ANL-6078
Reactor Technology
(TID-4500, 16th Ed.)
AEC Research and
Development Report

ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois

ZERO-POWER EXPERIMENTS ON THE ARGONNE
LOW POWER REACTOR (ALPR)

by

D. H. Shaftman

Reactor Engineering Division

May 1961

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

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.

TABLE OF CONTENTS

	<u>Page</u>
I. INTRODUCTION	9
II. DESCRIPTIONS OF REACTOR COMPONENTS	12
A. Fuel Assemblies	12
B. Control Rods; Rod Channels; Rod Drives	17
C. Source Assembly	19
D. Burnable-poison Strips	20
III. NUCLEAR INSTRUMENTATION FOR MEASUREMENT OF "NEUTRON FLUX"	22
IV. REMARKS ON THE ZERO-POWER REACTOR PHYSICS PROGRAM	26
A. Introduction.	26
B. Purposes of the Program.	26
C. Operating Instructions for the Nuclear Test Programs . . .	27
D. Outline of Zero-power Reactor Physics Measurements . . .	27
E. Temporary Devices and Procedures Used in the Zero- power Experiments.	28
1. Support Grid for Rod Drives.	28
2. Measurement of "Neutron Flux" Level	28
3. Source Assembly; Source Manipulation.	28
4. Procedures for Poisoning the Reactor Water with Boric Acid	29
5. Auxiliary System for Heating the Reactor Water	30
F. Fuel Loading Charts	30
V. APPROACH TO MINIMUM CRITICAL LOADING: TEN FUEL ASSEMBLIES; NO BURNABLE-POISON STRIPS	33
VI. HOLD-DOWN CAPACITY OF CONTROL RODS IN LOADINGS SMALLER THAN FORTY FUEL ASSEMBLIES	35
VII. PRELIMINARY EXPERIMENTS WITH THE INITIAL REFER- ENCE 3-Mwt CORE OF FORTY FUEL ASSEMBLIES (LOADING NO. 19) AT ROOM TEMPERATURE.	39
A. Introduction.	39
B. Control System Calibrations.	40
C. Reactivity Effect of Fuel Assembly Omission	44
D. Preliminary Calibrations of Burnable-poison Strips	44

	<u>Page</u>
VIII. EXPERIMENTS ON THE INITIAL CORE OF FIFTY-NINE FUEL ASSEMBLIES (LOADING NO. 38)	47
A. Introduction	47
B. Critical Control Rod Configurations	48
C. Reactivity Effect of Improper Orientation of a Fuel Assembly.	49
D. Heating Experiments	49
E. Reactivity Effect of Moving Control Rods in the Cold Reactor	51
F. Reactivity Effect of Boric Acid in the Reactor Water	52
IX. INTERMEDIATE CORE LOADINGS (48 or 51 Fuel Assemblies)	53
X. ADDITIONAL EXPERIMENTS WITH THE INITIAL REFERENCE FORTY-ASSEMBLY CORE	54
A. Reactivity Effects of Moving Control Rods	54
B. Reactivity Effects of Boric Acid in the Reactor Water . . .	56
C. Reactivity Controlled by Burnable-poison Strips with Control Rods Banked at ~ 20 In.	57
D. Experimental "Neutron Flux" Distributions in a Central Control Cell (Loading No. 46)	62
XI. EXPERIMENTS WITH THE FINAL REFERENCE 3-Mwt CORE OF FORTY FUEL ASSEMBLIES (LOADING NO. 57). . .	63
A. Introduction and Summary.	63
B. Reactivity Effect of Moving Control Rods in the Cold Reactor	64
C. Reactivity Effect of Boric Acid in the Reactor Water	66
D. Heating Experiments	67
E. Neutron Flux Mapping of the Core at ~ 420°F	68
XII. FINAL CORE LOADING OF FIFTY-NINE FUEL ASSEMBLIES.	70

	<u>Page</u>
APPENDICES	75
A. Remarks on the Evolution of the Reactor Core Design: Design Motivations and Influence of Zero-power Experiments.	75
1. The Fuel Assembly	75
2. Size of the Core	76
3. Control Rods	76
4. Burnable Poison	77
5. Interplay of Core Design and the Zero-power Reactor Physics Program	77
B. Composition of the Fuel "Meat" in the ALPR Fuel Plates . .	79
C. Burnable-poison Strips: B ¹⁰ Contents and Locations in the Final Core Loadings	87
D. Organization Chart and Operating Instructions for Zero- power Nuclear Test Programs.	91
1. Organization and Responsibilities	91
2. General	92
3. Routine Checks.	93
4. Control Tests.	94
5. Multiplication or Critical Experiments	95
E. Significance of Integration of Curves of Differential Reactivity Effects	97
REFERENCES	99
ACKNOWLEDGMENTS	100

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Locations of Neutron Detectors in Temporary Instrument Tubes.	25
2.	Functions of the Instrumentation Channels for Indication of Power Level and of "Neutron Flux" Level.	25
3.	Control Rod Configurations for Extrapolation to Minimum Critical Core Loading.	33
4.	Approach to Minimum Critical Core Loading without Burnable-poison Strips	34
5.	Hold-down Capacity of Cross Control Rods in Core Loadings without Burnable-poison Strips.	36
6.	Hold-down Capacity of Cross Control Rods in Core Loadings with One 0.5-g-B ¹⁰ Strip per Fuel Assembly	37
7.	Indicated Rod Positions for Criticality of Loading No. 19 . . .	41
8.	Calibration of Rods No. 1 and 9 by Partial Withdrawal of Rod No. 5 (Loading No. 19 at 93°F)	41
9.	Reactivity Loss upon Removal of a Fuel Assembly from the Otherwise Full Initial Reference Forty-assembly Core (Loading No. 19).	44
10.	Preliminary Reactivity Calibration of 0.5-g-B ¹⁰ Strips in Loading No. 19.	46
11.	Critical Control Rod Configurations for Loading No. 38.	48
12.	Indicated Critical Position of Rod No. 9 vs Orientation of Assembly in Location No. 45 (Loading No. 38).	49
13.	Reactivity Data from Heating Experiments with Loading No. 38.	50
14.	Reactivity Calibration of Burnable-poison Strips "in" Loading No. 45.	59
15.	One-group, One-dimensional Calculations in Slab Geometry of Relative Absorption of Thermal Neutrons by Burnable-poison Strips and by Effective Fuel Zones in a Cold Control Cell (Representative of Central Control Cell in Loading No. 57)	61
B-1.	Uranium Content of Fuel Plates in ALPR Fuel Assemblies . .	80-85
B-2.	Isotopic Composition of Rolled Plates for ALPR Fuel-plate Cores.	86

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
C-1.	Data for the Forty 0.5-g-B ¹⁰ Strips	87
C-2.	Data for the Forty 0.4-g-B ¹⁰ Strips	88
C-3.	Data for the Thirty-five Temporary 0.8-g-B ¹⁰ Strips	88
C-4.	Assignments of Burnable-poison Strips in Loading No. 57 . . .	89
C-5.	Assignments of Temporary 0.8-g-B ¹⁰ Strips in Loading No. 60.	90

