the full area of the building and all pits, with a maximum hook height of 15 ft above the floor and ability to reach $51\frac{1}{2}$ ft below floor level.

Crane speeds are as follows:

- Hoist - 8.5 to 17 fpm in 5 steps, with auxiliary drive separately controlled to give inching speed of 1.7 fpm.
- Trolley - 25 to 50 fpm max, in 3 steps.
- Bridge - 25 to 50 fpm, in 3 steps.

Speed controls must pass through the slow-speed range first. Two pushbutton stations are provided with the crane: one pendant from the trolley, with a cable tension reel; the other a remote control station mounted on the west wall.

2. Control Building

The control building, shown in Fig. 50, is located about one-half mile southeast of the reactor building and is adjacent to the EBR-I site. It has pumice block walls, a tiled concrete slab floor, and a flat builtup roof with a steel roof deck. The building, which measures 20 x 40 ft inside, is divided into a control room, office, utility room, and toilet. Covered trenches, beneath the control panel locations, are provided in the control room floor, for control and instrument cable.

Fig. 50

Control Bldg. Floor Plan

Constant temperature is maintained in the control building by a combination heating and air-conditioning unit.

3. Turbine Building and Cooling Tower

The existing Butler-type turbine building and the redwood forced-draft cooling tower are reused essentially unchanged.

4. Field Chemistry Laboratory

A trailer which has been modified for use as a chemistry laboratory in the field for analysis work on radioactive samples is located just west of the H&V building as shown in Fig. 58 (see p. 191).

5. Electrical Service Systems

a. General

Electric power is distributed through a radial system. Since the plant is an experimental facility supplying an artificial load, a high degree of continuity of service is not required. Therefore the simplest, safest, least expensive power distribution system is used. Figure 51 is the single-line diagram for electrical power distribution.

Power is supplied to the EBR-I and BORAX area by a $3\frac{1}{2}$ -mile-long distribution line from the Central Facilities Area. In order to limit voltage fluctuations caused by starting the 450-hp forced-convection pump, it was necessary to underbuild the existing 12,470-volt line with a parallel circuit.

b. Auxiliary Power

The forced-convection pump is supplied through a 500-kva step-down transformer and 4160-volt motor controller. The motor starts across the line and the controller provides short circuit, overcurrent, and undervoltage protection.

The remainder of the plant auxiliaries are supplied by a bank of three 167-kva transformers which step down from 12,470 to 480 volts. The reactor building motor-control center supplies power to motors, resistance heaters, reactor control circuits, lighting, and instruments. Control of these devices is accomplished in the control building and/or in the reactor building, depending on the service required. Standard overload short-circuit and undervoltage protection is provided. Auxiliaries for the turbine-generator are supplied through an overhead feeder from the reactor building to the turbine building. A motor-control center provides local control and protection for motors and other equipment in the turbine building.

c. Turbine-generator

The turbine-generator can be controlled from the turbine building, or from the control building after the turbine is at rated speed and the manual throttle valve is fully open. The power produced by the generator at 2400 volts is dissipated in a water rheostat. The controls

Fig. 51

Electrical Single-line Diagram

consist of generator breaker, field breaker, governor control, and water-rheostat level control. Protection for the turbine-generator is provided by an overspeed trip and differential current relays.

d. Control Building Power

Electric power to the control building is supplied from the EBR-I distribution system at 480 volts. Failure of power to either the reactor building or to the control building will shut down the reactor.

e. Lighting

Incandescent lighting is used for all buildings. There is a step-down transformer, operating at 480 volts to 120/240 volts, in each building. Standard distribution panels containing circuit breakers provide protection and control.

f. Instrumentation

Isolating transformers for instrument power are included at the reactor building and the control building to step down from 480 volts to 120/240 volts. This power is used for nuclear instruments, process instruments, and special test instruments.

g. Emergency Power

There are two sources of emergency power. One is the 24-volt dc battery and the other is the gasoline-engine-driven, 115-volt, 60-cycle generator.

The 24-volt battery has an 8-hr rating of 100 amp-hr, and is considered the most reliable source of power.

The emergency generator has a capacity of 10 kw, and starts automatically on power failure at the reactor or the control buildings. Automatic transfer switches connect essential instruments and controls to this power source when the generator reaches operating voltage. A $1\frac{1}{2}$ -hp, 6-scfm compressor operates from this generator to supply air to the essential control valves.

The items of equipment connected to the emergency power supplies are listed in Table XVII.

Table XVII

EQUIPMENT CONNECTED TO EMERGENCY POWER SUPPLIES

A. Emergency Air

1. Superheater-vent Valve (HIC-5aV)
2. Reactor-vent Valve (HC-1aV)
3. Experimental-steam, Back-pressure-control Valve (PIC-3V)
4. Pressure-equalizing Valves (HC-17V and HC-18V)

B. 120-volt ac Power

1. Emergency Air Compressor
2. Reactor-water Level Indicator (LI-1)
3. Superheater-water Level Indicator (LI-2)
4. Main-steam Flow Recorder (FR-1)
5. Linear Flux Recorder
6. Reactor-pressure Recorder (PR-2)
7. Superheater-vent-valve Control (HIC-5)
8. Experimental-steam, Back-pressure-control-valve Control (HIC-3)
9. Reactor-vent-valve Control (HC-1aV)
10. Superheater-vent Flow Indicator (FI-4)

C. 24-volt dc Power

1. Reactor-vent Valve (HC-1aV)
2. Superheater-flood Valve (HC-5V)
3. Pressure-equalizing Valves (HC-17V and HC-18V)
4. Boron-injection Valve (HC-2aV, HC-2bV, HC-2cV and HC-2dV)
5. Turbine-generator Breaker Trip

6. Communications and Alarms

The normal means of communication between the control building, the reactor building, and the turbine building is a two-way intercom system. There are four stations in the reactor building, one station in the turbine building, and one at the control console. Interconnected with the intercom system is a paging system which permits the control console operator to call the turbine and reactor buildings. One of the stations in the reactor building can also be used for paging.

For emergency conditions and for equipment calibration and checking, a sound-powered phone circuit is provided with plug-in jacks at various points in the reactor and control buildings as well as at the control console.

Each building has a fire-alarm signal box connected to the site fire-alarm system. Gongs are also provided.

Bell System phones are located in each building and connected to the EBR-I and CFA switchboard.

There is an evacuation alarm horn and evacuation alarm switch in each building. This system is tied into the EBR-I system.

A flashing red light indicates that the reactor is in operation. Two lights are located outside the reactor building, two on the main floor, one in the subreactor room, and one in the equipment pit. Warning horns located inside and outside the reactor building, in the subreactor room, and in the equipment pit sound for 10 sec on reactor startup.

IV. MANAGEMENT

Operation of the reactor facility will be through a three-level organization: manager, supervisor, and operator.

The managerial level will consist of an Operations Manager and an Assistant Operations Manager. These positions will be filled with personnel appointed by the Director of the Idaho Division.

The Reactor Operations Manager will be responsible for the program and safe operation of the reactor facility. He will have complete authority within limits as set forth in the Hazards Summary Report, Operating Manual, and directions from the Division Director. He will be directly responsible to the Division Director.

The Assistant Operations Manager will act for the Operations Manager when specifically instructed by the latter. He may not, without authorization of the Division Director, assume the duties of the Operations Manager.

The supervisory level will consist of an Operations Supervisor and Operations Foreman. They will be appointed by the Operations Manager from a list of qualified personnel with approval by the Division Director. Supervision of actual operation will be by the Operations Supervisor. He will comply with standard operating procedure and will not deviate from this by his own volition except to take the necessary action to safeguard the facility in case of an unpredicted emergency. During operation, the Supervisor will be readily available to the control building at all times.

The Operations Foreman will have responsibility for such routine matters as maintenance, inspection, and other duties as delegated by the Operations Manager. He will assist the Operations Manager in the assignment of Operators to specific tasks and will perform the necessary supervision to insure that procedures are properly followed.

The operational level will consist of qualified reactor Operators. They will perform actual operation of the reactor controls and system and of any auxiliary equipment which directly affects reactor operation. During reactor operation, at least two Operators will be present in the reactor control building. Operators and technicians will be stationed in the turbine-reactor building area as required.

The operating organization for this reactor has been planned for single-shift operation. However, for certain tests and experiments, it may be necessary to operate continuously for a limited number of days. In this case, a qualified Operations Supervisor and qualified Operators will be assigned to each shift by the Operations Manager.

A single set of operational keys shall be available to the operating organization. Their control shall be the responsibility of the Operations Manager who shall assign them to the responsible Operations Supervisor as required. When shift operation is involved, the keys may be re-assigned from one Supervisor to another with the Operation Manager's prior knowledge.

The policy concerning access to the BORAX V facility is to permit normal access by the BORAX project personnel as required by the performance of their duties and as administered by the Operations Manager or Operations Supervisor. Admittance of other Argonne National Laboratory personnel and visitors is by permission of the Operations Manager or Operations Supervisor.

Vehicle access to the reactor area is controlled by a barricade on the BORAX road at its intersection with Adams Blvd. (see Fig. 58, p. 191). This barricade is normally kept locked and may be opened by properly authorized ANL personnel only, to permit the passage of a vehicle. Warning signs posted at the barricade, at the intersection of Adams Blvd. and Van Buren Ave., and in the desert around the reactor area, instruct the casual visitor to contact the EBR-I guard post regarding entrance to the BORAX facility.

The reactor building interior is protected by a security alarm system which alarms in the EBR-I guard post. When the reactor building and area is occupied by the operating crew, the security alarm is inactivated, the reactor and turbine building doors are unlocked, and the gate to the security fence around the reactor building is unlocked. When the reactor building area is vacated by the operating crew, the security alarm is activated, the doors to the reactor and turbine building are locked, and the gate to the fence is locked. The control building is locked when not occupied by BORAX project personnel.

When the reactor area is vacated by personnel for hazardous experiments, a control point is set up at the intersection of Adams Blvd. and Van Buren Ave. to control access to the reactor area.

V. REACTOR OPERATIONS

Initially, the entire plant will be checked out by means of operational tests, hydrostatic tests, and instrument calibration. Following this, the operation can be divided into three classes: Loading, Open-Vessel, and At-Pressure. For any class of operation, a check-out procedure will be followed daily to assure that essential equipment is operating satisfactorily.

A. Loading

An antimony-beryllium neutron source will be installed in the reactor for loading or operation. For initial and other special loadings, it is necessary to locate the neutron detectors in temporary positions inside the reactor vessel. Loading shall proceed as required by changes in multiplication. Instrumentation will be located so changes in multiplication can be observed by both the control room operators and the loaders.

When changes in a core structure are made, all fuel is removed and the control rods are disconnected from their drives. The control rods will be reconnected and tested before fuel is reloaded into the reactor. Removal or maintenance of control rods or drives will be under the immediate supervision of a Supervisor. No maintenance work will be done that requires movement of control rods until the fuel loading is such that the reactor would still be subcritical with the rods involved completely removed. An operator will be stationed on the control room whenever movement of fuel or control rods is involved. During any control rod movement, instrumentation will be located so changes in multiplication can be observed by both the control room operator and the people working on the control rods.

All loading changes (unloadings, loadings, or any movement of fuel) will be made under the immediate supervision of the Operations Manager or the Assistant Operations Manager. Proposed loading changes will be checked before performance by at least 2 qualified persons for anticipated effect, and during the change for proper placement of the fuel. The excess reactivity shall be adjusted so the reactor cannot be made critical under any condition by the withdrawal of a single control rod.

A health physics representative will be present during all loading changes to assure that proper radiation-monitoring instruments are available, and that proper precautions for protection against radiation are taken.

B. Open-vessel Operation

After each change in loading, the reactivity effect will be measured by low-power critical runs. Open-vessel operation will be construed to mean any attempt to make the reactor critical without the top shielding in

place. Normally, in this type of operation all personnel are evacuated from the reactor building before control rods are moved. For low-power flux-distribution measurements, etc., personnel may re-enter the building after suitable radiation surveys.

C. At-pressure Operation

There are 4 general classes of power operations for this reactor: 1) natural-convection boiling with superheater; 2) natural-convection boiling; 3) forced-convection boiling; 4) forced-convection boiling with superheater. There are 3 core structures: two for superheating operation, and one for boiling only. After inserting the desired core structure, the reactor is loaded with the vessel head removed. When the desired loading is reached and the open-vessel operation is completed, the reactor vessel head will be bolted in place and the shielding positioned. The following operating description shall pertain to normal operation after necessary "At-pressure" calibrations have been performed and approval for operation has been given. Since operation of a reactor with nuclear superheating is unique and untried, a detailed description of proposed operation under this condition is presented. The following steps shall be taken to bring the reactor to full power and back to shutdown conditions:

1. Natural-convection Boiling with Superheater

The procedures will be the same for both central and peripheral superheater core operation. The only difference is that in the first case the central control rod partially controls the superheater, and in the second case the outside rods partially control the superheater. The operating procedures shall be as follows:

a. Starting Procedure

Perform the startup checkout. Heat the reactor to operating temperature and pressure. Procedures will differ here depending on whether or not the superheater is flooded. These procedures are as follows:

Superheater Flooded

- (1) Withdraw control rods to attain reactor criticality.
- (2) Increase the reactor power to the level necessary to heat the reactor and vessel at a rate of temperature rise no greater than 200°F/hr. Electric preheaters may be used in addition to, or in place of, reactor power.
- (3) When the pressure reaches about 575 psig (before the main-steam, back-pressure-control-valve opens), the reactor power will be reduced by inserting control rods in preparation for draining the

superheater. Superheater control rods will be fully inserted. Calculations show that at 0.4 Mw the superheater elements are safely cooled by radiation and conduction cooling. The superheater will then be drained by closing the superheater-flood valve and opening the superheater-drain valve to blow the water from the superheater to the condenser. When the drain valve is opened, the pressure-equalizing valves open to vent the superheater second-pass and the superheater water-level-measuring column to the reactor steam-dome. Upon closing the drain valve, the superheater-vent valve will be opened to flow sufficient steam to atmosphere in order to keep the superheater fuel-element temperature below 1200°F, since the pressure-equalizing valves close when the drain valve is closed.

(4) Withdraw the control rods to raise reactor power and return to operating pressure. The power will be limited to give a rate of temperature rise of 200°F/hr. Eighty percent of the power produced by the reactor must be used to cause steam to flow through the superheater for fuel element cooling. As the pressure rises, the main-steam, back-pressure-control valve opens to flow steam to the turbine building. As soon as steam flow to the turbine building becomes adequate for superheater fuel element cooling, the superheater-vent valve will be closed.

Superheater Not Flooded

(Since the superheater fuel elements are adequately cooled by radiation when the power production is below 0.4 Mw, it may sometimes be desirable to leave the superheater unflooded between shutdown and the next startup. Under this condition of startup, the reactor heating rate is limited to 0.3-Mw reactor power plus the electric heater power. The pressure may be any value from atmospheric to 600 psig.)

(1) Withdraw the control rods (principally the boiler control rods) to attain reactor criticality.

(2) Increase the power to 0.3 Mw to raise the pressure to either 600 psig, or to that at which the main-steam, back-pressure-control valve begins to open.

(3) The superheater-vent valve will be opened to cause the flow of sufficient steam for superheater cooling, and the reactor power is raised to increase pressure further and begin flow through the main-steam, back-pressure-control valve. When a steady flow of steam through the back-pressure-control valve has been established, the superheater-vent valve may be closed.

b. Procedure for Reaching Full Power

(1) As steam flow to the turbine begins, the desuperheater will be placed in operation.

(2) With 350-psig saturated steam flowing through the turbine-bypass valve at about 6000 lb/hr, the turbine can be started and slowly brought to full speed.

(3) To reach higher reactor power and superheated steam conditions, the control rods for the boiler and the control rods for the superheater will be adjusted. The turbine load will be connected and increased. The power produced in the boiler and in the superheater can be adjusted to a limited extent by use of control rods in the respective sections.

(4) The turbine-condenser system dissipates approximately 20 thermal Mw. For higher powers, the vent-steam, back-pressure-control valve can pass steam to atmosphere. Operation at these higher powers is possible for only a short time because of the limited makeup-water capacity.