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	Schematic of Argonne Low Power Reactor (ALPR) Plant	10
2.	Core Loading Configurations	11
3.	Fuel Assembly.	13
4.	Dummy Fuel Assembly	14
5.	Core Plan, Section	15
6.	Core, Vertical Section.	16
7.	Cross Control Rod	18
8.	Source Assembly	20
9.	Central Control Cell.	21
10.	Permanent Instrumentation for Measuring "Neutron Flux". . . .	23
11.	Power Levels Corresponding to Operating Ranges of Instrumentation Channels I to VI.	24
12.	Temporary Instrumentation for Measuring "Neutron Flux" . . .	24
13.	ALPR Fuel Loading Chart	31
14.	Minimum Critical (Unpoisoned) Loading.	32
15.	Approach to Minimum Critical Loading (All Control Rods at 30 in.).	34
16.	Initial Reference 40-assembly Core (Loading No. 19).	38
17.	Relative Critical Positions of Control Rods No. 1 and 9 (Loading No. 19 at 94°F; Rods No. 3, 5, and 7 at Zero)	40
18.	Reactivity Effect of Moving Control Rod No. 5 from 0 in. to 6 in. vs Indicated Critical Position of Rod No. 9 (Loading No. 19 at 93°F; Rods No. 3 and 7 at Zero; Rod No. 1 at Position Shown in Fig. 17)	42
19.	Reactivity vs Asymptotic Reactor Period ($\beta = 0.006398$; $\ell^* = 5 \times 10^{-5}$ sec).	43
20.	Estimated Reactivity Effect of Moving Control Rod No. 9 (Loading No. 19 at 94°F; Rods No. 3 and 7 at Zero; Rod No. 1 at Position Shown in Fig. 17)	43
21.	Initial 59-assembly Core (Loading No. 38)	47
22.	Differential Reactivity Effect of Moving Control Rods from the 9-rod Bank (Loading No. 38 at $\sim 94^\circ\text{F}$).	52

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
23.	Reactivity Effect of Boric Acid in the Reactor Water (Loading No. 38 at $\sim 94^{\circ}\text{F}$)	53
24.	Differential Reactivity Effect of (A) Moving Control Rods No. 1, 3, 5, and 7 as a Bank, and (B) Moving Control Rod No. 9 (Loading No. 45 at Room Temperature)	54
25.	Differential Reactivity Effect of Moving Control Rod No. 9 from 5-rod Bank (Loading No. 45 at Room Temperature) . . .	55
26.	Reactivity Effects of Moving Control Rods as a 5-rod Bank (Loading No. 45 at Room Temperature)	56
27.	Effect of Boric Acid Concentration on Reactivity and on Critical Position of 5-rod Bank (Loading No. 45 at Room Temperature)	57
28.	One-group, One-dimensional (DSN) S_{16} Computations of Thermal Neutron Flux in Slab Geometry: Central Control Cell of Loading No. 57 at Room Temperature	60
29.	Horizontal Flux Plots (Loading No. 46 at Room Temperature).	62
30.	Final Reference 40-assembly Core (Loading No. 57).	64
31.	Differential Reactivity Effect of Moving Control Rods from 5-rod Bank (Loading No. 57 at Room Temperature)	65
32.	Integral of Differential Reactivity Effect of Moving Rods from 5-rod Bank in Cold Final Reference 40-assembly Core (Loading No. 57).	66
33A.	Positions of Gold Wires for "Flux Plot" in Loading No. 57 ($\sim 420^{\circ}\text{F}$)	69
33B.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 57 ($\sim 420^{\circ}\text{F}$)	69
33C.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 57 ($\sim 420^{\circ}\text{F}$)	69
33D.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 57 ($\sim 420^{\circ}\text{F}$)	69
34.	Final 59-assembly Core (Loading No. 60).	71
35A.	Positions of Copper Wires for "Flux Plot" in the Hot 59-assembly Core (Loading No. 60 at $\sim 365^{\circ}\text{F}$)	72
35B.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 ($\sim 365^{\circ}\text{F}$)	72

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
35C.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	72
35D.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	72
35E.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	72
35F.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35G.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35H.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35I.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35J.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35K.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	73
35L.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	74
35M.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	74
35N.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	74
35P.	Axial Distributions of "Thermal Neutron Flux" in Loading No. 60 (~365°F)	74
36.	ALPR Operations Organization Chart.	91

ZERO-POWER EXPERIMENTS ON THE ARGONNE LOW POWER REACTOR (ALPR)

by

D. H. Shaftman

I. INTRODUCTION

The Argonne Low Power Reactor (ALPR)^(1,2,3) is a prototype of a natural-circulation, boiling-water-reactor nuclear power plant designed to supply the electrical power and space heat requirements of a proposed Auxiliary DEW (Distant Early Warning) Line radar station. The ALPR was designed and constructed at the behest of the Army Reactors Branch, Division of Reactor Development, United States Atomic Energy Commission, and it is located in the Army Reactor Experiment Area at the National Reactor Testing Station, Idaho. A schematic of the ALPR is shown as Fig. 1.

The nominal rate of heat generation of the reactor is 3 Mwt. From this are produced 200 kwe for the net maximum normal electrical load of the station; 60 kwe when needed for a temporary station overload; electrical power for the plant auxiliary equipment; and 400 kw (net) for space heating. The heat is generated largely by the fissioning of atoms of uranium enriched to approximately 93 w/o U^{235} , predominantly as a result of the absorption of thermalized neutrons by the U^{235} .

The reactor core contains a pseudo-cylindrical array of plate-type fuel assemblies, reflected, moderated, and cooled by high-purity water (H_2O) circulating by natural convection within the pressure vessel containing the core. The structural supports, the fuel-plate cladding, etc., in the core are fabricated of an aluminum-nickel alloy (Alcoa X 8001). The operating conditions in the vessel are 300 psig and 421°F. The reactivity control of the reactor is accomplished with the aid of fixed burnable-poison strips, welded to one or both side plates of selected fuel assemblies, and by the movement of aluminum-nickel-canned cadmium control rods from above with rack-and-pinion drive mechanisms.

The entire program of pre-power reactor physics experiments was accomplished at the ALPR site, in situ, over the period from Aug 11, 1958 to Oct 10, 1958. The primary purpose of this report is to summarize the results of that program. However, some discussion of the motivations of certain aspects of the design of the core and of the control system is included, both to establish a historical record and to provide a natural setting for the chronological sequence of experiments. Two "reference" cores were selected: a final reference forty-fuel-assembly core for 3 Mwt operation,

and a "reference" fifty-nine-fuel-assembly loading for possible use at higher power levels in another plant (cf. Appendix A). The loading configurations for the cores are shown in Fig. 2. The forty-assembly core fills out the spaces marked "R," plus a source assembly in location No. 22. In the case of the fifty-nine-assembly core, the source assembly was positioned in location No. 31

FIG. 1
SCHEMATIC OF ARGONNE LOW POWER REACTOR (ALPR) PLANT

FIG. 2
CORE LOADING CONFIGURATIONS

II. DESCRIPTIONS OF REACTOR COMPONENTS

Appendix A summarizes the motivations for and the evolution of the design of components of the core. In the present section, the final design characteristics of these components are discussed in some detail, primarily to facilitate an understanding of the reactor physics of the system. Also, certain features of the rod drives are described.