(5) To shut the plant down, the reactor power will be reduced by first inserting the superheater control rods and then the boiler control rods. The turbine load will be correspondingly reduced. The temperature of the steam flowing to the turbine is reduced by the desuperheater as required until the turbine is unloaded. Steam flow will be maintained at a rate which limits the superheater fuel-element temperature to the design maximum. When the control rods are fully inserted and the superheater fuel-plate temperature has been sufficiently reduced, either the flood valve may be operated to flood the superheater, or cooling may be accomplished by radiation and conduction alone.

2. Natural-convection Boiling

To operate without a superheater, the boiling-core structure is used. With the core structure in place, the reactor is loaded with fuel as described previously. The procedures for starting up and shutting down will be similar except for the operating temperature, which is that at saturation. Flooding, draining, and desuperheating systems are not operated.

3. Forced-convection Boiling

The same core support structure used for natural-convection boiling is used for this method of operation. The baffle for directing

forced-convection flow must be inserted in the reactor vessel. The forced-convection piping must be attached to the reactor vessel. Reactor loading, startup, and shutdown procedures are similar to those used for natural-convection boiling.

When the forced-convection piping is connected, the reactor vessel and the piping must be kept at the same temperature to prevent excessive stresses and to prevent large reactivity changes caused by the introduction of cold water. Also, the possibility of variations in reactivity due to large changes in forced-convection flow must be anticipated.

4. Forced-convection Boiling with Superheater

It is possible to use the forced-convection system while superheating. Flow capacity is reduced slightly because one of the forced-convection inlet nozzles is used for the entrance of the superheat drain and flood pipe. All of the precautions and scrams required for both forced-convection and for superheating will be effective.

D. Water Purification System Operation

The BORAX V power plant water purification components will be operated to maintain maximum attainable water purity and a pH of 7.0 under all operating conditions, with the following objectives:

1. To maintain halide (especially chloride) concentrations in the reactor coolant as low as possible, thus reducing the danger of halide-induced, stress-corrosion cracking of austenitic stainless steel, fuel element cladding and structural members.
2. To maintain concentrations of all water impurities, especially those subject to neutron activation, as low as possible, thereby reducing specific activity of the coolant and curtailing the formation of long-lived activities which can pose problems in plant accessibility, maintenance, and waste disposal. This also reduces the carry-over of impurities in the saturated steam from the boiling region, with possible deposition in the superheater region.
3. To reduce the danger of localized electrochemical corrosion due to deposition and reduction of heavy metals on cladding and core structures.
4. To reduce concentrations of ionic and particulate impurities which promote the dissociation of water under irradiation.

VI. EXPERIMENTAL PROGRAM

The proposed program of experiments in BORAX V is divided chronologically into two phases. The initial series of experiments after plant shakedown will be with superheating cores and fuel assemblies, using primarily natural convection. The purposes of these tests will be to prove thoroughly the integral superheating core concept; to investigate the operational aspects of integral nuclear superheaters; to compare the merits of a central versus a peripheral superheater zone location; to determine experimentally the statics and dynamics of the coupled cores; and to test superheater fuel elements of both the reference and advanced designs.

The second phase of experimentation will be done with a pure boiling core. The primary purpose of these tests will be to improve understanding of the factors which control the stability of boiling reactors at high power densities and limit the maximum stable power capability. Tests will be run using both natural and forced circulation.

A tentative experimental program is summarized in the following paragraphs:

A. Startup and Initial Operation

The reactor will be started up for the first time with a boiling natural-circulation core using boiling fuel assemblies which will later be used in the superheating cores. The reactor will be operated at power long enough to check out and calibrate the process piping, instrumentation and control systems, and to train operators. In-core instrumentation in the boiling fuel assemblies will also be proof-tested at this time. Finally, the forced-convection baffle will be installed and the operation with the forced-convection system will be tested briefly. The fuel and control rods will then be unloaded and the boiling core structure removed.

B. Superheating Experiments.

After installation of a natural-circulation superheating core structure (probably the central superheater), the reactor will be built up to full size, and the usual calibrations, measurements, and superheater flooding worth will be made at zero power. Before proceeding to higher powers, the superheater-vent, drain, and flood systems will be tested. In-core instrumentation will be used to observe core characteristics during operation. If necessary, power distribution in the boiling zone may be adjusted to attain the desired powers. The first superheating core will be operated until feasibility has been demonstrated. A forced-convection baffle may be installed and the superheating reactor tested with forced convection.

Following completion of these experiments, the core structure will be replaced and a similar series of tests will be made on the second type of superheating core (probably the peripheral superheater).

It is expected that superheater fuel elements of advanced designs will be tested either in the superheating core, if they are available at the time, or later by means of in-pile loops during the boiling core tests.

C. Boiling Core Experiments

Proposed experiments which are designed to gain understanding of boiling reactor characteristics may be divided into three categories: static measurements, dynamic measurements, and engineering tests. The nature of these experiments follow:

1. Static Measurements

These measurements will include the usual measurements and calibrations of various reactivity effects associated with the approach to critical, low-power, and full-power operation. Of prime interest here are reactivity changes affected by variations in operating conditions.

2. Dynamic Measurements

These measurements will investigate the stability characteristics of the system, with emphasis on testing prediction of the maximum stable power achievable. Oscillator techniques may be employed, but other techniques, such as noise analysis, may be used where appropriate.

3. Engineering Tests

These tests, which might actually be included under the above tests, will determine the effect of many possible design variables on the maximum stable power output of boiling reactors. Some of the variables are fuel-rod spacing or coolant-channel size, core size, flux flattening, chimney height, water level, coolant-channel inlet and outlet restriction. Such studies should permit a more confident optimization of the design of future boiling reactor systems.

D. Water Chemistry Experiments

It is intended that water chemistry experiments be conducted with every type of core tested in BORAX V. The water chemistry program for the BORAX V plant consists of two principal types of activity: chemistry monitoring and control, and chemical testing. The primary objectives of the program are to control water-chemistry conditions and to observe

the effect of basic processes occurring in the plant. Some of the processes of interest are:

1. corrosion
2. neutron activation of water and its impurities
3. transport and deposition of radioactive substances
4. release of fission products from the fuel to the coolant
5. water dissociation
6. chemical reactions promoted by radiation.

VII. HAZARDS EVALUATION

A. Inherent Safety of Boiling Reactors

A reactor may be called inherently safe against reactivity additions if it is able to tolerate these additions without deleterious effects to the surroundings, or, more restrictively, to the reactor itself. The inherent shutdown mechanisms of boiling water reactors as a type, with reactivity additions considerably greater than expected under normal operating conditions, have been shown to be high in the previous BORAX experiments (BORAX I-IV) and the continuing SPERT experiments. Detailed analyses of some of these experimental results have been applied to the EBWR, a boiling reactor of different design, to indicate similarly a rather marked degree of inherent shutdown capability. BORAX V should exhibit similar characteristics. However, if the large excess reactivity available in a cold, unvoided, boiling-water reactor is rapidly inserted, the inherent shutdown mechanism fails and a serious excursion can result.

The limiting safe accident for any particular reactor, defined as that accident which just fails to melt the hottest point of the hottest fuel element, must be a function of the design of that reactor. It is realized that generalization from particular studies may lead to difficulties. This is especially true for the BORAX V design, which incorporates a highly enriched, steam-cooled, superheating section either at the core center or at the core periphery. Nevertheless, the boiling core effects will shut down the reactor without any damage under certain conditions, and with only warping and distortion under other conditions.

The most direct evidence of the inherent safety of BORAX V stems from the transient tests of BORAX IV. Both reactors have about the same void coefficient ($-0.25\% \Delta k/k\text{-}\%$) and fuels with similar conductivity and specific heat. The BORAX V fuel has a 0.002-in. helium annulus between the fuel and the 0.015-in. stainless steel cladding, while the BORAX IV fuel element had a 0.013-in. lead annulus between the fuel and the 0.020-in. aluminum cladding. Thus, the thermal resistances are not greatly different. Because of the poor thermal conductivity of oxide fuels, only the fuel near the cladding transfers heat to the water during short period transients and thus contributes to the void formation shutdown mechanism. Therefore, the larger surface area associated with the larger fuel diameter of BORAX V is a favorable factor, since a greater percentage of the energy released during a transient is utilized in void formation. BORAX IV experienced an excursion at atmospheric pressure with a period of 0.083 sec, corresponding to a step reactivity increase of 0.74%, with no damage. Indeed, the total energy release to peak power was less than 10 Mw-sec. It is calculated that BORAX V with a central superheater can undergo an excursion of 75 Mw-sec from atmospheric pressure before the cermet in the superheater fuel plates begins to melt. This result is based on the heat capacity of the

cermet meat only, and it is assumed that no heat transfer occurs. It seems fairly certain that BORAX V can undergo excursions even more violent than the BORAX IV super-prompt-critical excursion without damage. (On the same basis, a 180-Mw-sec excursion from operating conditions is necessary to just melt the hottest boiling fuel rod.)

A comparison of BORAX V characteristics with EBWR characteristics indicates that the volumetric specific heats for the two fuels are comparable, and the heat conductances of the claddings are comparable, with EBWR having a thicker cladding of slightly lower conductivity. The void coefficient of BORAX V is about twice that of EBWR ($-0.23 \Delta k/k\%$ void for BORAX V as opposed to $-0.12 \Delta k/k\%$ void in a 4-ft EBWR core and $-0.067 \Delta k/k\%$ void in a 5-ft EBWR core). Hence, it is likely BORAX V will be safe under transients terminated by steam formation with periods slightly shorter than 0.083 sec (since EBWR calculates a safe period for steam shutdown of 0.057 sec).

For excursions of a shorter period, in which the thermal effects have less chance to operate, the Doppler effect in the U^{238} will be instrumental in terminating the excursion, although some damage may be done to the reactor if the period is short enough. This aspect is covered in Section VII, C, "Maximum Accident." Comparison of BORAX V with BORAX IV shows that the Doppler coefficient is greater for BORAX V. Depending on the core configuration and the temperature range considered, values of this coefficient for BORAX V from $-3.3 \times 10^{-5}/^{\circ}\text{K}$ to $-1.1 \times 10^{-5}/^{\circ}\text{K}$ have been calculated, compared to a value of $-0.7 \times 10^{-5}/^{\circ}\text{K}$ for BORAX IV.⁽²⁹⁾ Even with this strong Doppler coefficient, however, it is possible to seriously damage the reactor with a rapid addition of sufficient excess reactivity.

B. Possible Hazards

1. Hazards of Superheater Operation

a. Loss of Coolant

Since a superheater fuel assembly in BORAX V is essentially gas cooled, and also has a low heat capacity because of its thin-plate construction, the primary hazard associated with superheater operation is loss of coolant-steam flow followed by rapid heating and possible meltdown with attendant release of radioactivity. For instance, at a reactor power of 20 Mw and no steam flow, the maximum superheater fuel element temperature rises from 1200°F to the melting point, 2600°F, in about 5 sec.

Loss of steam-coolant flow can be caused by manual closing of a steam stop-valve, closing of a back-pressure-control valve as a result of loss of instrument air, or control-power failure. (The back-pressure-control valves are designed to fail-close in order to contain steam, water,

or radioactivity within the reactor vessel.) Rupture of the exit steam manifold inside the reactor vessel would balance pressure across the superheater and stop coolant flow.

The superheater fuel elements are protected against damage from loss of coolant steam by: (1) a power-failure scram; (2) a low-instrument-air-pressure alarm and scram; (3) a low-steam-flow alarm set at 4000 lb/hr, with scram set at 2000 lb/hr; (4) a high superheated-steam-temperature alarm set at 875°F; (5) an emergency air and power system to insure operation of essential valves and equipment, and, if possible, by (6) a high superheater-fuel-element temperature alarm set at 1250°F, with scram set at 1300°F.

Steam flow through the superheater may also be stopped by inadvertent flooding of the superheater fuel assemblies. If the superheater is flooded by raising reactor-vessel-water level to the top of the superheater-fuel-assembly inlets, the reactor is protected by a high-water-level alarm and scram and reactor water-level indicators. Flooding by opening the superheater-flood valve will cause the flood-valve-open scram to operate. If this interlock fails, the reactor power can rise because of the positive flooding coefficient of reactivity. In this case the power-level-trip safety circuits scram the reactor. Once the superheater is flooded and shut down, it is cooled by either boiling or natural circulation of the flood water.

Any of the aforementioned alarms give the reactor operator a warning to shut down and open the superheater-vent valve to keep the superheater fuel elements cool. On any scram the superheater-vent valve opens automatically to a preset position if air is available. Further means of insuring operation of the superheater-vent valve is provided by an emergency air compressor powered by the emergency generator, and by stored air in the air compressor tanks. Manual operation of this valve through a reach rod is also possible.

b. Flooding Superheater

Programmed flooding of the superheater is performed after shutdown, with control rods fully inserted and at a time when the temperature difference between the reactor water, the flooding medium, and the superheater fuel plates is low, to give a minimum thermal shock to the fuel plates. In this condition, even though flooding has a positive reactivity effect, the reactor remains subcritical due to the large amount of negative reactivity in the control rods. To prevent flooding during operation, an interlock shuts down the reactor if the superheater-flood valve is opened while the control rods are not fully inserted. Also, to inhibit inadvertent flooding of the superheater, the manual cable trip on the superheater flood valve and the reach-rod operator on the manual flood valve are secured with frangible padlocks. There is a high reactor-water-level alarm and scram

which prevents water from flowing over the tops of the superheater inlets in the steam dome. Nevertheless, an analysis has been made to determine the effect on reactivity of flooding the superheater during operation.

Since the results of flooding the central superheater are more severe than those for flooding the peripheral, only the central superheater will be discussed. Two cases were considered. In the first case, flooding was assumed to take place when the reactor was operating at pressure at a power level such that there was a 0.15 void volume fraction in the boiler moderator. Flooding the central superheater under this condition adds 2.6% $\Delta k/k$. In the second case, flooding was assumed to occur with the reactor critical at zero power at the operating temperature and with no voids in the boiler moderator. In this case, flooding the central superheater adds 1.8% $\Delta k/k$. In spite of the lower reactivity addition, the second case was considered to be the more serious, since it is doubtful that flooding could actually occur under the conditions of the first case. The high fuel plate temperatures in the superheater in the first case when the reactor is operating would cause rapid vaporization of the incoming water and thereby delay and perhaps prevent flooding from taking place.⁽¹²⁾

In the unlikely event of failure of the interlock which prevents flooding the superheater while the control rods are out, it would take three seconds to flood the active region of the central superheater if water entered all six of the first pass fuel assemblies at the same time. This would result in a 0.6% $\Delta k/k$ -sec ramp rate of reactivity addition. To reduce the effect of a failure of the high-water-level alarm system, one of the superheater-assembly-inlet risers has been shortened by 2 in. Thus, only one of the 12 central superheater assemblies becomes flooded initially, introducing a step addition of 0.15% $\Delta k/k$. About 5 sec later the remaining assemblies begin to flood, but by this time the reaction should have shut itself down, or the safety trip circuits or reactor operator should have shut the reactor down.

c. Excursions and Chugging

Assuming the reactor is either "chugging" (violently oscillating in power) or the rapid addition of a large amount of reactivity causes an excursion, it can be postulated that the superheater fuel elements may overheat and melt. This results from the time lag in flow of additional cooling steam from the boiler behind the power increase in the superheater. One Mw-sec of reactor energy release causes an 8.3°F rise in an average superheater fuel plate if none of the energy is transferred to steam. Therefore, a 105-Mw-sec excursion from full-power operating conditions, taking into account the radial and axial power distributions shown in Table I, results in only a 2450°F maximum fuel-plate temperature.

Protection against this accident is given, in addition to the negative power coefficient of reactivity, by the time delay required to reach the melting temperature. This allows the power-level scrams and, if available, the high superheater-fuel-element-temperature scram to operate.

d. Superheater Shutdown and Startup

Normal superheater shutdown procedures as presented in Section V, "Reactor Operations," do not constitute a hazard since steam flow is maintained until the reactor post-shutdown decay heat is down to less than 1% of maximum power or about 0.4 Mw. If the steam flow is stopped below this power, cooling by radiation and conduction keeps the fuel-plate temperature below 1200°F. In fact, radiation cooling will limit the temperature at 3.5 Mw to 2450°F. At this temperature, below the melting point of Type 304 stainless steel, severe scaling, distortion, and damage to fuel elements will probably occur, but no radioactivity should be released unless this condition persists for some time.

The superheater startup operation takes place either immediately after shutdown, when steam is still flowing, as in the case of a spurious scram, or long enough after shutdown to allow reactor decay heat to reach a low level. Another postulated danger point might be a startup operation with flooded superheater. Draining a flooded superheater to the condenser requires about 1 min. During this time before steam starts flowing through the superheater, the fuel elements are uncovered and no steam is flowing, but they are adequately cooled by radiation and conduction alone. Moreover, operating procedure is to reduce reactor power before draining the superheater, and superheater draining has a negative reactivity effect. Thus, this situation actually poses no hazard.

2. Core Melting

The BORAX V core can be made to melt and possibly to vaporize by the sudden addition of a large amount of excess reactivity. Loss of water from the core, though it will shut the reactor down, can result in core melting from decay heat under certain conditions of reactor operating history.

If the reactivity addition is sufficient to vaporize fuel elements, an explosion similar to the terminal experiment which destroyed BORAX I could result. The immediate area around that reactor was heavily contaminated with fission products and fuel element fragments, while a small radioactive cloud moved off downwind. There is no danger of concentration of fuel elements to form a second critical mass in this type of accident.

If the superheater fuel elements melt due to loss of coolant, the finely divided UO_2 dispersed in stainless steel is carried with the molten metal. The molten mixture may flow to the lower part of the core, or to a point below the core, before it is sufficiently dispersed to cool and solidify.

The boiling fuel rods are made of sintered UO_2 pellets clad with stainless steel. The melting point of stainless steel is about 2600°F , whereas that of UO_2 is about 5000°F . Because of the low conductivity of UO_2 , the temperature at the center of the fuel rod is much higher than that of the clad surface. If the rate of heat removal from the clad surface is decreased by loss of water, the cladding melts before the UO_2 , releasing some of the gaseous and volatile fission products. When sufficient cladding has melted, the pellets and fragments fall onto the lower grid in the boiling fuel assembly. If the heat from the pellets and fragments causes melting of the stainless steel grid and aluminum fuel assembly box, the pellets and fragments may fall through the holes or melt through the stainless steel of the thick core-support plate, coming to rest on the bottom of the reactor vessel. The nine re-entrant control-rod nozzles in the bottom of the reactor vessel serve to separate the mixture of stainless steel and UO_2 pellets and fragments so that, in the unlikely event that water is still present, no critical mass is formed. By this time, the UO_2 is cooled down by dispersion in the molten steel. The mixture does not have sufficient heat content to melt the re-entrant control rod nozzles or melt through the bottom of the reactor vessel.

The possibility of a metal-water reaction under core melting conditions is covered in Section VII, B, 18, "Metal-Water Reactions."

3. Cold-water Accident

Since the reactor has a negative temperature coefficient, introduction of cold water increases the reactivity. The temperature coefficient for the various cores is estimated to be about $-0.03\% \Delta k / (k)(^\circ\text{C})$. Cold water can be injected into the reactor by two postulated methods:

a. If the feed pumps are turned on with the reactor just critical at operating temperature, cold water can be injected at a maximum possible rate of 300 gpm. The minimum operating volume of water in the reactor vessel is about 1600 gal. Four minutes are required to fill the reactor vessel and the water temperature is reduced by 85°C . This results in a rate of reactivity addition of about $0.01\%/\text{sec}$ for a total addition of 2.55% . The reactor power level merely rises to a point at which the increase in temperature and steam formation nullify the increase in reactivity due to the addition of cold water.

b. The worst cold-water accident that can be postulated involves the forced-convection system. Assume that the reactor with central superheater is just critical at 600 psig, the forced-convection baffle is installed, and the forced-convection pump is off. Also assume the operating procedure to keep the forced-convection system hot has been ignored and that the low-forced-convection flow or low-auxiliary-water-pump-flow scrams, alarms, and startup interlocks and the differential

temperature interlock do not operate, so that the temperature of the water in the forced-convection system is at room temperature. (Incidentally, under this condition, the lower nozzles on the reactor vessel are severely stressed.)

When the 10,000-gpm pump is turned on with system valves open, full flow is reached in about 6 sec. The 489°F water in the core is replaced with 70°F water in 0.4 sec, giving a reactivity addition rate of 17%/sec. The superheater fuel plates begin melting shortly after onset of the excursion; if this does not shut down the reactor, then the cladding on the boiler fuel rods starts to melt soon afterwards. An analysis of the excursions caused by cold-water accidents is given in Section VII, C, "Maximum Accident".

To prevent this cold-water accident, the following interlocks are provided:

(1) A flow interlock to assure that either the 10,000-gpm forced-convection pump or the 150-gpm auxiliary-water pump is causing flow of water through the forced-convection system and reactor vessel before reactor startup.

(2) A differential temperature interlock, to prevent reactor startup if the ΔT between reactor vessel water and water in the forced-convection system is greater than 15°F. Interlocks (1) and (2) insure approximately equal temperature throughout the system prior to startup.

(3) A pump starter interlock to prevent starting of the forced-convection pump after the reactor is in operation. This prevents a cold-water accident even though interlocks (1) and (2), above, have failed.

(4) An interlock that requires the forced-convection-suction valve to be closed in order to run the auxiliary-water pump. This prevents short-circuiting of the auxiliary system water.

(5) A low-flow alarm and scram on the forced-convection system.

(6) A low-flow alarm on the 150-gpm auxiliary-water system.

Interlocks (5) and (6) above either scram the reactor or warn the operator if flow rate decreases in the forced-convection system. Cooling of this system is thus inhibited and protection is afforded against a cold-water accident or high reactor-vessel-nozzle stresses.

The maximum possible cold-water reactivity-addition rate from the 150-gpm auxiliary-water-pump is only 0.015 of the rate associated with the forced-convection pump and is not deemed hazardous.

4. Power and Air Failure

In case of loss of power from the incoming utility line, the solenoid-operated latches holding the control rods fail-safe and release. The rods are inserted into essentially their maximum effective shutdown position in about 0.25 sec.

An independent 24-volt dc battery, emergency-power system, which is considered the most reliable source of power, energizes certain critical valves.

A second emergency-power system is supplied by a gasoline-engine-driven generator which starts automatically on utility-power failure at either the reactor or the control building. Automatic transfer switches connect certain essential instruments and controls to this power source when the emergency generator reaches operating voltage.

Loss of utility power eventually results in a loss of instrument-air pressure for control-valve operation. The capacity of the air-compressor storage tanks maintains sufficient control pressure at a probable maximum bleed rate of 45 scfm for about 1 min. An emergency air compressor operating from the emergency generator supplies air to certain control valves essential for reactor containment and cooling. In case of total loss of control-air pressure, the valves which contain the reactor fail-close.

Table XVII contains a detailed list of the valves, controls, and instruments supplied by the emergency power systems.

5. Pump Failure

The most probable reason for failure of pumps is loss of power. The effect of individual pump failure is discussed in the following paragraphs:

Failure of the 10,000-gpm forced-convection pump or inadvertent closing of one of the valves in the forced-convection system stops forced-convection flow. Since the heat is not being carried out of the core by forced flow, the void content is increased, and the reactor power decreases before the fuel-element temperature rises to the point at which "chugging," film boiling, and burnout occur. Natural-circulation cooling within the fuel assemblies will remove the decay heat after reactor shutdown, in spite of the presence of the forced-convection baffle. Backup protection against this hazard is afforded by means of a low-forced-convection-flow alarm and scram.

If both reactor-feed pumps fail, the reactor continues to operate, feedwater flow ceases, reactor inlet subcooling is reduced, reactor voids increase, the reactor automatically reduces power, and water is evaporated

from the reactor. Alarms for low reactor-water level, low feedwater-pressure and high feedwater-storage-tank level notify the operator of these events. If these alarms are inoperative or unheeded and heat from the reactor continues to evaporate water, the low-reactor-water-level scram shuts the reactor down. The reactor water may still be maintained above core level, at pressure, by manual operation of the batch-emergency-feed system. The capacity of this system is sufficient to handle the most severe conditions of decay heat. Also, the boron-addition system can be used to maintain reactor water level.

Failure of the condenser main-circulating pump leads to loss of condenser vacuum. A pump-failure alarm is energized and eventually a reactor scram occurs if condenser pressure exceeds +2 psig. Backup protection is given by the condenser pressure-relief valve set at 5 psig.

Failure of the well pump stops the flow of raw cooling water to the shield, demineralizer system heat exchangers, and pump bearings. If water flow stops in the shield-cooling system, the concrete shield can heat to approximately 520°F at 40-Mw reactor power. It is probable that some spalling and crumbling of concrete in the shield may occur at this temperature. Though not hazardous, this situation is not desirable, and a high-shield-temperature alarm set at 200°F warns the operator. Other indications of cooling-water failure may come from the high demineralizer-water-temperature alarm and the high forced-convection-pump-bearing and auxiliary-water-pump-bearing temperature alarms.

Failure of either or both condensate pumps causes alarms. Condensate-pump failure causes the hotwell-water level to rise, resulting in loss of condenser vacuum. If the operator takes no action, the condenser high-pressure scram eventually shuts down the reactor.

6. Instrument Failure

Instruments which are capable of scrambling the reactor upon receipt of the proper signal also scram the reactor when electric power to the instrument fails. Similarly, most alarm circuits are designed to give alarms when the circuits do not function properly (see Table XV). Failure of the supply of high-voltage power to neutron detectors causes an alarm. The pressure-control system fail-closes to contain steam, water, and radioactivity within the pressure vessel.

7. Control Rod Failure

Experience with the control rod drives on EBWR has indicated that rod sticking does occur. Even though the EBWR drives have been modified and improved for use on BORAX V, the possibility of malfunction of one or two rods or drives at some time cannot be ruled out. Another cause of

rod sticking may be swelling of the rod resulting from a leak in the stainless steel cladding of the Boral section. Because of the higher temperature inside the rod under operating conditions, any water which has leaked in will be converted to steam at a higher pressure than the water outside the rod. This will cause the rod to swell, and the increased gap between the cladding and Boral will result in higher internal temperatures and perhaps a progressive expansion. Corrosion of the Boral in a leaking rod also causes a slow increase in rod thickness.

Prior to the first reactor startup each day, control rod drives and scram latches will be checked for proper operation, and any control rod or drive which does not function properly will be repaired or replaced. The time for each control rod to drop from the full-out position to the in position will be checked weekly. If the rod fails to drop in the prescribed maximum time (0.5 sec), the rod will be repaired or replaced.

Standard operating procedure requires a large negative shut-down reactivity in the core before repairing, removing, or installing control rod drives (see Section V, A, "Reactor Operations"). Installation and removal of a control rod drive and seal does not require moving of the control rod. The excess reactivity shall be adjusted so the reactor cannot be made critical under any condition by the withdrawal of a single control rod (see Section V, A, "Reactor Operations").

Should a failure of electrical circuitry occur which would cause control rods to drive out (the direction causing an increase in excess reactivity), two failures would have to occur simultaneously in order to produce a rate of reactivity addition greater than $0.05\%/sec$.

a. If a fault occurs within a single intermediate rod motor or controller, causing it to run at the high speed rather than the low speed, the rate of reactivity addition would be less than $0.05\%/sec$. The maximum speed for driving a rod is set for less than this rate of reactivity addition by gearing. For a similar failure of the center or an outside rod drive motor or controller, the rate of reactivity addition would be less than $0.05\%/sec$.

b. If two intermediate rods run at high speed or if the center and one intermediate rod run at the high speed simultaneously, the rate of reactivity addition would be $0.1\%/sec$. This would require two simultaneous failures in the control circuits or motors.

c. When an operator is operating the intermediate rods as a group, if a failure occurs, in which one rod moves at the high speed rather than the low speed, the maximum rate of reactivity addition would be $0.0875\%/sec$. This would require a control or motor circuit failure and the failure of the operator to observe the change in speed on the position indicator.

d. When an operator is operating the intermediate rods as a group, and in the unlikely event that all 4 rods operate at high speed rather than low speed, the rate of reactivity addition would be 0.2%/sec. A minimum of 3 simultaneous control circuit failures would be required to produce such an addition rate.

In the event that reactivity is added at this rate (0.2%/sec) the high-flux safety trip circuits or the reactor operator should shut the reactor down.

If not enough control rods can be inserted to shut down the reactor, or if the reactor has been experimentally or inadvertently loaded to some reactivity in excess of that available in the control rods for shutdown, the boron-addition system may be manually operated, remotely or locally, to shut down the reactor by adding a boric acid solution into the reactor water.

8. Change in Power Demand

Since the power produced by BORAX V is disposed of locally, i.e., through either the water rheostat or atmospheric venting of steam, any change in power demand is normally made by the operators.

An increase in electrical load opens the turbine throttle valve. The turbine bypass valve closes, but, if the reactor power is not adjusted to compensate for the increased demand, the steam-line pressure is reduced. The reactor pressure should not be affected, however, because of the back-pressure-control valve in the main steam line which senses and controls reactor vessel pressure. Since opening of the vent-steam back-pressure-control valve is dependent on reactor pressure, it is not considered a load change.

A decrease in electrical load causes the turbine throttle valve to close and the turbine bypass valve to open. If the reactor operator does not adjust power, the capacity of the bypass system is exceeded and pressure increases in the steam line. In this case the excess steam is vented by the low-pressure-relief valves which act at 388 and 400 psig. A secondary effect might be overloading of the condenser, with resultant loss of condenser vacuum and a scram. The reactor upstream from the main-steam, back-pressure-control valve is protected by the reactor-vessel safety valves.

If a decrease in steam flow is accomplished by sudden closing of the vent-steam, or main-steam, back-pressure-control valve and the reactor operator does not adjust power, a positive feedback is introduced. The result is an increase in power and pressure.

Protection against this hazard resides in the high-reactor-pressure alarm, set at 625 psig, the two high-power-level scrams, and the two reactor vessel pressure-relief valves, set at 650 and 675 psig.

9. Too Much or Too Little Water in Reactor

The reactor is protected by alarms and scrams against both high and low levels of reactor water. In addition, a continuous, reliable indication of water level is furnished by a level recorder and a TV-monitored gage glass.

a. Too Much Water

If the high-water-level alarm and scram fail to operate, if the high-water-level indication is ignored, and if the reactor vessel becomes filled to the top with water, the reactor is less safe under fast excursion conditions because expulsion of water from the core would be difficult, as is the case in pressurized-water reactors. However, it is not sufficient just to fill the reactor vessel completely with water. To obtain an excursion some method must be found to add a large amount of reactivity suddenly. This is discussed in Section VII, B, 17, "Human Error and Maloperation."

In the superheating cores too much water can also flood the superheater. This is discussed in Section VII, B, 1, b, "Flooding Superheater."

b. Loss of Water

Normal removal of decay heat from the boiling fuel elements immediately following a shutdown occurs by boiling of the coolant. The steam then flows through the superheater to the condenser or to the atmosphere. Water level is normally maintained by means of the feed pumps or, in an emergency, by means of the batch-emergency-feed system.

Gradual loss of water can take place while the reactor is at power, if the feed pumps fail, and the operator takes no action, and the low-reactor-water-level scram fails. In this case, the core is slowly uncovered and, because of loss of moderator, power is reduced until shutdown occurs. The uncovered portion of the boiling fuel elements and the superheater fuel elements are cooled for a while by steam, but as the water level nears the bottom of the core, the steam flow reduces and the core can eventually melt.

In the event of a rupture of the pressure vessel or its connecting piping below core level, the water is lost from the reactor vessel and the reactor shuts down immediately, but sufficient decay heat can melt the reactor.