A. Fuel Assemblies

Each fuel assembly contains nine clad fuel plates, flanged and spot-welded to two side plates. The fuel plate, 27.8 in. long, has an active length of 25.8 in.; the active width is 3.5 in., and the "meat" thickness is 0.050 in. The "meat" is an alloy of uranium, enriched to approximately 93 w/o U^{235} , and aluminum to which has been added ~ 2 w/o nickel. The 0.035-in.-thick fuel-plate cladding and the side plates are made of Alcoa X 8001, an alloy of aluminum with ~ 1 w/o nickel. On the average, a fuel assembly contains 350 g U^{235} . Other dimensions, of cladding, coolant channel, etc., are shown in Fig. 3. A listing of the U^{235} content and the weight percentage of uranium in the "meat" of the individual fuel plates in each fuel assembly is available in Appendix B.

The fuel assemblies fit into an aluminum-nickel core structure of sixteen boxes, including twelve designed to accommodate square arrays of four assemblies and four contoured to contain three assemblies each. Assemblies fit into a grid structure located below the active core. Near the top of each assembly are four stainless steel spacer springs that serve to center the fuel assembly laterally in its location. Consequently, contiguous assemblies are separated by $\sim \frac{1}{8}$ in. of water, and $\sim \frac{1}{16}$ in. of water separates the assembly from the control rod guide shroud; these separation distances are nominal, and no corrections have been made for the presence of burnable-poison strips.

"Dummy" fuel assemblies (Fig. 4) were added to the core, where needed to fill out the boxes containing fuel assemblies, to provide lateral support for the fuel assemblies. In the final reference forty-assembly core, two dummy assemblies were inserted in locations No. 12 and 21 to support the source assembly, when it was positioned in location No. 22 at the completion of the zero-power program (see Fig. 2).

A section of the core plan and a vertical section view of the core are shown in Figs. 5 and 6.

FIG. 4
DUMMY FUEL ASSEMBLY

FIG. 5
 CORE PLAN, SECTION

0 2 4 6
SCALE-INCHES

FIG. 6
CORE, VERTICAL SECTION

B. Control Rods; Rod Channels; Rod Drives

The reactor is controlled by a system of fixed burnable-poison strips and an array of movable cadmium control rods. A total of five cross-shaped control rods are used in the 3-Mwt reference core of forty fuel assemblies. The design provides for a supplementary set of four T-shaped rods at the periphery of the full fifty-nine-assembly core (Fig. 2). The rod drive circuitry is designed so that only one control rod may be withdrawn at a time.

The central cross rod is designated No. 9; the four off-center cross control rods are numbered 1, 3, 5, and 7. Each cross rod has a total span of $14\frac{1}{4}$ in., of which ~ 14 in. is cadmium, 0.060 in. thick and 34 in. long. The cadmium is contained within two sheets of aluminum-nickel alloy (X 8001) and is perforated at regular intervals, vertically and laterally, by 0.5-in.-diameter holes into which the X 8001 alloy has been dimpled and spot-welded to confine the cadmium locally and to keep it in closer contact with the canning. The sheets are edge-welded for greater rigidity and to seal off the rod from water circulation. The center rod has a solid follower of X 8001, 17 in. long; each off-center cross rod has a 5-in. follower. The rod follower protects the absorbing section of the rod from abrasion by the guide shrouds, prevents the flow of coolant water into the rod from below, and tends to reduce the peaking of the thermal neutron flux in the control channel. When the control rod is positioned at "indicated zero," the cadmium overlaps the bottom of the active core by $\sim 3\frac{1}{8}$ in. and the top of the active core by $\sim 5\frac{1}{16}$ in. When the rod is disengaged from the rack-and-pinion control rod drive mechanism, it is inserted an additional $\sim 3\frac{5}{8}$ in. The rod is prevented from dropping through toward the bottom of the pressure vessel by a positive mechanical stop that comes to rest against the top of the control rod shroud. The metered rod travel is 30 in. Additional details of the composition and structure of a cross-type control rod are offered in Fig. 7.

The T-shaped rods are numbered 2, 4, 6, and 8. The fabrication of a tee rod is similar to that of an off-center cross rod, with a 5-in. follower; the long span has an absorbing length of $11\frac{1}{2}$ in., and the active portion of the tee pointing toward the core center is 7 in. long.

The reactor core is subdivided by control rod channels to provide for the guided vertical motion of the control rods. Each channel is 0.5 in. thick, leaving a clearance of ~ 0.14 in. on each side of the 0.22-in.-thick control rod. The control rod guide shrouds are made of X 8001 and are $\sim \frac{5}{32}$ in. thick. In the region of the core, the inner shrouds are perforated at regular intervals, both laterally and vertically, by 2-in.-diameter holes so as to permit a relatively unobstructed flow of coolant fluid in the vicinity of the control rods. Above the core, the slots left by the incompletely bounding

SECTION A-A

SECTION B-B

ALL DIMENSIONS IN INCHES

FIG. 7
CROSS CONTROL ROD

inner shroud structure provide space for lateral fluid transport. The shrouds at the periphery of the core are not perforated, so as to prevent the free flow of steam bubbles to the downcomer volume at the level of the active core.

Normally, the control rod drives are installed on nozzles in the head of the pressure vessel. However, to facilitate core loadings in the initial sequences of experiments, a special, temporary grid structure was installed to support the rod drives. Prior to the performance of "hot critical" experiments with the final reference core loading, this framework was removed.

The external drive is positively engaged with the pinion shaft by a magnetic clutch. If power to the clutch is cut off, the rod falls into the reactor under the acceleration of gravity, accomplishing its full travel within 2 sec. However, the mechanism is so designed that the drive will continue to attempt to move the rod down in the event of a scram signal, by positive action through a "free-wheeling" clutch. Thus, if the rod should stick at any point during its fall the external drive will engage to drive it down.

The speed of the controlled rod motion is restricted by mechanical gearing. At the beginning of the zero-power program the speed of each cross rod was set temporarily at approximately 3 in./min, and the speed of each tee rod at 6 in./min. Subsequently, the gears of rod No. 9 were replaced to reduce the rod speed to 1.8 in./min and to limit the maximum rate of reactivity addition in the cold, final reference 3-Mwt reactor to ~ 0.02 dollars/sec. The gearing of the tee rods was not changed to reduce their speed since the tee rods were not so effective as the cross rods. In the final reference 40-assembly core, the speed of each off-center cross rod was determined to be ~ 2.85 in./min. The nature of the gearing system of the rack-and-pinion mechanism and the inaccuracies in reading the meter indicating rod position introduce an uncertainty of approximately ± 0.05 in. in the indicated rod position.

C. Source Assembly

A source assembly containing a fixed beryllium block and a movable irradiated antimony rod was used for startup in the zero-power program (see Fig. 8). The antimony rod had been activated in the Materials Testing Reactor to an estimated level of approximately 1000 curies at the time of withdrawal. Source neutrons with a kinetic energy of ~ 30 kev are produced when the beryllium captures gammas emitted by the Sb^{124} in the antimony rod. The half-life of Sb^{124} is 61 days.