10. Leaks from External System

The predominant radioactivity in the steam system, the forced-convection system, the reactor demineralizer-system supply and the preheat system is due to the fast-neutron irradiation of O^{16} to form N^{16} . The short (7.4-sec) half-life of N^{16} limits accumulation of this activity in the vicinity of a leak.

The forced-convection, reactor demineralizer, and preheat systems contain N^{16} and other radioactive materials. These systems are located in the subreactor room, access shaft, pipe trenches, or the equipment pit, all of which are kept at a negative pressure and ventilated by the reactor-pit-exhaust system. Thus, if power is available to the pit-exhaust blowers, the radioactivity is confined below floor level in the reactor building and exhausted through AEC-type filters up a stack.

Since the mass flow rate of steam is low compared to pressurized water, the amount of N^{16} carried over in the steam is correspondingly lower than the carryover in a pressurized-water system. A leaky steam line can release radioactivity into either the reactor building or the turbine building. Even if all the water in the reactor vessel flashes to atmospheric pressure, the integrated exposure from the N^{16} is not lethal for operating personnel in the buildings.

Leakage of condenser cooling water into the condenser increases the amount of impurities in the condensate and tends to increase the radioactivity of the reactor water and steam. However, the reactor demineralizer system and the condensate filter and demineralizer are designed to maintain water purity.

Area radioactivity monitors in both the reactor and turbine buildings sound an alarm when an undue amount of radioactivity is released.

11. Fuel-cladding Failure

The fuel-cladding material, Type 304 stainless steel, is considered corrosion-resistant to water and steam at the boiling-fuel-rod surface temperature of 515°F maximum, and it also has excellent corrosion resistance to superheated steam at the maximum superheater-fuel-plate temperature of 1200°F. Stainless steel was chosen as the first cladding material for the boiling fuel elements to insure maximum reliability. Aluminum alloy X-8001, used in the core shroud and considered for future use as a cladding material on the boiling fuel rods, is also deemed feasible for use even though it has a higher corrosion rate than does stainless steel. However, in the fabrication of thousands of fuel rods and plates, it cannot be guaranteed that all welds are perfect, that all bonds are sound, or that cladding is completely free of defects. Fuel-cladding failure may sometimes occur.

Submelting cladding failure on the stainless steel- UO_2 cermet superheater-fuel plates would result in only a minor amount of radioactivity release. Only a small amount of the dispersed fuel would be exposed by even a complete cladding failure.

An aggravated condition could occur if cladding failure or fuel-element burnout coincided with an over-pressure accident which opens the pressure safety valves or occurred while venting steam. In either case, fission products would be immediately released to the atmosphere. Protection against an excessive pressure accident is discussed in Section VII, B, 13, "Excessive Pressure." The area monitors, air monitors, and fission break monitors should warn personnel in the vicinity of the reactor.

Fortunately, the fuel-element cladding rupture on BORAX IV has given first-hand experience on the effects of an oxide-rod cladding failure in a direct-cycle, boiling-water reactor. In the BORAX IV incident, about one-third of the fuel assemblies developed small leaks in the ends of the elements. The fuel elements were made of uranium oxide-thorium oxide pellets, lead-bonded in aluminum tube-plates. Essentially, only fission products which were gaseous or had gaseous precursors were identified; Xe^{138} , Kr^{88} , and Kr^{89} accounted for the major portion of radioactivity released from the fuel elements. Only trace amounts of products, such as Mo^{94} , with nongaseous precursors were detected in the reactor water, indicating that these were retained in the ceramic fuel. The major contaminant released from the steam system was Cs^{139} . Long-lived radio-nuclides that were identified in trace amounts in the turbine and condenser after operation were Ba^{140} , La^{140} , Sr^{90} , Y^{90} , Sr^{89} , and Cs^{137} .

The major release of radioactivity to the atmosphere was via the condenser-air-ejector exhaust, even though the exhaust gases had passed through an AEC-type filter. A summary of radioactivity release is given in Table XVIII.

Table XVIII

LOCATION AND AMOUNT OF RADIOACTIVITY IN BORAX IV
FUEL-CLADDING RUPTURE

Reactor Power, Mw	Estimated Air-ejector Gas Activity, curies/min		Reactor Building Steam Line, mr/hr	Air-ejector Filter, mr/hr	Main Door of Turbine Building, mr/hr	Main Door of Reactor Building, mr/hr	500 ft Downwind,* mrep/hr
	Xe^{138}	Kr^{88}					
2.4	3.45	0.7	500	30,000	20	20	120
6	8.64	1.8	500	>50,000	27	30	-

*Average on three high-volume air samples, representing approximately 7250 ft³ of air sampled.

During operation the reactor and turbine buildings were entered for brief periods by personnel wearing protective clothing and independent breathing apparatus; no excessive exposure occurred. Within 24 hours after shutdown the radioactivity had returned to background levels in these buildings.

Since BORAX V is operated from a remote location a fuel-element rupture is not considered hazardous. Fission-product monitors reading steam and reactor-water activities give an alarm in case of cladding failure. Personnel in the reactor, turbine, and control building are also warned of radioactivity release by local area monitor alarms.

12. Startup Accident

A low neutron background in a freshly loaded reactor can result in a dangerous situation, because knowledge of the multiplication or period is lacking. As is customary in startups of reactors or critical facilities, a strong neutron source is always used in BORAX V. A permanent, pre-irradiated Sb-Be source, maintained at strength by operation, is installed in the core. Both compensated and uncompensated ion chambers and counters are part of the nuclear instrumentation.

The rate of reactivity addition by control-rod withdrawal is limited to 0.05%/sec. With this rate of addition, the water begins boiling as soon as the fuel elements reach a temperature slightly higher than the saturation temperature of the water. The negative void and temperature coefficient automatically compensates for the reactivity added by rod withdrawal. The power-level-trip circuits prevent the reactor from reaching the burnout heat flux.

13. Excessive Pressure

Since an increase in pressure in a boiling water reactor can lead to a power increase and a still further increase in pressure, mechanisms for limiting the pressure are particularly important. In BORAX V, protection against excessive pressures is furnished by a high-pressure alarm and scram set at 625 and 640 psig, respectively, together with dual safety valves set at 650 and 670 psig. Opening of these valves will reduce the power level of the reactor while flashing occurs.

14. Improper Charging of Fuel

Calculations have shown that a minimum size of critical configuration composed of 49-rod boiling assemblies would be formed with 11 such assemblies under cold conditions. The calculations are based on a cylindrical geometry and give as reactivity worths for boiling assemblies

in the central and peripheral positions 6.96% and 1.9%, respectively. Similar calculations have shown that it will require 33 central superheating assemblies to become critical under cold and flooded conditions. Only 16 such assemblies are available.

The reactivity worths of peripheral assemblies in the three full-size (60-assembly) cores are shown in Table VII. Although calculations have predicted that 9 closely packed peripheral superheater assemblies could become critical under cold, flooded conditions, the two types of superheater assemblies are distinctly marked.

Loading always takes place with the reactor at atmospheric pressure and with the reactor well below critical. As set forth in Section V, "Reactor Operations," loadings are checked for anticipated effect, by at least two qualified persons before any change, and during the change for proper placement of fuel in the core. Neutron monitors are always used during loading operations.

15. Earthquake

There exists some risk of seismic activity at the National Reactor Testing Station. The BORAX V reactor is located in a region which The Pacific Coast Uniform Building Code designates as a Zone 2 area. Therefore, the building structures and plant are designed for Zone 2.

The most serious consequences of a severe earthquake might be to rupture the reactor vessel or piping and jam the control rods. In case water is lost from the reactor vessel during operation, the reactor shuts down, but might subsequently melt from decay heat. If control rods become jammed and the core is still immersed in water, the boron-addition system can probably be used to shut down the reactor. The concentration of fuel in the core of the reactor by an earthquake is unlikely; thus there is little danger of an increase in reactivity.

It is of interest to note that the recent major earthquake of August 19, 1959, centered near West Yellowstone, Montana, was severe enough to be felt at the NRTS. However, no damage was done to any reactor or installation.

16. Fire

The reactor, its building, and equipment are essentially fire-proof. Hazardous amounts of combustible materials are not allowed in the building.

17. Human Error or Maloperation

a. Reactivity Additions

The fuel in the reactor can be made to melt by the sudden addition of an arbitrarily great amount of excess reactivity. This can be done by sudden injection of fissionable material or sudden withdrawal of a poison. As mentioned in Section VII, B, 14, "Improper Charging of Fuel," it is difficult suddenly to add a large amount of fuel to BORAX V. Sudden withdrawal of poison can be accomplished by removing control rods from the top of the core. To accomplish this, the control-drive mechanisms and seal housing have to be removed from the bottom of the vessel and the control rods have to be disconnected from the extension shafts. As an alternative, the rods can be actuated from the subreactor room, after disconnecting the control-drive mechanisms, by some sudden means which overcomes reactor pressure, if present. Any one of these methods seriously endangers the personnel involved.

The excess reactivity is adjusted so that the reactor cannot be made critical under any condition by the withdrawal of a single control rod (see Section V, A, "Reactor Operations").

b. Maloperation

An unsafe condition caused by human error could most simply be accomplished by opening the reactor-vessel-blowdown valve, control-rod-drive-seal-blowdown valves, or the forced-convection-system-drain valve and releasing the water from the vessel, with a possible resulting core meltdown.

One of the operating staff who has the necessary keys and knowledge can start up the reactor. Causing an excursion by means of normal control-rod withdrawal alone is ineffective because of the limited rate of reactivity addition available. If the reactor vessel safety valves are "gagged," an excessive pressure excursion can be set up. If the proper interlocks and scrams are bypassed, a maximum cold-water accident can be set up by using the forced-convection system.

When the forced-convection baffle is installed and if the low forced-convection-flow scram is bypassed, then the reactor can be started up with no adequate provisions for recirculating cooling water. Under this condition operation at high power would probably cause boiling-fuel-rod burnout.

18. Metal-Water Reactions

The importance of a metal-water reaction is derived from the additional heat liberation possible during a reactor accident. Two types of

reactor accidents are generally considered as being capable of producing the requisite conditions for metal-water reactions: one is the loss-of-coolant accident, which may result in core melting by decay heat after the reactor has shut itself down; the other is a power-excursion accident, in which an uncontrolled increase in neutron flux causes an exponential increase in power level, resulting in the possibilities of melting, vaporization of fuel elements, and an explosion. Of the two accidents mentioned, the power excursion is considered much more likely to initiate metal-water reactions because of the higher temperatures and vaporization achieved.

Since the fuel material in BORAX V is UO_2 , which does not react, the only metals available for reactions are the cladding and structural materials. The cladding is Type 304 stainless steel for the first cores, and possibly aluminum alloy for some future boiling fuel elements. The core structural materials are Type 304 stainless steel and X-8001 aluminum alloy.

The superheater may be regarded as the most vulnerable region of the BORAX V reactor to be subjected to a chemical reaction of this kind. This statement is substantiated by consideration of a higher stainless steel operating temperature, a large ratio of surface to volume, and the possible scavenging action of the steam. Available evidence, however, does tend to establish the stainless steel-water reaction as having the lowest energy release of all common reactor materials, yielding only 256 Btu/lb of metal.

Some information on this reaction may be found in the literature.^(13,14) Of particular significance to the present problem, however, is the correlative work being conducted by the Chemical Engineering Division, ANL.

a. TREAT Experiments

Recently, in-pile results have been obtained on the reactions of molten aluminum and stainless steel^(15,16,17) in water. For this work, mixed-oxide-core fuel pins contained in a water environment in an autoclave were exposed to neutron bursts in the TREAT reactor operated by the ANL Idaho Division. The extent of reaction is determined by measuring the hydrogen produced. This method gives a realistic test in that meltdown of the fuel specimens is achieved by fission heat in a transient on a given period. Examples of the data obtained to date on stainless steel-water reactions are given in Table XIX. Three types of fuel samples have been investigated: ceramic core, cermet core pins, and cermet plates. For all, the fission energy generated in the enriched uranium oxides served as a source to heat and melt the metal. Figures 52 and 53 give typical oscillograph records of metal-water transient experiments. Figure 52 shows that for a stainless steel-jacketed, mixed-oxide-core fuel pin there is a thermal lag of about one sec.

Table XIX

SUMMARY OF IN-PILE DATA ON STAINLESS-STEEL WATER REACTIONS

	CEN Transient Experiment Number					
	21	24	48	50	51	52
Fuel Pin Material						
Clad	SS	SS	SS	None	None	SS
Core	Oxide**	Oxide**	Oxide**	SS-UO ₂ *	SS-UO ₂ *	SS-UO ₂ *
Environment of Pin	In H ₂ O	Above H ₂ O	In H ₂ O	In H ₂ O	In H ₂ O	In H ₂ O
Reactor Characteristics						
Burst Energy Mw-sec	420	560	460	510	435	512
Peak Power Mw	765	1705	1405	2850	2770	2880
Period msec	78	65	72	50	50	49
Thermal Neutron Dose, nvt (gold foil activation)	6.5×10^{14}	10.5×10^{14}	5.70×10^{14}	6.75×10^{14}	6.0×10^{14}	6.93×10^{14}
Core Fission Energy Input, cal/g (Mo ⁹⁹ analysis)	--	363	--	--	--	--
Per Cent of Metal Reacted with Water	0	1.4	4.2	9.6	6.6	10.2
Appearance of Fuel Pin after Transient						
Clad	No Change	Melted	Ruptured Partially Melted	--	--	Melted
Core	Cracked	Fragmented, Partially Fused	Fine Particles	Melted, Fine Black Particles	Melted Fine Black Particles	Melted
Peak Surface Temperature of Pin °C	>140	>640	--	>1500	>1500	500
Peak Pressure Rise psi	10	12	--	100	960	0

* 90 w/o SS-304 10 w/o UO₂ 93% enriched

** 81.51 w/o ZrO₂ 9.06 w/o CaO 8.69 w/o U₃O₈ (93% enriched), 0.74 w/o Al₂O₃

Test Conditions Water initially at 25°C under 20 psia of helium pressure

Fig. 52

Treat Transient Experiment with a Stainless Steel-clad, Oxide-core Pin (CEN-24)

Fig. 53

TREAT Transient Experiment With a Stainless Steel-UO₂ Cermet Sample (CEN-50)

Observation of the pin after the transient showed (see Fig. 54) that the central portion of the jacket had melted and the oxide was completely ejected from the pin through a hole. Because of the steep axial temperature gradient, the heavy end plugs did not melt.

1/2"
Scale

Fig. 54

Sample of Stainless Steel-clad, Oxide-core Pin After Transient (CEN-24)

Figure 53 gives a record of an experiment with a stainless steel-urania cermet. In this run, the fuel pin melted and 9.6 per cent of the metal reacted with the water. Figure 55 shows the appearance of the fuel after it was subjected to the TREAT burst.

Fig. 55

Sample of Stainless Steel- UO_2 Cermet
After Transient (CEN-50)

Upon comparison of the percentage of metal reacted with the water during the in-pile meltdowns with the corresponding changes in geometry of the fuel pins, it was evident that the physical changes are more extensive than the chemical changes. In the destructive meltdown runs in which extensive damage to the fuel and particle formation occurred, from 0.3 to 10.2 per cent of the metal was observed to react with the water.

It is of interest, for purposes of orientation, to make some comparisons of energy of these TREAT metal-water tests with the final destructive test of BORAX I. The integrated power of the final BORAX I excursion was 135 Mw-sec on a 2.6-msec period.⁽¹⁸⁾ The core of BORAX I consisted of plates containing a 13.3 w/o uranium-86.7 w/o aluminum alloy, clad with aluminum. The integrated power of some of the

metal-water meltdowns in TREAT was about 560 Mw-sec with periods as short as 49 msec. Thus, on an overall basis the TREAT transients were more energetic than the final BORAX I test, but on a longer period. However, a comparison on a specific energy basis is of more general interest. With 188 lb of metal in the BORAX I core (as estimated from ANL-5211),⁽¹⁹⁾ the 135 Mw-sec corresponds to 377 cal/g of core. Data from direct fission product (Mo^{99}) determinations of the irradiated pins and calculations of nvt_{th} from gold-foil activation have shown that for the TREAT metal-water tests on the particular oxide and cermet core fuel samples, there is approximately a one-to-one correspondence between the energy of the TREAT burst in Mw-sec and the fission energy absorbed by the fuel pin in cal/g. Thus a 500-Mw-sec transient would be equivalent to an energy input of 500 cal/g. This is 1.6 times as much energy as the 377 cal/g for the BORAX I excursion. Thus, it is evident that the in-pile metal-water tests reported herein represent rather severe reactor transients.

b. Conclusions

In-pile metal-water meltdown experiments of an energy comparable to the final destructive BORAX I transient have shown that, for aluminum, one would expect about 1% metal-water reaction for oxide cores clad in aluminum, and about 4% reaction for the aluminum in close contact with uranium dioxide in cermet fuel pins. Since for BORAX V the aluminum is only in the structural members and therefore not in direct contact with the fuel, it is likely that the lower number of 1% would apply in the event that aluminum somehow became molten (660°C) and did not attain temperatures greater than 1200°C , at which the vapor-phase reaction becomes significant. For the case of stainless steel, the data again indicate that the metal-clad oxide-core specimens give less reaction than the cermet fuel pins. Up to 10% of the stainless steel can react with water as a result of bursts having a specific energy input of approximately 500 cal/g of metal. For these observed extents of reaction for aluminum and stainless steel, the chemical energy release to be expected in a reactor excursion is less than 8% of the energy of the excursion.

It is appropriate to emphasize that more work remains to be done on the metal-water hazards problem, particularly with respect to aluminum and stainless steel in their application to BORAX V. Condenser-discharge studies have demonstrated that pressure is an important variable in determining the extent of reaction of aluminum wires with water at high temperature.⁽²⁰⁾ It is possible that the initial pressure is also significant in stainless steel-water reactions. Thus far, the in-pile, metal-water meltdown studies have been conducted at an initial pressure of 20 psia as compared to the 600-psi operating pressure of BORAX V. Plans for high-pressure tests in TREAT are in progress.

Another factor which might be pertinent is the geometry of the stainless steel-uranium dioxide fuel. In BORAX V the fuel elements in the superheater section are plates rather than cylinders. The plates, having a higher ratio of surface to volume, might be expected to give increased metal-water reaction for a given burst energy. The presently available data on cermet plates is presented in the last two columns of Table XIX. These data indicate that an increase in the surface-to-volume ratio has only a minor effect on the amount of the reaction.

Calculations based on the assumption of completely melted superheater fuel plates and a 10% stainless steel-steam reaction were made to obtain an approximation of possible energy release for comparisons with the associated nuclear excursion energy discussed in Section VII, C, "Maximum Accident." The results as given in Table XX emphasize the relatively minor energy release to be expected from this source.

Table XX

COMPARISON OF SS-WATER REACTION ENERGY WITH
NUCLEAR-EXCURSION ENERGY IN BORAX V

Core Type	SS in SH Fuel Plates, lb	Amount of Metal Reacting, lb	Chemical Energy Release,		Excursion Energy to Melt SH Fuel Plates, Mw-sec
			Btu	Mw-sec	
Central SH	175.5	17.6	4506	4.8	380
Peripheral SH	226	22.6	5786	6.1	500

c. Hydrogen Reaction⁽²¹⁾

As a result of metal-water reactions, hydrogen gas (H_2) is released. The hydrogen gas is generated under water, in which there is essentially no oxygen gas present. Therefore, it is assumed that any hydrogen explosions which occur will do so after explosions resulting from fission and/or chemical energy releases. Consequently, the fission and/or chemical energy explosions and hydrogen explosions are not simultaneous, but occur sequentially.

A hydrogen explosion will occur only when H_2 is combined with air into an explosive mixture and is ignited. Although the hydrogen-air mixture immediately around the reactor shortly after the excursion may be within the explosion range, it will be accompanied by a large amount of water vapor. This water vapor will tend to reduce both the possibility and severity of explosion. The occurrence of an explosive mixture around

the reactor will be a transient condition, and only a small fraction of the hydrogen would be contained in an explosive mixture with air at any particular instant. Therefore, while the potential energy of the hydrogen evolved from a maximum accident is nearly equal to the initial excursion energy, the probability of its detonation in a single explosion is remote. Any reaction which does occur would probably consist of small areas of inflammation and/or detonation surrounding the reactor.

19. 17-4 PH Stainless Steel Components

The Belleville springs, control rod extension shafts, and latch screw, nut and washer, which are made of 17-4 PH stainless steel, are heat treated at 1150°F for 4 hours and air cooled to give a hardness of approximately Rockwell C-33. This heat treatment is generally accepted as producing satisfactory resistance to stress corrosion. Components made of 17-4 PH have been given a thorough visual inspection and the following non-destructive tests: Belleville springs - radiograph and magnaflux; hold-down latch components - magnaflux; control rod extension shafts - ultrasonic and magnaflux. It is believed that these components are completely reliable.

However, even if the Belleville spring, which holds down a core structure, should fail completely, the core structure can move upwards, under forced-circulation conditions, only $\frac{1}{4}$ in., and it moves onto the poison sections of the control rods. The resulting reactivity change is negative.

If a control rod extension shaft breaks inside or outside the vessel and the broken shaft ends become offset enough to slip past each other, it is possible for the control rod to fall into the core. This decreases the net reactivity in the system. If an extension shaft breaks within the seal, leakage may be increased slightly and the broken end of the rod might hang up in the seal rings preventing down motion of the rod. However, reactor pressure will hold the rod in place. Even under the maximum hydraulic lifting loads than can be imposed by the forced-circulation system on a small core, a control rod with an extension shaft broken inside the vessel will not lift.

There are two hold-down latches bearing on each hold-down box and cell of boiling fuel assemblies. Complete failure of a single hold-down nut or screw allows no movements of the fuel. In the unlikely event of the failure of two hold-down nuts in latches clamping the same cell, the fuel assemblies in that cell can move upward about 1 in. under the highest hydraulic loads. If two hold-down latch screws for the same cell fail completely under maximum hydraulic load conditions, the boiling fuel assemblies in the cell may be ejected from the core. Any movement of the fuel out of the reactor should result in a net decrease in reactivity.

C. Maximum Accident

An unlikely chain of events (described in Section VII, B, 3, "Cold-Water Accident") could lead to rapid injection of cold water by the 10,000 gpm forced-convection pump into the hot, boiler region of a critical core. The reactivity additions and ramp rates of reactivity addition presented in Table XXI imply that this "cold-water accident" might lead to a severe power excursion. Such an excursion should initiate a scram signal from the high-flux and short-period instrumentation.

Table XXI

COLD-WATER ACCIDENTS

Core	$\Delta T, ^\circ K^{a,b}$	$\Delta \rho, \%^c$	$\Delta t, \text{sec}^d$	$A = \frac{\Delta \rho}{\Delta t}, \frac{\%}{\text{sec}}$
BC	234	6.87	0.49	14.0
BC	67	3.16	0.49	6.45
BC	21	1.32	0.49	2.69
BC	9	0.62	0.49	1.27
CSH	234	6.68	0.40	16.7
CSH	67	3.07	0.40	7.7
CSH	21	1.28	0.40	3.2
CSH	9	0.60	0.40	1.5
PSH	234	6.90	0.37	18.6
PSH	67	3.17	0.37	8.6
PSH	21	1.33	0.37	3.59
PSH	9	0.62	0.37	1.68

^aCore is initially at 527°K with boiler unvoided and superheater unflooded.

^bTemperature of water injected into boiler is 527°K - ΔT . Assumption is that no mixing occurs between incoming cold water and hot water already in pressure vessel.

^cReactivity increases ($\Delta \rho$) for boiling core (BC), for central superheater core (CSH) with $\Delta T = 234^\circ K$, and for peripheral superheater core (PSH) with $\Delta T = 234^\circ K$ are from RE-6 calculations summarized in Figs. 27, 28 and 29. Remaining $\Delta \rho$ values were obtained by assuming that $\Delta \rho$ vs ΔT has the same shape for all three cores.

^d Δt is time required for cold water to reach top of boiler. Assumption is that pump starts from zero flow to full flow linearly in 6 seconds.

Tests of the control-rod drives (see Fig. 40) indicate that significant insertion of the control rods will take place within less than 0.3 sec after the scram signal is initiated. Since an excursion is squelched almost immediately when reactivity is reduced below prompt critical, it is possible that the high-flux and short-period scrams will prevent a dangerous power excursion. In case these scrams fail, then the excursion may run its course, limited only by the inherent shutdown mechanisms of the core. The main purpose of this section is to calculate the energy release from such an excursion. If this release is sufficiently large to cause melting of the stainless steel jackets on the fuel, then fission products may escape from the fuel and chemical reactions involving stainless steel and water may add slightly to the energy release.

Several inherent shutdown mechanisms probably come into play during a postulated excursion of BORAX V: (1) As the fuel temperature rises, Doppler broadening of neutron absorption resonances causes increased capture of resonance neutrons in the boiler. (2) Thermal expansion of fuel causes expulsion of moderator from the core. (3) Gas from radiolytic decomposition of the moderator may form bubbles in the moderator. (4) Heat is transferred from the fuel to the moderator by conduction and direct gamma and neutron heating, causing reduction of moderator density by thermal expansion. (5) The heat transferred to the moderator causes steam bubbles. The above effects are shutdown, or negative, effects. Doppler broadening of U^{235} fission resonances in the highly enriched superheater may lead to a positive effect. It is assumed that this effect is negligibly small, that all the inherent effects in BORAX V are shutdown, or negative, mechanisms, and that these effects are active in the boiler only.

Mechanism (3) above, radiolytic gas formation, is probably unimportant.⁽²²⁾ Satisfactory quantitative comparisons of mathematical description of mechanisms (2), (4) and (5) with experiments are available.⁽²³⁾ Unfortunately, the fuel used in these experiments differs from the BORAX V boiler fuel in some important characteristics: the experimental fuel consisted of all-metal, highly enriched, plates, 0.060 in. thick, whereas BORAX V boiler fuel consists of UO_2 rods, 5% enriched, 0.345 in. in diameter with 0.015-in. metal jackets. These two types of fuel differ so drastically in their thermal and Doppler shutdown characteristics, that it appears almost hopeless to attempt to apply the experimental results to excursions in BORAX V. Unanswered questions with regard to mechanisms (2), (4), and (5) make it very difficult to obtain from theory quantitative descriptions which would be satisfactory and conservative for use in analysis of an excursion involving short periods.⁽³⁰⁾ Quantitative description of the Doppler shutdown mechanism (1) for a wide temperature range is available from Monte Carlo calculations⁽²⁴⁾ and from experiments.⁽²⁵⁾ The description from Ref. 24 is used below, since it implies a less effective Doppler shutdown than the experimental description. Because of the low thermal

conductivity of UO_2 , most of the energy generated in a rapid excursion remains in the oxide during the excursion. The assumption that all energy generated in a rapid excursion remains in the oxide during the excursion leads to a reasonable overestimate of the energy generated by the excursion.(26)

In the formulation given in Ref. 24, the resonance escape probability is

$$p(T) = \exp \left[- Z R I(T) \right] ,$$

where T is the temperature in $^{\circ}\text{K}$,

$$Z = \frac{N_{\text{ox}} V_{\text{ox}}}{N_{\text{H}_2\text{O}} (\xi \sigma_s)_{\text{H}_2\text{O}} V_{\text{H}_2\text{O}}} ,$$

and

$RI(T)$ = effective resonance integral in barns

$$\begin{aligned} &= RI(T_0) \exp \left[\alpha (\sqrt{T} - \sqrt{T_0}) \right] \\ &= \left[RI(T_0) \exp (-\alpha \sqrt{T_0}) \right] \exp (\alpha \sqrt{T}) . \end{aligned}$$

The value of α depends on the effective surface-to-mass ratio DS/M , where D is a Dancoff correction factor. For BORAX V at 527°K with 0% void in the moderator, $D \cong 0.66$ and $DS/M \cong 0.32 \text{ cm}^2/\text{g}$, so that $\alpha = 0.0066(^{\circ}\text{K})^{-1/2}$ by Ref. 24. Under these conditions for BORAX V boiling fuel rods in a square lattice on a 0.50-in. pitch, $Z = 0.0129 \text{ barn}^{-1}$. Since the T_0 in $RI(T_0)$ may be treated as a temperature used in calculating Doppler effect only, apply the formula⁽¹⁰⁾

$$RI(T_0) = \left[11.0 + (24.5) \frac{DS}{M} \right] \text{ barns}$$

with $T_0 = 300^{\circ}\text{K}$ to obtain the result

$$p(T) = \exp \left[-0.218 \exp (0.0066 \sqrt{T}) \right] .$$

The Doppler shutdown term may now be written in the form

$$C \left[\frac{p(T)}{p(T_1)} - 1 \right] ,$$

where $T_1 = 527^{\circ}\text{K}$. The coefficient $C \leq 1$ takes into account the fact that this mechanism operates in the boiler region only. For the boiling core case, C is unity. RE-6 problems (discussed in Section III, A, 7, "Core Design - Reactor Physics") reveal that the effectiveness of boiler fuel

for the central superheater core is 0.74 relative to the boiling core; so a reasonable value of C is 0.74 for the core with central superheater. For the core with peripheral superheater, similar reasoning yields $C = 0.85$.

Some mechanical and thermal characteristics of BORAX V cores used in the excursion calculations are listed in Table XXII.

Table XXII

SOME MECHANICAL AND THERMAL CHARACTERISTICS OF BORAX V CORES

Core	Volume of Boiler, cm^3	Weight of UO_2 in Boiler, g	Weight of UO_2 in SH, g	Weight of SS in SH Fuel Plates, g	Heat Capacity of Boiler UO_2 ,* $\text{Mw-sec}/^\circ\text{K}$	Heat Capacity of SH Fuel Plates,** $\text{Mw-sec}/^\circ\text{K}$
BC	4.72×10^5	10.79×10^5	-	-	0.360	-
CSH	3.77×10^5	8.63×10^5	0.629×10^4	7.96×10^4	0.288	0.0421
PSH	3.46×10^5	7.91×10^5	1.326×10^4	10.25×10^4	0.264	0.0559

*Heat capacity of $\text{UO}_2 = 0.08 \text{ cal}/(\text{g})(^\circ\text{K})$

**Heat capacity of stainless steel = $0.12 \text{ cal}/(\text{g})(^\circ\text{K})$

In these calculations the melting point of UO_2 is taken to be 3150°K and its heat of fusion as 59 cal/g . The corresponding quantities used for stainless steel are 1700°K and 65 cal/g . It is assumed that, during the excursion, 19% of the energy is generated in the superheater. The number of neutrons per cm^3 , n , refers to the boiler in these calculations; and $E^* = \int_0^t n \, dt$ also refers to the boiler. The power in Mw developed in the boiler is given by

$$\frac{V \Sigma_f v n}{3.3 \times 10^{16} \text{ fissions}/(\text{Mw})(\text{sec})},$$

where

V = volume of boiler region (cm^3),

Σ_f = thermal fission cross section (cm^{-1}),

v = average thermal neutron speed (cm/sec).

Use of $v = 4.11 \times 10^5 \text{ cm/sec}$, $v \Sigma_f = 2.82 \times 10^4 \text{ sec}^{-1}$, and V values from Table XXII leads to the conversion factors listed in Table XXIII. Numbers for the prompt neutron lifetime, l_p , and the effective delayed neutron fraction, $\bar{\beta}$, are also needed. The l_p was calculated by (a) the $1/v$ -poison method with the aid of codes RE-6,⁽⁴⁾ SOFOCATE,⁽⁶⁾ and MUFT-4⁽⁵⁾ and (b) a perturbation theory method with the aid of codes RE-7,⁽⁴⁾ RE-27,⁽⁴⁾ and RE-185. For the various cores, (a) gave values of l_p ranging from 19 to 29 μsec and (b) gave values ranging from 20 to 40 μsec . The four-group structure (described in Section III, A, 5) turned out to be too coarse to give

reliable values of $\bar{\beta}$ by perturbation theory. Since about 3% of the fissions occur in U^{238} , direct weighting of β_{25} and β_{28} gives a safe minimum $\bar{\beta} = 0.0067$.⁽²⁷⁾ The six-group delayed-neutron data⁽²⁷⁾ (normalized to make $\sum_i \beta_i = \bar{\beta}$) were used.