Prior to the first loading of fuel assemblies, the source assembly was placed in location No. 42 (see Fig. 2). Later, when that location was

required for a fuel assembly, the source was shifted to location No. 31, where it remained until the completion of the zero-power experimentation and the preparation for power operation. At that time it was moved to its permanent position in location No. 22; there the source assembly is held in position laterally by dummy assemblies in locations No. 12 and 21.

FIG. 8
SOURCE ASSEMBLY

D. Burnable-poison Strips

Two lots of burnable-poison strips were prepared for use in the reactor physics experiments: forty 0.026-in.-thick strips, each containing (nominally) 0.5 g B^{10} , and forty 0.021-in.-thick strips, each with 0.4 g B^{10} . The poison strips were fabricated by hot extrusion and rolling of a mixture of powders of aluminum-nickel alloy (X 8001) and a material consisting primarily of boron, and containing approximately 87 w/o B^{10} , into flat plates, 3.875 in. wide and 25.8 in. long.

Temporary attachments of burnable-poison strips to fuel assemblies were made by taping the strips to side plates with polyethylene. Subsequently, the polyethylene tape was removed, the assemblies and the strips were cleaned, and selected strips were tack-welded to one or both side plates of designated fuel assemblies. In the initial reference 3-Mwt core, one full-length, 0.5-g-B¹⁰ strip was fastened to a side plate of each fuel assembly, and the assemblies were oriented so that the strips formed a cross at the center of a typical control cell unit of four fuel assemblies (see Fig. 9).

FIG. 9
CENTRAL CONTROL CELL

Eight 0.4-g- B^{10} strips were cut in two to yield sixteen 12.9-in.-long strips. One half-length strip was attached to the other side plate of each of the central sixteen fuel assemblies, at the bottom half of the active core region, to generate the final reference forty-assembly core from the initial reference loading.

Additional strips were fabricated, each 0.020 in. thick and containing ~ 0.8 g B^{10} , solely for zero-power tests on a fifty-nine-assembly core loading. These strips were not suitable for use at power, and all were returned to Argonne National Laboratory upon completion of the zero-power program.

The measured B^{10} content of each burnable-poison strip is given in Appendix C.

III. NUCLEAR INSTRUMENTATION FOR MEASUREMENT OF "NEUTRON FLUX"

Six channels of instrumentation comprise the permanent devices providing indications from which the neutron flux level in the reactor is inferred (see Figs. 10 and 11). In the operating plant, the neutron detectors are positioned essentially diametrically opposite the source, in four permanent instrument wells arranged in an arc beyond the shell supporting the pressure vessel (cf. Fig. 1). In these locations, the permanent neutron monitors, channels I to VI, are intended to detect neutron multiplication of the strong fixed source by a forty-assembly core. This instrumentation would have been unsuitable for measurements of neutron level in situations in which much smaller cores were sought and in which the antimony rod was not in close proximity to the beryllium in the source assembly. Alternatively, if the antimony rod had been left in its normal position, in order to overcome the perturbing influence of the source during measurements of small changes in reactivity and during rod calibration experiments, it would have been necessary to raise the neutron-flux level to a point where the fuel assemblies would have been too radioactive for convenient handling during the zero-power program. Therefore, four sealed aluminum tubes which could fit into the core structure, in vacant fuel assembly locations, were used to position channels II to VI closer to the periphery of the smaller cores of the early loadings. Three additional channels were set up as temporary instrumentation, to detect very low levels of neutron flux (see Fig. 12).

During the room-temperature zero-power experiments, the recorder for Channel III T was kept in the operating room to provide a direct indication of neutron-flux level to the physicist observing the changes in core loading.

FIG. 10
PERMANENT INSTRUMENTATION FOR
MEASURING "NEUTRON FLUX"

POWER LEVEL IN WATTS
BY DECADES

FIG. 11
POWER LEVELS CORRESPONDING TO OPERATING RANGES
OF INSTRUMENTATION CHANNELS I TO VI

CHANNEL

FIG. 12
TEMPORARY INSTRUMENTATION FOR
MEASURING "NEUTRON FLUX"

Table 1 details the positions of the various neutron detectors in the temporary instrument tubes. Table 2 summarizes the functions of the permanent and temporary channels, supplementing the information provided in Figures 10 to 12.

Table 1

LOCATIONS OF NEUTRON DETECTORS IN
TEMPORARY INSTRUMENT TUBES

Tube Number	Channel Designation	
	Upper Section of Tube	Lower Section of Tube
1	I T	IV
2	III T	V; VI
3	II T	II
4	II T	III

Table 2

FUNCTIONS OF THE INSTRUMENTATION CHANNELS FOR INDICATION
OF POWER LEVEL AND OF "NEUTRON FLUX" LEVEL

Channel Number	Intended Information Output ^a	Signal for Control Rod Scram
I ^b	indication of reactor power level, from approximately 1% to 150% of full power (3 Mwt)	high flux level; preset at point corresponding to a power level above 3 Mwt
II	reactor power level, from approximately 5 decades below full power to 150% of full power	high flux level; preset at point corresponding to a power level above 3 Mwt
III	log flux level, and pseudo-period $[(d/dt)\log \text{flux}]$; range of ~ 6 decades in flux, from $1 \times 10^{-4}\%$ to 300% of full power	pseudo-period shorter than 10 sec, when reactor power level is within 10% of full power
IV	reactor power level over range of approximately 7 decades	
V; VI	level of neutron flux, at (the) very low levels (encountered in zero-power work and in initial reactor startup of the 40-assembly core)	
II ^c	level of neutron flux, at the very low levels encountered in zero-power experiments	flux level in excess of full-scale reading
II T	level of neutron flux in zero-power experiments	flux level in excess of full-scale reading
III T	level of neutron flux in zero-power experiments	

^a For Channels I to VI this column lists the normal information output.

^b Channel I was not used during the zero-power work.

^c This instrumentation channel included a recorder of the (linear) flux level.

IV. REMARKS ON THE ZERO-POWER REACTOR PHYSICS PROGRAM

A. Introduction

The entire program of reactor physics experimentation with various core loadings was carried out at the ALPR site, using the actual core structure, control rods, and control rod drives, etc., fabricated for use with the final core loadings. Serious consideration had been given to the advisability and feasibility of undertaking a room-temperature zero-power program in one of the critical facilities at Argonne National Laboratory, Argonne, Illinois, modified to accommodate a core structure with ALPR fuel assemblies and using motor-and-pulley systems to drive the ALPR control rods. However, since the ALPR pressure vessel was to be used for the zero-power "hot" criticals, at temperatures ranging to the design operating temperature ($\sim 421^\circ\text{F}$), and in view of the scheduling of fuel-assembly fabrication and of training of military personnel, it was decided to perform the room-temperature experiments at the site as well.