Table XXIII

CONVERSION FACTORS FOR USE IN
EXCURSION CALCULATIONS

Core	F, Mw-cm ³	G, Mw-cm ³	H, Mw-cm ³
BC	0.403 x 10 ⁻⁶	0.403 x 10 ⁻⁶	-
CSH	0.322 x 10 ⁻⁶	0.398 x 10 ⁻⁶	0.076 x 10 ⁻⁶
PSH	0.295 x 10 ⁻⁶	0.364 x 10 ⁻⁶	0.069 x 10 ⁻⁶

Let n = neutron number density in boiler (cm⁻³) and
E* = $\int_0^t$ n dt.

To get	Multiply	by
Power in Boiler, Mw	n	F
Energy Release in Boiler, Mw-sec	E*	F
Power in Core, Mw	n	G
Energy Release in Core, Mw-sec	E*	G
Power in SH, Mw	n	H
Energy Release in SH, Mw-sec	E*	H

The actual excursion calculations were performed with the help of the space-independent reactor kinetics code RE-129C.⁽²⁸⁾ This code requires, among other input, values of A and B in

$$k_{ex} = At + BE^* \quad ,$$

where

$$E^* = \int_0^t n \, dt$$

and t = time in seconds. The values of A and B can be changed by the code at given values of the time. This feature is exactly what is desired for the factor A, because (see Table XXI) A takes on a positive value until the cold water reaches the top of the core and then becomes zero.

The standard code RE-129C is not well adapted to treating the form of Doppler shutdown term derived above; therefore a slight modification (RE-129C') was written. This modification causes the code to change B at given values of E^* . The utility of this modification lies in the fact that, by use of data given above, the Doppler shutdown term can be expressed as a function of E^* . This function is presented graphically in Fig. 56.

Fig. 56

Doppler Shutdown Factor vs. E^*

$$E^* = \int_0^t n \, dt$$

(The first break in the curve occurs when the boiler oxide begins to melt, and the second break when the oxide is completely melted.) This curve is adequately fitted by a series of straight lines with the B values listed below. The B is in cm^3/sec and the E^* in sec/cm^3 .

If	$0 < E^* < 0.25 \times 10^9$,	then $B = -3.72 \times 10^{-11}$,
	$0.25 \times 10^9 < E^* < 0.586 \times 10^9$,	-3.10×10^{-11} ,
	$0.586 \times 10^9 < E^* < 1.172 \times 10^9$,	-2.64×10^{-11} ,
	$1.172 \times 10^9 < E^* < 2.343 \times 10^9$,	-2.15×10^{-11} ,
	$2.343 \times 10^9 < E^* < 3.000 \times 10^9$,	0,
	$3.000 \times 10^9 < E^* < 5.902 \times 10^9$,	-1.70×10^{-11} .

The B values listed above apply to the boiling core only. For the central superheater core and for the peripheral superheater core, the above B values were multiplied by a coefficient $C < 1$ (discussed above) to take into account the fact that the shutdown mechanism operates in the boiler region only. By use of these C values and of the numbers in Tables XXII and XXIII,

Doppler temperature coefficients of reactivity corresponding to the above B values may be calculated. These temperature coefficients range from $-3.3 \times 10^{-5}/^{\circ}\text{K}$ (boiling core, $0 < E^* < 0.25 \times 10^9 \text{ sec/cm}^3$) to $-1.1 \times 10^{-5}/^{\circ}\text{K}$ (central superheater core, $3.000 \times 10^9 \text{ sec/cm}^3 < E^* < 5.902 \times 10^9 \text{ sec/cm}^3$).

Several series of problems were run by RE-129C and RE-129C'. Probably the most significant conclusion from the RE-129C problems is the following: except for the runs with $\Delta T = 9^{\circ}\text{K}$ (in all of which the excursion was incomplete at problem termination), the total energy released in the excursion is very insensitive to initial power level and to prompt neutron lifetime. This conclusion is based on runs with $n_0 = 2.5, 2.5 \times 10^3$, and $2.5 \times 10^6 \text{ cm}^{-3}$ (an n_0 of 2.5 cm^{-3} corresponds approximately to 1 watt developed in the core) and $l_p = 20, 30$, and $40 \mu\text{sec}$. In the RE-129C' runs, $n_0 = 2.5 \text{ cm}^{-3}$ and $l_p = 30 \mu\text{sec}$ were used. In Fig. 57 are plotted some results of a typical RE-129C' problem.

Fig. 57

Excursion Following Cold-water Accident ($\Delta T = 67^{\circ}\text{K}$) in Core With Central Superheater

$$E^* = \int_0^t n \, dt$$

In Table XXIV are presented some of the numerical results of the series of sixteen RE-129C' problems. The total energy release from the excursions with $\Delta T = 9^{\circ}\text{K}$ is not significant because these excursions were far from complete at the time (5 sec) the problem was terminated. The maximum k_{ex} and the minimum period involved in these problems with $\Delta T = 9^{\circ}\text{K}$ are 0.62% and 67 msec, respectively, and these extreme values appeared before 0.5 sec. Since BORAX IV experienced an 83-msec excursion without damage,⁽²⁹⁾ it seems likely that an excursion of BORAX V

resulting from a $\Delta T < 9^\circ\text{K}$ will not be extremely dangerous. This conclusion should be all-the-more trustworthy because, for such relatively slow excursions, an appreciable fraction of the heat generated should escape from the rod and enter the moderator where it should be more effective in shutting the reactor down. In the excursions with $\Delta T \geq 21^\circ\text{K}$, almost all the energy in the excursion has been developed by problem termination time (1.5 or 3 sec). Perhaps a second or so after one of these more severe excursions begins, shutdown mechanisms other than the Doppler effect should also begin to take effect. Nevertheless, if $\Delta T \geq 67^\circ\text{K}$, it seems likely that the excursion would result in severe damage to the core if only inherent shutdown mechanisms are active.

Table XXIV
SOME RESULTS OF REACTOR KINETICS CALCULATIONS

Core	$\Delta T = 527^\circ\text{K}$ - Temp of Incoming Water $^\circ\text{K}$	Problem Termination Time sec	E^* at Prob Term Time 10^9 sec/cm^3	Fission Energy ^e Release in Whole Core at Problem Term Time Mw-sec	k_{ex} at Prob Term Time, %	Min Period, 10^{-3} sec	Time Min Period Occurred, sec	Max Neutron Number Density, $10^9/\text{cm}^3$	Max Power ^e in Whole Core, 10^3 Mw
BC	234	1.5	4.0	1610	-0.86	2.5	0.14	73	29
	67	1.5	1.1	440	-0.27	3.8	0.23	36	15
	21	3.0	0.48	190	-0.33	6.0	0.44	13	5.2
	9 ^b	5.0	0.013	5.2	+0.57	80 ^c	0.49	0.023	0.009
CSH	234	1.5	5.4	2150	-0.79	2.3	0.12	150	60
	67	1.5	1.6	640	-0.24	3.4	0.21	54	21
	21	3.0	0.68	270	-0.36	5.4	0.39	22	8.8
	9 ^b	5.0	0.00043	0.17	+0.60	81 ^c	0.40	0.00097	0.0004
PSH	234	1.5	4.8	1750	-0.92	2.2	0.11	141	51
	67	1.5	1.4	510	-0.28	3.2	0.19	54	20
	21	3.0	0.62	230	-0.42	5.2	0.35	21	7.6
	9 ^b	5.0	0.015	5.5	+0.57	67 ^c	0.37	0.027	0.01

Core	$\Delta T = 527^\circ\text{K}$ Temp of Incoming Water, $^\circ\text{K}$	Time Max Power Occurred sec	E^* to Begin Melting SH Fuel Plates 10^9 sec/cm^3	Time SH Plates Begin Melting sec	E^* to Com- pletely Melt SH Plates 10^9 sec/cm^3	Time SH Plates Melt Completely sec	E^* to Begin Melting Boiler Oxide 10^9 sec/cm^3	Time Boiler Oxide Begins Melting, sec	E^* to Com- pletely Melt Boiler Oxide 10^9 sec/cm^3	Time Boiler Oxide Melts Completely, sec
BC	234	0.147	--	--	--	--	2.34	0.47	3.00	0.51
	67	0.249	--	--	--	--	2.34	--	3.00	--
	21	0.47	--	--	--	--	2.34	--	3.00	--
	9 ^b	5.0	--	--	--	--	2.34	--	3.00	--
CSH	234	0.134	0.491 ^d	0.133	0.953	0.137	2.34	0.30	3.00	0.34
	67	0.220	0.491	0.22	0.953	0.37	2.34	--	3.00	--
	21	0.41	0.491	0.61	0.953	--	2.34	--	3.00	--
	9 ^b	5.0	0.491	--	0.953	--	2.34	--	3.00	--
PSH	234	0.125	0.712 ^d	0.13	1.38	0.23	2.34	0.32	3.00	0.34
	67	0.204	0.712	0.21	1.38	1.23 ^a	2.34	--	3.00	--
	21	0.38	0.712	--	1.38	--	2.34	--	3.00	--
	9 ^b	5.0	0.712	--	1.38	--	2.34	--	3.00	--

^a This time is not very meaningful because shutdown mechanisms not included in the calculation have probably come into play

^b Excursion is far from complete at problem termination time. See text for further discussion

^c Minimum period occurs at time at which ramp reactivity input is terminated

^d This value takes into account the facts that 2/3 of superheater power is generated in fully loaded superheater fuel plates and, only 1/2 of the heat capacity of the superheater fuel plates is in fully loaded plates. See Section III A 7 "Core Design - Reactor Physics"

^e Calculated from preceding column by use of conversion factors in Table XXIII

The methods of calculation used above probably give high values of total energy release for several reasons: (a) In the calculations it is assumed that all the fission energy generated in the boiler region remains in

the oxide fuel pellets and that all the fission energy generated in the superheater remains in the cermet-stainless fuel plates. Actually, some of this energy reaches the moderator. There it decreases moderator density by thermal expansion and by formation of steam bubbles and is probably more effective in shutting down the reactor than the above calculations indicate. (b) According to experimental results,⁽²⁵⁾ the coefficient α used above in the expression for temperature dependence of the resonance integral is too small. (c) The functional form used in the above calculations for temperature dependence of the resonance integral gives a more rapid decrease with temperature of resonance escape probability than does the functional form used in Ref. 25 to describe the experimentally determined resonance integral. (d) The power has been assumed to be uniformly distributed throughout the boiler region. (The power has also been assumed to be uniformly distributed over the superheater, except that a correction has been made for difference in power generation between fully loaded and half-loaded superheater fuel plates.) Actually the power density in the boiler is greater in regions of higher importance, and the temperature rise is greater in these regions of higher importance; therefore the actual Doppler shutdown caused by increased resonance absorption is greater than the calculated shutdown. Circumstances (a) and (d) have other implications: (a) implies that energy which reaches the moderator is not available to melt fuel, and (d) implies that local melting in regions of higher power density occurs before enough energy has been generated to raise the whole "region" (boiler oxide or fully loaded superheater fuel plates) to its melting point. It is not obvious whether this local melting begins before or after the calculated time elapses (or calculated energy release takes place) for onset of melting.

Even some of the milder excursions summarized in Table XXIV are rather severe; for example, the total energy release for the boiling core with $\Delta T = 21^\circ\text{K}$ is calculated to be 190 Mw-sec. BORAX I was destroyed by an excursion with an estimated total energy release of 135 Mw-sec.⁽³¹⁾ All of the excursions calculated above begin with the reactor hot, boiler unvoided, and superheater unflooded; and the maximum ramp rates of reactivity addition given in Table XXI range approximately from 14%/sec to 19%/sec. If the cold-water accident occurs at power (15% average void in boiler moderator),* then from Figs. 27, 28, and 29 and from Table XXI the maximum ramp rates of reactivity addition range approximately from 22%/sec to 30%/sec. For the cold-water accident beginning at power, the temperature rise in the superheater fuel plates is retarded somewhat by transfer of heat to superheated steam flowing rapidly past the fuel plates. Nevertheless, it

---

*When the forced-convection baffle is in position and the 10,000-gpm forced-convection pump is not running, a low power may cause 15% average void in boiler moderator. Operating under these conditions at even this low power is abnormal and unsafe in itself. See Section VII, B, 17, c. "Human Error or Maloperation."

seems likely that the cold-water accident beginning at power is more dangerous than one beginning at essentially zero power for the following reasons: (a) The ramp rate of reactivity addition is greater. (b) Fuel temperatures are nearer the melting point initially, so the fuel is more susceptible to damage by melting. (c) Since fuel temperatures are higher initially, the Doppler shutdown coefficient is smaller initially. A large value of the Doppler shutdown coefficient at the beginning of the excursion is important in limiting the excursion.

In summary, a brief description of the cold-water accident is now presented. Through an extremely unlikely combination of human error and failure of interlocks, alarms, and scrams, the 10,000-gpm forced-convection pump is started when the reactor is hot (perhaps at power) and the water in the forced-convection system is cold. The reactivity of the core rises rapidly as the cold water enters the boiler region from the bottom. The power and temperature of the core begin to rise sharply, and several inherent shutdown mechanisms come into play. (It is assumed that the control rods remain stationary as a result of failure of high-flux and short-period scrams.) Within less than a second, before these inherent mechanisms stop the excursion, some of the superheater fuel plates begin to melt. The molten metal may react chemically with water, run down into the interpass plenums below the core, or be blown out of the core by steam. TREAT experiments reveal that boiler oxide may be expelled from the jackets in small particles or drops. As superheater fuel moves to regions of lower importance and as boiler oxide is dispersed, the reactivity of the core decreases. Probably generation of significant nuclear power ceases soon after this stage is reached; but a steam explosion and/or exothermic chemical reactions may be under way. Release of fission products (with or without failure of the reactor vessel or steam piping) to the system and to the atmosphere can result. For further discussion of the course and consequences of the accident after generation of significant nuclear power ceases, see Section VII, E, "Evaluation of Radioactive Hazards to Surrounding Population."

D. Site

The BORAX V facility is built at the National Reactor Testing Station, Idaho, on the site of the existing BORAX installation. As shown in Fig. 58, the reactor building is located about one-half mile northwest of the control building and the EBR-I, ZPR-III, and AFSR area. This figure also shows the relative location of the several structures at the BORAX V site. Figure 59 is a map of the NRTS and adjacent areas, showing the other installations on the station. Table XXV shows the populated areas near BORAX V. The populations listed for on-site areas include construction personnel proposed for the next year.

NATIONAL REACTOR TESTING STATION

Fig. 59

Table XXV
POPULATED AREAS NEAR BORAX V

Area	Estimated Population	Direction from BORAX V	Distance, Miles from BORAX V
<u>ON-SITE AREAS</u>			
Central Facilities	800	East Northeast	$3\frac{1}{2}$
MTR-ETR	700	North Northeast	5
Chemical Processing Plant	300	Northeast	5
Naval Reactor Facil.	1,400	North Northeast	10
OMRE-EOCR	250	East	6
SPERT	140	East Northeast	7
AREA	100	East	9
Aircraft Nuclear Propulsion	200	Northeast	27
EBR-I, ZPR-III, AFSR	90	South Southeast	$\frac{1}{2}$
EBR-II, TREAT	250	East Northeast	18
<u>OFF-SITE AREAS</u>			
Atomic City	230	Southeast	12
Arco	1,600	West Northwest	17
Howe	75	North	18
Carey	100	West Southwest	55
Terreton-Montevieu- Mudlake Area	1,000	Northeast	36
Roberts	750	East Northeast	47
Rigby	2,300	East	56
Idaho Falls	33,000	East	50
Shelley	2,700	East Southeast	45
Blackfoot	7,500	Southeast	38
Pocatello-Alameda- Chubbuck	40,000	South Southeast	50

E. Estimation of Radiation Hazards to the Surrounding Population

1. Vessel Rupture

To obtain an estimate of the radiation danger to the surrounding population, it is assumed that an excursion has occurred and the integrity of the reactor system is destroyed. The manner in which radioactive materials reach the atmosphere and how far they are carried depends on the severity of the accident and the meteorological conditions prevailing at the time.