B. Purpose of the Program (cf. Appendix A)

The primary purposes of the reactor physics zero-power program were to select and to investigate characteristics of an array of fuel assemblies, burnable-poison strips, and control rods that would have: (a) desirable physics characteristics for stable operation at the design power (nominally 3 Mwt); (b) sufficient explicit and inherent reactivity for reactor operation for as long as three years (nominal) at average power; and (c) sufficient control capacity in the fixed system of burnable-poison strips and in the movable control rods to provide an adequate margin of shutdown at any time during the core cycle. A reactor core loading of forty fuel assemblies had been chosen as the design array. It was expected that in the selection of a final reference core any mismatch between the predicted operating reactivity margin and the experimentally observed margin would be compensated by variation of the number and the B^{10} content of burnable-poison strips. An alternative procedure was possible, namely, the variation of the number of fuel assemblies in the core. The design array included a set of five cross control rods. Here too there was flexibility in that four tee rods were available for additional control. The design goals for a 3-Mwt reference reactor included a reactivity margin of approximately $5\frac{1}{2}$ dollars in the fresh reactor operating at 3 Mwt and with equilibrium xenon, and a shut-down margin of at least four dollars in the cold reactor without xenon. As samarium-149 would build in gradually in the first months of reactor operation, approximately one dollar of the initial reactivity margin would be counteracted by neutron absorption by the samarium. It was calculated that reactor operation would result initially in an increase in available reactivity inasmuch as the burnable poison would be lost to neutron absorption at a rate much faster than the fuel, but that the shut-down margin would not be reduced significantly, rather that very likely it would be increased because of an increase in total rod control

capacity as these absorbers were consumed. Therefore it was believed that attainment of a minimum shut-down margin of four dollars in the cold fresh reactor would ensure an adequate margin over the entire core cycle.

An additional goal for the 3-Mwt reference core was control of the cold reactor with the central rod (No. 9) and three off-center cross control rods fully inserted. It was realized that the system would be supercritical in the cold fresh reactor without xenon, with the central rod fully withdrawn, but it was calculated that the forty-assembly reactor could be "held down" with any one off-center cross rod at 30 in. Thus another major purpose of the measurements was to determine the adequacy of shut-down with one or more off-center cross rods stuck out, and to ascertain the critical position of rod No. 9 with the other cross control rods at zero.

Secondary purposes of the zero-power work were to designate a configuration of burnable-poison strips for a full core loading (59 fuel assemblies), and to investigate the physics characteristics of that core with regard to its suitability for operation at higher power levels in another plant. Further discussion of this subsidiary program is included in Appendix A and in Section XII.

C. Operating Instructions for the Nuclear Test Programs

The operating instructions followed during the zero-power reactor physics program are reproduced in their entirety in Appendix D.

D. Outline of Zero-power Reactor Physics Measurements

In Sections V to XII the various subprograms of reactor physics measurements at zero power are discussed in detail. The experiments are discussed essentially in the chronological order in which they were performed, although to some extent this procedure tends to disrupt the continuity of the written presentation. The seeming digression from the initial reference 40-assembly loading to the initial 59-fuel-assembly loading and then back to the initial 40-assembly loading was occasioned by the advisability of obtaining preliminary information on the large core so as to arrange for the fabrication of additional burnable-poison strips of greater B^{10} content to be used in the designation of the "final" 59-assembly core. The final experiments with the initial 40-assembly loading directly led to the specification of the final reference 3-Mwt core, and the zero-power experimentation was continued without further interruption on that core, both at room temperature and at temperatures as high as $\sim 420^\circ\text{F}$. In the interim period, 35 poison strips were made, each containing (nominally) 0.8 g B^{10} , and these were used in the selection of loading No. 60, the final 59-assembly core, in the last set of experiments prior to operation at power.

A concise summary of the pre-power reactor physics measurements is available in the portion of the Table of Contents relating to Section V through XII.

E. Temporary Devices and Procedures Used in the Zero-power Experiments

1. Support Grid for Rod Drives ⁽⁴⁾

For the room-temperature experimentation, a special, temporary grid structure was constructed to rest upon the pressure vessel and provide support for the rod drive mechanisms. Therefore, instead of following the normal procedure of loading fuel assemblies through control rod nozzles, it was possible to carry out the multiplicity of loadings and unloadings through spaces in this supporting grid without disengaging control rods from their drives. The easier access to the core structure served another important purpose in making it possible to insert long instrument tubes into vacant assembly locations for the nuclear instrumentation of the zero-power work (cf. Section III).

The temporary supporting frame was removed prior to the pressurized criticals on the final reference forty-fuel-assembly core.

2. Measurement of "Neutron Flux" Level

The special instrumentation used for experiments with core loadings as large as forty fuel assemblies is described in Section III. Before preliminary experimentation with 59-fuel-assembly core loadings, the temporary instrument tubes were removed from the pressure vessel, and the neutron detectors were placed in the permanent instrument wells. At this point Channel III T was taken out of service.

3. Source Assembly; Source Manipulation

Details of the source assembly are presented in Section II and in Fig. 8. As mentioned in that section, the source was positioned first in assembly location No. 42, where it was left until experiments with 24-assembly core loadings had been completed. For the remainder of the zero-power work the source assembly was situated in location No. 31. The final, permanent position for the operating system is location No. 22, where it is given lateral support by dummy assemblies in locations No. 12 and 21.

During any change in core loading the antimony was left in its beryllium enclosure. The intensity of the neutron source was sufficiently high that it could be observed by the nuclear instruments at a low level of neutron multiplication by the fuel assemblies positioned between the source and those instruments. For the room-temperature zero-power work a simple rod pull device was arranged. A flexible stainless steel cable was attached to the antimony source rod and was passed over a series of pulleys to a hook on the wall in the lower part of the stairwell to the operating room.

Two loops were formed in the cable at this end, one corresponding to the "source-in" position and the other to a "source-out" position. In accordance with the rules adopted for operating procedure during the cold criticals (Appendix D), not until the operating room floor had been cleared of personnel was it permitted to raise the antimony rod. This was done, by a specifically authorized person, by pulling the cable slowly until the "source-out" loop could be placed on the hook. Later, for the zero-power "hot" criticals on the final reference forty-assembly and fifty-nine-assembly cores, a source gripper was attached to the extension rod of control rod drive No. 4, and, with control rod No. 4 in place as a guide, the drive was used to move the source. By these means it was possible to reduce the neutron source level to a point where it offered essentially no interference with measurements of critical rod position and of the positive asymptotic period corresponding to a slight withdrawal of rods during calibration experiments. During preparations for power operation, the rod-pull mechanism was disassembled, leaving the antimony source rod in place in the beryllium block.

4. Procedures for Poisoning the Reactor Water with Boric Acid

At times during the pre-power reactor physics investigations, the reactor was "poisoned" systematically with boron, in the form of boric acid in solution in the reactor water. The neutron absorption effectiveness of the boron made it possible to withdraw control rods to various configurations, ranging to full withdrawal of all rods, and to calibrate rods from these configurations. (Typically, rod calibrations proceeded from a rod-bank critical.) A designated mass of reagent-grade boric acid powder was dissolved in a fixed volume of water and the solution was pumped into the vessel in specified increments. Initially, a uniform distribution of H_3BO_3 in the reactor water was attained by mechanical stirring. Check-criticals were made to insure reproducibility of the measurements; otherwise, better mixing was sought by additional stirring. Later, mechanical stirring was replaced by a bubbler device consisting of a pressure bottle of nitrogen gas and flexible copper tubing with an outlet below the reactor core area. This device introduced the possibility of gas bubble adherence to metal surfaces in the core, and the reproducibility of the critical rod positions was checked after allowing time for diffusion of the nitrogen out of the core. During the pressurized criticals, boron was injected into the reactor water by pumping boric acid solution from the 120-gal backup storage tank with the manually operated pump.

Dilutions were accomplished either by pumping out reactor water and replacing it with clean, demineralized water, or by pumping reactor water through resin beds.

5. Auxiliary System for Heating the Reactor Water

A portable, automatic steam boiler with a capacity of 2,000 lb steam/hr was installed on the operating floor to supply steam for preliminary testing of the various steam-handling components of the plant and for non-nuclear heating of the reactor water. For the latter purpose the boiler was connected to two heating coils installed in the radial water reflector region (the "downcomer" region). This system functioned effectively and proved to be a very useful means of taking the reactor to operating conditions of pressure and temperature without neutron irradiation of the fuel assemblies.⁽⁵⁾

F. Fuel Loading Charts

The loading chart format selected (Fig. 13) was designed to depict the fuel assembly loading to be achieved; to describe the experiment and the environmental conditions, such as the loading configuration of control rods, the water temperature, etc.; and to provide sufficient space for "Remarks" to summarize the data of measurement and interpretation. Each experiment and prospective core loading was reviewed and initialled by the four responsible people indicated on the loading chart. (Formal approval by the Coordinator and the Chief Physicist was required, but each intended loading operation was reviewed carefully by the Chief Operator and the Chief Loader, and their recommendations were taken into full account.)

The number of each fuel assembly was written in the appropriate assembly location (whose number is written in the upper right-hand corner of each square, in matrix notation), using the color ink indicated on the loading chart as corresponding to the thickness of the burnable-poison strip attached. Dummy fuel assemblies in the core were designated by the letter "D," and the orientation of the dummy assembly in its position was shown by lines corresponding to the lineup of the full surfaces. The positions of burnable-poison strips ("B-strips") were shown by drawing lines of the appropriate color in the approximate positions. For convenience, these orientation indices are drawn as dashed or dotted lines in the figures included in the present report (e.g., Fig. 14).

LOADING NO. _____

DATE: _____
 LOADING NO. _____
 EXPERIMENT NO.: _____

TIME: _____
 START _____
 FINISH _____

DESCRIPTION OF EXPERIMENT:

APPROVAL: _____ (COORDINATOR) _____ (CHIEF PHYSICIST)
 REVIEWED: _____ (CHIEF OPERATOR) _____ (CHIEF LOADER)

FUEL ASSEMBLY INVENTORY *

26-mil B-STRIP	BLUE
21-mil B-STRIP	RED
NO B-STRIP	GREEN (BLACK)
(20-mil B-STRIP	GREEN)

CONTROL ROD INVENTORY

	CROSS					TEE			
ROD NUMBER	1	3	5	7	9	2	4	6	8
ROD POSITION (in.)									

WATER TEMPERATURE (°F) _____ H_2O CONCENTRATION (g/liter OF H_2O) _____
 PRESSURE (psig) _____ POWER LEVEL (lb steam/hr) _____

REMARKS: *FOR 59-ASSEMBLY LOADINGS CONTAINING 20-mil B-STRIPS, THE LOADING CHART WAS REVISED TEMPORARILY SO THAT A GREEN LINE DENOTED A 20-mil FULL LENGTH OR HALF-LENGTH BURNABLE-POISON STRIP (B-STRIP).

FIG. 13
 ALPR FUEL LOADING CHART

FIG. 14
MINIMUM CRITICAL (UNPOISONED) LOADING

V. APPROACH TO MINIMUM CRITICAL LOADING: TEN FUEL ASSEMBLIES; NO BURNABLE-POISON STRIPS

With the source assembly in location No. 42, on August 11, 1958 the first loadings of fuel assemblies without burnable-poison strips were carried out with the intention of determining the minimum critical unpoisoned loading. Four sealed tubes serving as temporary containers of neutron flux detectors were in the positions shown in Fig. 14, the loading diagram for the minimum critical. The relative positions of the detectors in the instrument tubes are detailed in Table 1 (Section III).

Three separate sets of readings of the nuclear instruments were recorded during the approach to initial criticality, corresponding to the three control rod configurations given in Table 3. Before any revision in fuel loading was begun, the control rods were arranged in configuration No. I.

Table 3

CONTROL ROD CONFIGURATIONS FOR EXTRAPOLATION TO MINIMUM CRITICAL CORE LOADING

Cross Control Rod Number	Indicated Rod Position (in.)		
	I	II	III
1; 3	0	30	30
5; 7	30	30	30
9	12.0	12.0	30 (if possible)

In Fig. 15 are curves representing the reciprocals of the count rates of the various neutron detectors used, for rod configuration No. III and with the antimony rod in its "out" position. The extrapolations to criticality are in good agreement with the final result.

At this time it was believed that the effectiveness of the source was very small when the Sb was "out," i.e., with the bottom of the antimony rod approximately 7 in. above the top of the beryllium block. This position had been selected so as to simplify the procedure of a remotely operated manual source insertion and withdrawal. In view of the measured effectiveness (~ 0.4 dollar/in.) of control rod No. 9 at its indicated "critical" position (23.6 in.) in the ten-assembly loading, there is little doubt that the loading would have been supercritical in the usual sense, that is, in the absence of the external source, if all control rods had been withdrawn to 30 in.

FIG. 15
APPROACH TO MINIMUM CRITICAL LOADING
(ALL CONTROL RODS AT 30 in.)

A description of the loading sequence is provided in Table 4 and in Fig. 14.

Table 4

APPROACH TO MINIMUM CRITICAL CORE LOADING WITHOUT
BURNABLE-POISON STRIPS

Loading Number	Number of Fuel Assemblies Added	Assembly Number	Assembly Location Number	Total Number of Fuel Assemblies in Core
0	0	Source	42	0
1	4	21	43	1
		22	44	2
		23	45	3
		24	46	4
2	2	27	55	5
		28	56	6
3	1	26	54	7
4	1	25	53	8
5	1	29	65	9
6	1	30	64	10

Notes: 1. "Criticality" was achieved with Loading No. 6, control rods No. 1, 3, 5, and 7 at 30 in., rod No. 9 at 23.6 in., and antimony rod in "out" position.

2. The reactor water temperature was $(108 \pm 1)^{\circ}\text{F}$.

VI. HOLD-DOWN CAPACITY OF CONTROL RODS IN LOADINGS SMALLER THAN FORTY FUEL ASSEMBLIES

The ALPR plant was designed primarily as a prototype for power plants to be set up and refueled in remote locations.⁽¹⁾ Therefore, part of the zero-power work was programmed to yield detailed information about safety margins of fresh core loadings. Two sequences of loadings were performed to test the hold-down capacities of several control rod configurations.