In the analysis of a maximum accident given in Section VII, C, it is predicted that the superheater fuel plates of the reactor with central superheater would be completely melted by an excursion of 380 Mw-sec and those of the peripheral superheater by an excursion of 500 Mw-sec. If 10% of the stainless steel in the superheater fuel plates enters into a chemical reaction with the steam, between 4.8 and 6.1 Mw-sec is added to the total excursion energy. This depends on the particular configuration and is calculated on the basis of 256 Btu/lb of metal reacted. Because of the low thermal conductivity of the oxide in the boiler, the heat transfer to the water is possibly slow enough to not produce an explosion. The enthalpy increase of the 12,500 lb of water within the vessel from these excursions (365,000 Btu for the core with central superheater and 476,000 Btu for the peripheral) would result in a final pressure well below the static load limit of the vessel, even with no steam leaving the system. However, since melting of fuel in the hottest part of the reactor would probably take place at an excursion energy of 180 Mw-sec, and since the heat transfer to the water from molten and almost-molten fuel rods would be very rapid, a pressure loading somewhat in excess of the static case would certainly occur.

If, on the other hand, the excursion is violent enough to melt a larger fraction of the boiling fuel rods, the evidence obtained from the TREAT experiments (see Section VII, B, 18) indicates an expulsion of the molten oxide into the water in sufficient quantity to establish a destructive pressure shock to the system. Table XXVI gives comparative values of the heat content considered available to cause instantaneous boiling (50% of boiling rods are assumed to spatter).

The energy releases given in Table XXVI are similar to those associated with a total excursion energy of the order of 1000 Mw-sec, and it has been assumed that the melting superheater has not immediately terminated the excursion, but that the power level has continued to rise for perhaps 0.1 sec more until enough heat has been generated to melt part of the boiler oxide. While the approximate values are on the pessimistic side, they do point to the fact that an explosion of sufficient violence to rupture the system could result.

Table XXVI

HEAT RELEASED TO CORE WATER BY RUPTURE OF
50% OF BOILING FUEL RODS

Core	Weight of Oxide in Boiler, lb	Heat Content of 50% of Oxide,* Btu	Approximate Water Within Boiler,** lb	Enthalpy Increase, Btu/lb
Boiling	2360	571,000	414	1380
Central Superheater	1903	460,000	331	1390
Peripheral Superheater	1710	421,400	304	1390

* Δt (assumed) = 4720°F, specific heat of oxide = 0.08 Btu/(lb)(°F), heat of fusion of oxide = 106.2 Btu/lb

**Density of water at 490°F taken as 50 lb/ft³. Only water within core is considered, i.e., no diffusion of heat beyond this region is assumed for the short period of time considered.

2. Dosage Calculations

In the following discussion, the assumption is first made that an incident of explosive proportions has occurred following an operating period of sufficient length to justify consideration of the steady-power fission products only. It is also assumed that 50% of the fission products escape, and it is further assumed that these are concentrated in a cloud that diffuses from a point source over the reactor site.

The treatment given here is largely guided by the extensive analysis carried out for the EBR-II reactor, because the problem is similar insofar as the external dosage calculations are made. The weather conditions assumed are characteristic of those occurring at the NRTS as determined by observations over the two year period, September, 1950 to August, 1952.⁽³²⁾

The γ and β source strength are both approximated in Ref. 33, pp. 107 and 157:

$$Q = 2.3 \times 10^{14} P t^{-0.21} \text{ Mev/sec} \quad .$$

If P is taken as 10^4 kw, representing 50% of fission products, we get

$$Q = 2.3 \times 10^{18} t^{-0.21} \text{ Mev/sec (see Ref. 33, pp. 107 and 157)} \quad .$$

a. Atmospheric Diffusion

If W is the deposition rate in energy/(m²)(sec), TID is the total integrated dose, and Q is the source strength, with corresponding

energy units, then it is convenient to write the standard expressions in the following form:

$$\frac{TID}{Q} = \text{time integral of activity concentration at location of distant observer from gaseous and small particulate airborne activity, for a unit source (sec/meter}^3\text{)}.$$

$$\frac{W}{Q} = \text{concentration of activity on the ground at the observer for a unit source (meter}^{-2}\text{)}.$$

The required formulae may then be given:

$$\frac{TID}{Q} = \frac{2 \exp \left[-h^2 / (C^2 D^{2-n}) \right]}{\pi \bar{u} C^2 D^{2-n}},$$

and for fumigation conditions, i.e., inversion above and lapse below,

$$\frac{TID}{Q} = \frac{1}{\sqrt{\pi} C_y H \bar{u} D^{(2-n)/2}}.$$

For washout at "optimum" scavenging rate (rain falls at a steady rate from time of disaster),

$$\frac{W}{Q} = \frac{1}{e \sqrt{\pi} C_y D^{2-(n/2)}}.$$

For instantaneous washout, or for the case when a rain storm occurs with the radioactive cloud immediately above the observer,

$$\frac{W}{Q} = \frac{1}{\pi C^2 D^{2-n}},$$

where

D = downwind distance to observer, m

h = height to center of cloud, m

u = mean wind speed, m/sec

n = Sutton's stability parameter

H = the height of the lapse layer under fumigation conditions, m

C = the generalized diffusion coefficient for isotropic turbulence, m^{n/2}.

Calculated values of the TID/Q and the W/Q are given in Table XXVII. The meteorological parameters chosen are also listed there.

Table XXVII

TOTAL INTEGRATED DOSE AND RELATIVE GROUND CONCENTRATION DOWNWIND FOR UNIT SOURCE

Distance Downwind		$\frac{TID}{Q} = \frac{\text{Total Integrated Dose, sec}}{\text{Source Strength}} \frac{m^3}{m^3}$							$\frac{W}{Q} = \frac{\text{Relative Ground Concentration}}{\text{Concentration}}, m^{-2}$	
		Daytime Conditions, Strong Lapse				Nocturnal Conditions Large Inversion		Fumigation Conditions	Total Instantaneous Washout	Washout at "Optimum" Scavenging Rate
		Average Wind Speed		Low Wind Speed		Average Wind Speed				
Miles	Meters	n = 0 20 ū = 4 h = 10 C ² = 0 088	n = 0 20 ū = 4 h = 70 C ² = 0 053	n = 0 20 ū = 4 h = 500 C ² = 0 027	n = 0 20 ū = 1 h = 10 C ² = 0 110	n = 0 50 ū = 1 h = 10 C ² = 0 006	n = 0 50 ū = 3 h = 70 C ² = 0 0025	n = 0 50 ū = 2 h = 70 C = 0 050	n = 0 20 ū = 4 h = 500 C ² = 0 027	n = 0 25 ū = 4 h = 500 C = 0 080
0 5	800	1 08(-5)	1 03(-5)	(-29)	3 42(-5)	2 24(-3)	(-40)	5 2(-4)	7 01(-5)	9 35(-6)
3 5	5,630	3 21(-7)	5 24(-7)	2 02(-7)	1 03(-6)	2 41(-4)	1 94(-6)	1 2(-4)	2 09(-6)	2 41(-7)
5 0	8,045	1 69(-7)	2 78(-7)	2 32(-7)	5 40(-7)	1 44(-4)	7 80(-6)	8 3(-5)	1 10(-6)	1 23(-7)
10 0	16,100	4 84(-8)	8 02(-8)	1 23(-7)	1 55(-7)	5 15(-5)	1 59(-5)	5 6(-5)	3 16(-7)	3 36(-8)
27 0	43,500	8 09(-9)	1 34(-8)	2 53(-8)	2 59(-8)	1 17(-5)	7 54(-6)	2 7(-5)	5 27(-8)	5 21(-9)
(TID/Q) _{max}		5 86(-4)	1 20(-5)	2 34(-7)	2 34(-3)	2 34(-3)	1 59(-5)	-	--	--
D _{max} TID/Q, m		49 9	571	7380	43 2	635	15,600	--	--	--

Note: The numbers in parentheses indicate the powers of 10 by which the tabulated values are to be multiplied.

b. External Dosage Calculation

The β dosages given in Table XXVIII are obtained by using the TID/Q values according to the expression

$$\beta \text{ dose (rep)} = \left(\frac{TID}{Q} \right) QKG, \quad ,$$

where

K, a conversion factor = $1 \text{ rep} / (6.8 \times 10^{10} \text{ Mev/m}^3)$

G, a geometry factor = $(0.5)(0.64) = (0.32)$.

Table XXVIII

EXTERNAL β DOSE (REP) AND γ DOSE (ROENTGENS)
FROM AIRBORNE FISSION PRODUCT ACTIVITY

Distance Downwind		Daytime Conditions								Nocturnal Conditions				Fumigation Conditions			
		Average Wind Speed						Low Wind Speed									
		h = 10 ū = 4 n = 0.20 C ² = 0.088		h = 70 ū = 4 n = 0.20 C ² = 0.053		h = 500 ū = 4 n = 0.20 C ² = 0.027		h = 10 ū = 1 n = 0.20 C ² = 0.110		h = 10 ū = 1 n = 0.50 C ² = 0.0060		h = 70 ū = 3 n = 0.50 C ² = 0.0025				h = 70 ū = 2 n = 0.50 C = 0.050	
		β	γ	β	γ	β	γ	β	γ	β	γ	β	γ			β	γ
Miles	Meters																
0.5	800	3.83	28	3.67	27	<<1	18	9.1	260	597	1000	<<1	32	162	2530		
3.5	5 630	076	75	124	12	048	65	181	17	42.6	180	43	29	25	400		
5.0	8 045	037	4	061	70	051	48	088	8.0	23.5	140	1.61	27	18	270		
10.0	16 100	009	01	015	15	023	023	022	2.5	7.3	57	2.84	24	9.3	142		
27.0	43,500	001	-	002	-	004	-	003	-	1.3	14	1.09	17	3.4	50		

Note: β doses have been divided by a factor of ten to allow for shielding effects of clothing.

The factor 0.5 corrects to the fact that the β flux at the surface of the skin is half that in free air. The other factor, 0.64, represents an average reduction factor for a man 1.8 m tall, arising as a consequence of the ground effect on the free-air-radiation flux.

Gamma dosages are obtained by use of the Holland nomograph⁽³³⁾ and are also given in Table XXVIII. Because of the longer range of the γ radiation, parts of the cloud not immediately over the observer also contribute to the dose; thus the Holland nomograph is considered to give a better approximation than that obtained by calculating directly with the TID/Q. The γ doses under fumigation conditions were calculated using the same expression as that for the β dose due to airborne activity, except that G is equal to 0.5.

The results indicate that under nocturnal conditions the control building (as well as EBR-I, ZPR-III, etc.) could be within the 300-r exclusion radius. Fumigation conditions constitute the most severe case, in which even the Central Facilities Area would fall within the 300-r exclusion distance.

It must be emphasized that the assumption of an accident severe enough to discharge 50% of the reactor products into a cloud and with no fallout enroute presents a virtually impossible situation. Fumigation conditions are present for short periods of time only, occurring during the transition from nocturnal to daytime conditions. Fumigation may persist for a maximum period of one hour. With a $\bar{u}$ of 2 m/sec, the propagation time to the Central Facilities Area would be 47 min.

Table XXIX gives the predicted activity on the ground resulting from washout using the following expression for the (γ and β) dose rate: $(W/Q)[E_\gamma + (E_\beta/10)]$, where E_γ and E_β are conversion factors which give the β and γ dosage rates per unit ground concentration of β and γ activity. For a ground concentration of one Mev/(sec)(m²), $E_\gamma = 1.51 \times 10^{-13}$ r/sec and $E_\beta = 1.51 \times 10^{-11}$ r/sec.

Table XXIX

EXTERNAL RADIATION HAZARD DUE TO GROUND DEPOSITION

Distance Downwind		External $\beta + \gamma$ Dose Rate at Time D/ $\bar{u}$ in rep/hour		External $\beta + \gamma$ Dose During First Three Hours After the Disaster in rep				Radius of Contaminated Area, m
		Average Windspeed		Average Windspeed		High Windspeed		
		Total Instantaneous Washout	Washout at 'Optimum' Scavenging Rate	Total Instantaneous Washout	Washout at 'Optimum' Scavenging Rate	Total Instantaneous Washout	Washout at 'Optimum' Scavenging Rate	
Miles	Meters	$n - 0.20$ $\bar{u} - 4$ $h - 500$ $C^2 - 0.027$	$n - 0.25$ $\bar{u} - 4$ $h - 500$ $C - 0.080$	$n - 0.20$ $\bar{u} - 4$ $h - 500$ $C^2 - 0.027$	$n - 0.25$ $\bar{u} - 4$ $h - 500$ $C - 0.080$	$n - 0.20$ $\bar{u} - 12$ $h - 500$ $C^2 - 0.027$	$n - 0.25$ $\bar{u} - 12$ $h - 500$ $C - 0.080$	$n - 0.20$ $\bar{u} - 4$ $h - 500$ $C - 0.16$
0.5	800	3.2(5)	4.2(4)	4.9(5)	6.7(4)	4.9(5)	6.8(4)	66
3.5	5 630	6.3(3)	7.24	1.3(4)	1.4(3)	1.4(4)	1.7(3)	380
5.0	8 045	3.1(3)	343	5.9(3)	680	720	8.4(2)	524
10.0	16 100	760	81	1.2(3)	137	1.8(3)	198	978
27.0	43, 500	103	10	0	0	237	22	2392

Note All β doses have been reduced by a factor of 10 to account for the shielding effect of clothing. The numbers in parentheses indicate the powers of 10 by which the tabulated values are to be multiplied.

Columns 5-8 were obtained by integrating the dose rate from $t = D/\bar{u}$ to $t = 3$ hr. The radius of the contaminated area is essentially the radius of a cloud puff coming from an instantaneous point source, i.e., $\text{radius} = CD^{(2-n)/2}$.

d. Inhalation and Ingestion Dosages

To obtain an estimate of the severity of the internal dosage that might be received at the populated points previously listed, it is necessary to consider the fission product inventory. A convenient tabulation by Burnett⁽³⁴⁾ lists those products having half-lives greater than 10 hr after sustained operation for 40 and 400 days. Those listed account for only about 44% of the total, although they are considered of controlling importance. With a more conservative assumption on the release of fission products, as suggested by Bates and Beek⁽³⁵⁾ (100% of noble gases, 50% of iodines, and 1-5% of particulates), Table XXX is constructed.

Table XXX

FISSION PRODUCTS CONTRIBUTING TO INGESTION HAZARD

Product	Organ Affected	Amount Present 100 sec after Incident $t_0 = 400$ days, curies/kw	Fraction Released, %	Activity Escaping to Atmosphere, curies/20 Mw
Iodines	Thyroid	330	50	3.3×10^6
Particulates	Bone	907	5	9×10^5
	Kidneys	152	5	1.5×10^5
	Muscles	2	5	2×10^3
Total		1394		

In this listing, Burnett's values have been adjusted to conform with the reactor power by the same expression as used before, but where the average γ energy is now taken as 0.7 Mev/disintegration and 100 sec is allowed for escape from the building,

$$A = 8880 [t^{-0.21} - (t_0 + t)^{-0.21}] \text{ curies/kw}$$

and

$$0.44 A = 1350 \text{ curies/kw for } t_0 = 40 \text{ days}$$

$$0.44 A = 1394 \text{ curies/kw for } t_0 = 400 \text{ days.}$$

The amount of activity, A , accumulated in the critical organ due to inhalation of I^{131} and Sr^{90} (isotopes giving greatest dosage) at

significant points and under nocturnal conditions is given in Table XXXI. The values listed were obtained by calculation with the formula

$$A = \left(\frac{TID}{Q} \right) (QBf) ,$$

where the TID/Q are taken from Table XXVII, B is the breathing rate ($5 \times 10^{-4} \text{ m}^3/\text{sec}$ for moderate activity), and f is the fraction of the inhaled material that reaches the critical organ.