The first sequence was a natural generalization and a continuation of the approach to the minimum critical loading. The core was augmented, essentially symmetrically, by the loading of additional fuel assemblies without burnable-poison strips. It was observed that the central cross control rod could control a 16-assembly core arranged symmetrically with respect to that rod. Of course, the off-center cross rods are not very effective in the small cores since the cadmium absorbs essentially only those neutrons escaping the core as thermalized neutrons or returning to the core from the water reflector as thermal neutrons. When the loading was increased to 20 fuel assemblies, filling positions No. 73 to 76, it was clear that the central rod alone was insufficient for control. The loadings in this sequence are summarized in Table 5.

Note: After the minimum critical loading of ten fuel assemblies was attained, the source assembly was moved to location No. 31, its position for the major portion of the subsequent zero-power experiments.

In the second sequence of core loadings, one B-strip, nominally 0.026 in. thick and containing ~ 0.5 g B^{10} , was taped to the designated side plate of each fuel assembly loaded into the core. The experimental criticality data are summarized in Table 6 for loadings up to, but not including, the initial reference core of forty fuel assemblies.

Table 5

HOLD-DOWN CAPACITY OF CROSS CONTROL RODS IN CORE LOADINGS
WITHOUT BURNABLE-POISON STRIPS

Fuel Assembly Loading ^a			Indicated Critical Rod Positions ^b (in.)					Approximate Differential Reactivity Effect of Rod Withdrawal ^c (dollars/in.)
Loading Number	Total Number of Fuel Assemblies Present	Assembly Locations	1	3	5	7	9	
10	16	square array bounded by locations No. 33,36,63,66	30 0	30 0	30 0	30 0	9.25 approx. 14 to 14.5	1.1 (No. 9)
11	20	rectangular array bounded by locations No. 33,36,73,76	0 0	0 30	0 0	0 0	11.2 6.65	1.2 (No. 9) 0.6 (No. 9)
12	21	same as loading No. 11 plus assembly in location No. 25	0	30	0	0	6.05	0.5 (No. 9)
13	24	rectangular array bounded by locations No. 23,26,73,76 ^d	0 0	24.3 0	0 0	0 0	0 9.25	0.1 (No. 3) 1.3 (No. 9)

^a During these experiments, the reactor water temperature was 104 (± 1)°F.

^b At an indicated position of "0" in., the leading edge of the cadmium is approximately $3\frac{1}{8}$ in. below the bottom of the active core. At 30 in., the leading edge of the cadmium is approximately 1 in. above the top of the active core.

^c The numbers in this column represent literal interpretations of the readings of indicated rod position, and of the three positive periods measured over three decades following an incremental rod withdrawal. In the region of ~1.1 dollars/in. "rod worth," the rod was withdrawn only 0.15 in. to 0.2 in., with an uncertainty of as much as ± 0.05 in. at each position. However, on a number of occasions during the entire zero-power work, reproducibility of critical rod position was tested, and in almost every case agreement to better than ± 0.05 in. was observed when test conditions could be matched closely.

^d Multiplication measurements made with rod No. 3 at 15.0 in. and all other cross rods at zero imply that this rod configuration should not be relied upon to control more than 26 fuel assemblies if the final two assemblies are each worth as much as the 23rd and 24th assemblies added, e.g., if they are added in locations No. 52 and 62.

Table 6

HOLD-DOWN CAPACITY OF CROSS CONTROL RODS IN CORE LOADINGS
WITH ONE 0.5-g-B¹⁰ STRIP PER FUEL ASSEMBLY

Fuel Assembly Loading			Indicated Critical Rod Positions (in.)					Approximate Differential Reactivity Effect of Withdrawal of Rod No. 9 (dollars/in.)
Loading Number	Total Number of Fuel Assemblies Present	Assembly Locations	1	3	5	7	9	
15	16	square array bounded by locations No. 33,36,63,66	30	30	30	30	22.4	~0.7
16	20	rectangular array bounded by locations No. 33,36,73,76	30	30	30	30	15.4	1.0
16 ^a	20		3.1	0	0	0	30	-
			30	0	0	0	20.9	0.65
			30	30	0	0	16.8	0.8
17	24 ^b	rectangular array bounded by locations No. 23,26,73,76	30	30	0	0	15.5	1.15
			30	0	0	0	18.7	1.1
			0	0	0	0	22.2	0.6
18 ^a	32	same as loading No. 17 plus fuel assemblies in locations No. 37,47,48,57, 58, 67,84,85	0	0	0	0	16.5	-

^aDuring these experiments the reactor water temperature was 94°F; in the other cases the temperature was 104°F.

^bAfter loading No. 17, the temporary instrument tubes were moved to different locations within the pressure vessel, shown in Fig. 16.

Additional Notes for Table 6

1. It is estimated that loading No. 15 would be supercritical by approximately 2.5 dollars with all control rods at 30 in.
2. Loading No. 16 could be controlled, marginally, by the four off-center control rods alone at a reactor water temperature of 94°F. It is questionable as to whether this control would be possible at 70°F.
3. Although the actual experiment was not performed, it appears that loading No. 17 could be controlled by the center rod alone. This 24-assembly loading would be supercritical by ~2.3 dollars with the off-center rods fully inserted and the center rod at 30 in.

NUMBERS IN PARENTHESES REPRESENT THE MASS OF B^{10}
IN THE BURNABLE-POISON STRIP

FIG. 16
INITIAL REFERENCE 40-ASSEMBLY CORE
(LOADING NO. 19)

VII. PRELIMINARY EXPERIMENTS WITH THE INITIAL REFERENCE 3-Mwt CORE OF FORTY FUEL ASSEMBLIES (LOADING NO. 19) AT ROOM TEMPERATURE

A. Introduction

An initial reference 40-assembly core loading, with one 0.5-g-B¹⁰ strip taped to a side plate of each assembly, had been selected as the base system with which experiments leading to a final core selection would be made. It had been estimated that with approximately 16 dollars of reactivity controlled by the burnable poison in the operating core, the control rods would be required to control $\sim 5\frac{1}{2}$ dollars in the fresh reactor operating at 3 Mwt and with equilibrium xenon. It was anticipated that the loading of poison strips would be modified to account for reactivity mismatch of design and experiment, and that additional fixed poison could be added by taping half-length strips to the bottom half of selected fuel assemblies in the core.⁽⁶⁾

Two important effects alter the worth of the boron when it is distributed heterogeneously in the core, in the form of full-length strips arranged in the shape of an internal cross in the central control cell (Fig. 9). Although in this arrangement the strip is surrounded by thin, low-absorption regions of water and aluminum, where local thermal neutron flux peaking would normally occur, it is sufficiently gray that the net neutron flux is depressed in its vicinity.

The other effect is directly dependent on the location of a strip relative to the control rod channels. To maximize the control rod worth in the cell reactor, and also to reduce the burnup rate of the boron during the first year or so of power operation, the reference fuel assembly orientation shown in Fig. 9 was chosen. With this orientation, when control rods are raised the thermal neutron flux distribution shifts so that the relative flux level in the poison strips is reduced further. An indication of the magnitude of this flux-shift effect on the reactivity control by the full-length poison strips is provided in Sections VII and X.