Inclusion of a fallout factor would have the effect of increasing near and reducing distant dose values.

Table XXXI

INGESTION DOSAGE, INVERSION CONDITIONS
(Microcuries to Critical Organ)

Isotope		Activity Assumed to Leave Reactor, μc	Site and Distance Downwind in Meters				
^{131}I ^{90}Sr - ^{90}Y		3.5×10^{11} 1.9×10^9	800 EBR-I	5630 CFA	8045 MTR	16100 NRF	43500 ANP
$\frac{TID}{Q}$		$n = 0.5$ $\bar{u} = 3$ $h = 70$ $C^2 = 0.0025$	(-40)	1 94(-6)	7 80(-6)	1 59(-5)	7 54(-6)
Isotope	f, Inhalation Uptake	Maximum Permissible Amount in Critical Organ, μc					
^{131}I ^{90}Sr - ^{90}Y	0.15 0.22	0.12 0.7	- -	5 1(1) 4 1(-1)	2 1(2) 1 6(0)	4 2(2) 3 3(0)	2 0(2) 1 5(-1)

Note: The numbers in parentheses indicate the power of ten by which the tabulated values should be multiplied.

F. Hydrology, Seismology, and Meteorology

1. Hydrology⁽³⁷⁾

a. Soil and Subsurface Characteristics

The National Reactor Testing Station (NRTS) is located on a level plain at an average elevation of 4865 ft ranging from an elevation of 4788 to 4965 ft above sea level.

The surface of much of the plain is covered by waterborne and windborne top soil, under which there is a considerable depth of gravel, ranging in size from fine sand to 3 in. in diameter. At the several locations inspected to date, the gravel lies from approximately 1 to 50 ft under the top soil. Lava rock extends below this gravel layer and downward to a considerable depth, ranging at least to the water table. The lava rock is honeycombed with openings of about $\frac{1}{8}$ in. in diameter. Frequently, large openings occur, and these range upwards to the size of tunnels, tubes, and caves.

What little surface drainage there is, is toward the north-east, opposite to the main body of water flow. Normally, surface drainage is small due to the high porosity of the gravel overburden.

b. Drainage

The National Reactor Testing Station overlays a natural underground reservoir of water having an estimated lateral flow of not less than 500 ft³/sec (323,136,000 gal/day).

The main sources of water for this reservoir are the streams which start in the mountains to the north and disappear into the porous soils of the NRTS area. These streams include Big Lost River, Little Lost River, and Birch Creek.

The path of water flow from the surface to the ground water level is unknown. However, it is expected that the drainage would be rapid. The flow would be very rapid through the gravel overburden, while the drainage pattern through the lava rock would be less rapid, but still very high as compared to flow through sands or clays. It is expected that the flow would be around, rather than through, the clay beds. Therefore, in case of a major accident with loss of a large volume of liquid wastes, the ground water would undoubtedly become contaminated in a very short time.

The estimated rate of flow of the main body of water through the lava is approximately one-half mile per year. Based on this estimate, the contaminated water would reach the Snake River Canyon Springs and enter the Snake River in about 120 to 140 years, depending upon the exact location of the reactor plant within the Testing Station area.

2. Seismology⁽³⁷⁾

The NRTS site is located in a region which The Pacific Coast Uniform Building Code, 1949, designated as a Zone 2 area, as given by the Seismic Probability Map of the United States, published by the United States Coast and Geodetic Survey.

Quoting J. Stewart Williams:⁽³⁶⁾

"Earthquake risk at this site (NRTS) is appreciable, but not great. Since isoseismal maps of principal earthquakes have been drawn, beginning in 1925, the isoseismals of only one earthquake reached Cerro Grande. (Cerro Grande is a stop on the Union Pacific Railroad located near the south boundary of the NRTS.) Prior to this time several earthquakes recorded for surrounding areas may have been felt at Cerro Grande. There is no record of a major earthquake originating close to Cerro Grande.

"However, Cerro Grande is surrounded by areas of comparatively high seismic activity. Furthermore, it lies in a region of geologically young faults, any of which must be considered potentially active. For these reasons earthquake risk at the NRTS site should not be dismissed from consideration in planning any structure to be built at the site.

"Cerro Grande is situated within 150 miles of several areas of pronounced earthquake activity. Any one of these might produce a shock stronger than it has yet produced with a corresponding greater intensity at Cerro Grande. The earthquake history of 100 years for this area is very short, from the geological point of view. An earthquake might occur any day that would alter substantially our ideas of the distribution of seismic activity in the area about the Snake River Plains.

"Earthquake risk in any area is relative to the type of structure to be built. Reinforced concrete buildings, well constructed in every way, with high factors of safety and incorporating features recommended by engineers acquainted with earthquake-proof design, stand less risk of being damaged. Such buildings, set on the lava bedrock at Cerro Grande, certainly would be reasonably safe from earthquake damage.

"No traces of recent faults are known by the writer to cross the Snake River Plains. The chances, then, of displacement in the ground that would cut water supplies are small enough to be eliminated from consideration.

"In spite of the fact that a Zone 3 area exists both north and south of the Arco area, the distances are so great that Zone 2 has been considered completely safe."

3. Meteorology⁽³⁸⁾

The National Reactor Testing Station (NRTS) covers about 680 square miles of the upper Snake River Plain. The most important topographic features are the northeast-southwest orientation of the Plain, and the Continental Divide to the north and east. About 90 per cent of the time, surface winds are either southwesterly or northeasterly. The Divide prevents the invasion of many of the cold continental air masses in winter. All air masses that enter the Plain must first cross a mountain barrier, regardless of the direction from which they enter. Because they are subject to lifting, these masses usually precipitate moisture over the mountains, and enter the Plain sufficiently dry to give the region its desert-like characteristics.

The dryness and the infrequency of low cloudiness permit intense solar heating of the ground during the day, and rapid radiational cooling at night, which results in a very large daily range of temperature. The average range is greatest (about 37°F) in summer and least (about 23°F) in winter.

Temperature inversions based at the surface are present for at least a few hours almost every day of the year and with snow cover may persist for several consecutive days. Temperature inversions through the lower 250 feet of the atmosphere are present for about half of the total time for the year as a whole. While temperature measurements have not been made to the top of the inversions at the times of maximum depths, the average maximum depth is estimated to be approximately 2,000 feet.

There is a strong tendency for up-valley flow (southwest wind) during the day and down-valley flow (northeast wind) at night. The local effect is sometimes countered and sometimes augmented by the pressure gradient, or the larger scale features of the wind flow pattern. Frequently the local effects dominate the wind flow near the ground in the face of strongly opposing pressure forces. Part of the station will experience northerly winds while another part has southerly winds. This situation arises at times from drainage effects and at other times from a topographically induced eddy. A wind rose at the Central Facilities area is shown in Fig. 60.

The heaviest precipitation falls as scattered rain showers during the warmer months, and the rain from a single passing shower may in a matter of minutes exceed considerably the normal total for the month in which it occurs. Yet two or three months may pass in summer without a single shower passing the station. Despite this extreme variability, annual totals over the NRTS have not varied greatly from year to year or from place to place. One in. of rainfall in an hour is about the maximum intensity to be expected, and two in. in 24 hr are not likely to be exceeded. Average annual precipitation at the site is 7.7 in. Maximum winter snow depth has been extremely variable during the period of record.

Destructive winds have not been a problem. Tornadoes have never been reported in the region, although funnel-shaped clouds have been sighted over the NRTS on several occasions. Structures designed to withstand average winds of 60 mph and peak gusts to 80-85 mph are not likely to be damaged.

Four to six thunderstorms can be expected during each summer month and some of these produce hail that can be damaging to crops. Lightning protection is a must for all man-made structures, since the lack of comparable natural targets makes them particularly vulnerable.

Fig. 60

Central Facilities 20 Foot Level Wind Rose
1950 Through 1956

Supercooled fog or low stratus clouds occur occasionally in winter with a snow cover and a persistent high pressure system, and may last for several consecutive days. These conditions permit the accumulation of considerable rime icing on power and telephone lines as well as on trees. Destructive glaze icing has not occurred in this area during the period of record.

The dust content of the air is very small while the ground is snow covered. Dust-favoring winds (gusts of 20 mph or more) occur on the average for 6 hours during 13 windy days per month. Dust concentrations have varied from a low of 0.014 mg/m^3 over snow cover to a high of 0.77 mg/m^3 on a day when dust devils were present near the sampling station. Concentrations vary widely from place to place, depending on human activity. Average daytime concentrations during the warmer months run about 0.07 mg/m^3 . Visibility is not significantly reduced by dust except briefly and infrequently during dry thunderstorms.

ACKNOWLEDGEMENTS

Any document which covers the scope of this Design and Hazards Summary Report, BORAX V, must necessarily owe a great deal to a great many persons. The editors and authors of this report are indebted to the following persons: R. P. Anderson, J. B. Heineman and E. A. Spleha for their work and contributions on experimental heat engineering; R. Liimatainen for contributions on metal-water reactions; J. A. Thie* for his advice and assistance on the experimental program and reactor physics calculations; R. P. Hearn for his work on the reactor vessel and process piping; C. C. Miles for his work on the water-chemistry program; M. K. Butler, and members of her staff of the Applied Mathematics Division, S. L. Slate and N. J. Price, for their assistance and computations on reactor physics and heat engineering problems; W. J. Kann for the modification and testing of the control rod drives; R. W. Adamson** and D. H. Drummond*** for their work on the stress analysis of core components; D. R. Greenwood, M. E. Brown, J. K. Christensen, R. R. Delfs, C. M. Ekelund, K. K. Mansfield, L. R. Monson, and K. E. Tibbetts for their efforts in producing the illustrations herein; members of the staff of the Metallurgy Division and Central Shops Department for the development and specification of fuel elements; Norman Engineering Company, Los Angeles, California, the architect-engineer on building and plant; W. P. Rosenthal for his assistance in editing this report; and B. N. Lloyd, D. R. Muir, and B. P. Sorensen for their stenographic services.

---

* Formerly with ANL, now with McLain-Rodger Associates.

** On loan from California Polytechnic Institute.

*** On loan from Montana State College.

BIBLIOGRAPHY

1. J. B. Heineman, An Experimental Investigation of Heat Transfer to Superheated Steam in Round and Rectangular Channels, ANL-6213 (June 1960).
2. W. H. McAdams, W. E. Kennel, and J. N. Addoms, Heat Transfer to Superheated Steam at High Pressures, Trans. ASME 72, 421-428 (1950).
3. S. Levy, R. A. Fuller, and R. D. Niemi, Heat Transfer to Water in Thin Rectangular Channels, Trans. ASME, Series C, Journal of Heat Transfer, 81, No. 2, 129, (1959).
4. M. K. Butler and J. M. Cook, UNIVAC Programs for the Solution of One-dimensional Multigroup Reactor Equations, ANL-5437.
5. H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan, MUFT-4-Fast Neutron Spectrum Code for the IBM-704, WAPD-TM-72 (July 1957).
6. H. Amster and R. Suarez, The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code, WAPD-TM-39 (Jan 1957).
7. L. Dresner, The Effective Resonance Integral of U^{238} and Th^{232} , Nuc. Sci. Eng. 1, 68 (1956).
8. E. Hellstrand, Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries, J. Appl. Phys. 28 1493 (1957).
9. G. G. Bilodeau, W. R. Cadwell, J. P. Dorsey, J. G. Fairey, and R. S. Varga, PDQ, An IBM 704 Code to Solve the Two-dimensional Few-group Neutron Diffusion Equations, WAPD-TM-70 (Aug 1957)
10. A. M. Weinberg and E. P. Wigner, The Physical Theory of Neutron Chain Reactors, University of Chicago Press, Chicago (1958).
11. B. G. Carlson, The S_n Method and the SNG Code, LAMS-2201.
12. General Nuclear Engineering Corp., Nuclear Superheat Development Program, Fourth Quarterly Progress Report, GNEC 138 (April-June 1960).
13. R. O. Brittan, Reactor Containment, Including a Technical Progress Review, ANL-5948 (1959)
14. L. Epstein, Metal-Water Reactions: VII. Reactor Safety Aspects of Metal-Water Reactions, GEAP-3335 (1960)
15. Chemical Engineering Division Summary Report, April, May, June, 1960, ANL-6183

16. Chemical Engineering Division Summary Report, July, August, September, 1960, ANL-6231
17. R. C. Liimatainen, R. O. Ivins, Experimental Study of Metal-Water Reactions Initiated by Nuclear Reactor Transients, Transactions of the American Nuclear Society, Vol. 3, No. 2, p. 430-31 (1960).
18. J. R. Dietrich, Experimental Investigation of the Self-limitation of Power during Reactivity Transients in a Subcooled Water-moderated Reactor, BORAX I Experiments, AECD-3668, (1954).
19. J. R. Dietrich and D. C. Layman, Transient and Steady-state Characteristics of a Boiling Reactor, The BORAX Experiment, ANL-5211 (1953).
20. Chemical Engineering Division Summary Report, October, November, December, 1960, ANL-6287
21. Preliminary Hazards Report for the ML-1 Nuclear Power Plant, Aerojet-General Nucleonics, p. B-3, IDO-28537.
22. S. G. Forbes and P. French, The Mechanism of Moderator Expulsion in SPERT I Shutdown, Trans. Am. Nuc. Soc. 2 (No. 1) 214 (1959).
23. S. G. Forbes et al., Analysis of Self-shutdown Behavior in the SPERT I Reactor, IDO-16528 (July 1959).
24. W. H. Arnold, Jr. and R. A. Dannels, A Monte Carlo Study of the Doppler Effect in UO_2 , WCAP-1572 (May 1960).
25. E. Hellstrand, P. Blomberg, and S. Hörner, The Temperature Coefficient of the Resonance Integral for Uranium Metal and Oxide, Nuc. Sci. and Eng. 8, 497 (1960)
26. J. I. Owens, Metal-Water Reactions: II. An Evaluation of Severe Nuclear Excursions in Light Water Reactors, GEAP-3178 (June 1959).
27. R. Avery (Editor), Reactor Physics Constants, ANL-5800.
28. R. O. Brittan, Some Problems in the Safety of Fast Reactors, ANL-5577. (RE-129 is an IBM-704 Version of RE-29.)
29. B. S. Maxon, O. A. Schulze, and J. A. Thie, Reactivity Transients and Steady-state Operation of a Thoria-Urania-Fueled Direct-cycle Light Water Boiling Reactor (BORAX IV), ANL-5733 (Feb 1959).
30. K. Hikido, Metal-Water Reactions: IV. Heat Transfer Conditions during Severe Nuclear Excursions in Water-cooled Reactors, GEAP-3204 (Sept 1959)
31. J. R. Dietrich, Experimental Determinations of the Self-regulation and Safety of Operating Water-Moderated Reactors, Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Volume 13, p. 88, United Nations, New York (1956).

32. L. J. Koch, et al., Hazard Summary Report EBR-II, ANL-5719 (1957).
33. Meteorology and Atomic Energy, AECU-3066, Supt. of Documents, Washington, D. C. (1955).
34. T. J. Burnett, Reactors, Hazard vs. Power Level, Nuclear Science and Engineering, 2, 382-393 (1957)
35. M. B. Biles and C. K. Beck, Some Safety Matters Relating to Small Nuclear Power Plants, IAEA, Conference on Small and Medium Power Reactors, Vienna, Austria, September 5-9, 1960.
36. S. McLain and R. K. Winkleblack, Hazards of the Materials Testing Reactor, ANL-SM-236 (June 15, 1950).
37. L. J. Koch et al., Hazard Summary Report, Experimental Breeder Reactor II (EBR-II), ANL-5719 (May 1957).
38. U. S. Dept. of Commerce, Weather Bureau, Local Climatological Data, 1959, Idaho Falls 46 W. Idaho.
39. A. Amouyal, P. Benoist, and J. Horowitz, Nouvelle Methode de Determination des Facteurs D'Utilisation Thermique D'Une Cellule, J. Nuclear Energy 6, 79 (1957).