The final reference forty-assembly core was determined on the basis of these experiments and other calibrations of the control system. In Section VII some preliminary control system calibrations are described, and data from assembly-omission experiments are presented.

Loading No. 19 is shown schematically in Fig. 16. During experiments with this core loading the four tee control rods were fully withdrawn.

B. Control System Calibrations

With cross rods No. 3, 5, and 7 at 0 in., the indicated critical position of rod No. 9 was determined as a function of the indicated position of rod No. 1; the data are graphed in Fig. 17 (curve "B"). Subsequent to these calibration experiments, it was observed that in the "out" position a 7-in. separation of the antimony rod and the beryllium block was insufficient, and an additional 2-in. separation was tested. As a precaution against source effects on measurements of differential worths, the source-puller was reset so that in its "out" position the antimony was approximately 4 ft from the beryllium block. The calibration was then repeated; the results are depicted in Fig. 17 as curve "A." Curve A is fitted to a much larger number of measurements than is curve B; therefore the two curves should not be compared in detail. Table 7 contains some additional criticality data for loading No. 19.

FIG. 17
RELATIVE CRITICAL POSITIONS OF CONTROL RODS NO. 1 AND 9
(LOADING NO. 19 AT 94°F; RODS NO. 3, 5, AND 7 AT ZERO)

Primarily to estimate the shut-down capacity of the five cross control rods in the initial reference 40-assembly loading (loading No. 19), a "three-rod" method was adopted for an approximate calibration of rod No. 9 below 12.85 in. In this scheme, rods No. 3 and 7 were held at 0 in., and data were obtained for the variation of the indicated critical position of rod No. 9 versus the position of rod No. 1. Two sets of pairings were determined, one with rod No. 5 at 0 in., and the other with No. 5 at 6.0 in. or at 7.85 in., as shown in Table 8, and, in part, as curve A of Fig. 17.

Table 7

INDICATED ROD POSITIONS FOR CRITICALITY
OF LOADING NO. 19

Indicated Rod Position (in.)					Reactor Water Temperature (°F)	Source "Out" Position
1	3	5	7	9		
9.2	9.2	9.2	9.2	9.2	94	old (~7 in.)
12.5	12.5	12.5	12.5	0	94	old
30	0	0	0	4.5	94	old
30	7.05	0	0	0	94	old
30	0	7.3	0	0	94	old
0	0	0	0	12.6	94	old
0	0	0	0	12.85	93	~ 9 in.
0	0	0	0	12.85	93	new (~4 ft)
0	0	0	0	12.90	97	new
6.05	0	0	0	12.6	93	new
0	0	6.0	0	12.6	93	new

Table 8

CALIBRATION OF RODS NO. 1 AND 9 BY PARTIAL WITHDRAWAL OF ROD NO. 5
(Loading No. 19 at 93°F)

Curve A		Curve B	
Rod No. 1 (in.)	Rod No. 9 (in.)	Rod No. 1 (in.)	Rod No. 9 (in.)
Rods No. 3, 5, and 7 at 0 in.		Rods No. 3 and 7 at 0 in.; rod No. 5 at 6.0 in.	
0	12.85	0	12.6
6.05	12.6	6.0	12.30
7.85	12.30	7.85	12.0
9.15	12.0	9.15	11.7
10.2	11.7	11.2	11.2
11.2	11.45	12.15	10.95
12.15	11.2	13.0	10.7
13.0	10.95	13.8	10.4
13.8	10.7	14.6	10.15
14.6	10.4	15.3	9.9
15.3	10.15	Rods No. 3 and 7 at 0 in.; rod No. 5 at 7.85 in.	
16.05	9.9		
17.5	9.4		
18.8	8.9		
20.2	8.4		
21.5	7.85		
22.95	7.25		
24.45	6.6		
26.2	5.9	16.05	9.4
		17.5	8.9
		18.8	8.4
		20.2	7.85
		21.5	7.25
		22.95	6.6
		24.45	5.9

The reactivity worths of the partial withdrawals of rod No. 5 were calibrated with respect to the indicated position of rod No. 9; some of the data

FIG. 18
REACTIVITY EFFECT OF MOVING CONTROL ROD NO. 5 FROM 0 IN. TO 6 IN. vs. INDICATED CRITICAL POSITION OF ROD NO. 9 (LOADING NO. 19 AT 93°F; RODS NO. 3 AND 7 AT ZERO; ROD NO. 1 AT POSITION SHOWN IN FIG. 17)

are presented in Fig. 18. With rod No. 9 at 5.9 in. or at 12.0 in., a pull of No. 5 to 7.85 in. was found to be worth very nearly twice the worth of the withdrawal of No. 5 from 0 to 6.0 in. Accordingly, in the analysis of rod worth it has been assumed that this latter relationship applies over the entire range from 5.9 in. to 12.85 in. In Fig. 19 is a curve of reactivity (dollars) vs. the asymptotic positive reactor period, based on constants for the delayed neutron groups reported by Keepin, Wimett, and Ziegler.⁽⁷⁾ Using a literal interpretation of the rod calibration data, the differential and integrated worths of rod No. 9 were computed and are shown in Fig. 20. The major result is that

the control margin in the cold (93°F), fresh, initial 3-Mwt reference loading is estimated to be 5(±1) dollars. This is the additional reactivity hold-down capacity of rod No. 9 from 0 to 12.85 in. with all other cross rods fully inserted, or, equivalently, the worth of the five-rod bank from 0 to ~9.3 in.

In the analysis it was assumed that the integrated worth of rod No. 9 from 0 to 12.85 in., with the four other cross control rods positioned at zero, could be determined by measurements involving the partial withdrawals of two of these four off-center rods. The accuracy of the procedure therefore hinges importantly on the extent of shadowing of the center rod by the others, and on the extent of shift in local importance of rod absorptions when rod control of a sector of the core is reduced relative to that of other sectors. Another uncertainty arises from the calibration procedure itself in those cases where the criticality or positive-period measurements were made by moving rod No. 9 after all other rods were in place. A net uncertainty of as much as ± 0.1 in. in the change in position of rod No. 9 is conceivable, on grounds already discussed, but the deviations of the data from smooth curves imply a much smaller uncertainty in the differential worth at any given point, unless some unexpected consistent error is present. A maximum uncertainty of $\pm 20\%$ is estimated on this basis for the differential measurements, and the integration process should have reduced this uncertainty markedly.

FIG. 19
REACTIVITY vs. ASYMPTOTIC REACTOR PERIOD
($\beta=0.006398$, $\lambda^*=5 \times 10^{-6}$ sec)

FIG. 20
ESTIMATED REACTIVITY EFFECT OF MOVING CONTROL ROD NO. 9
(LOADING NO. 19 AT 94°F; RODS NO. 3 AND 7 AT ZERO;
ROD NO. 1 AT POSITION SHOWN IN FIG. 17)

C. Reactivity Effect of Fuel Assembly Omission

The procedure proposed to effect a loading change in the reactor at an ultimate site is to remove or to introduce assemblies through control rod nozzles in the pressure vessel lid. It is possible, though unlikely, that a fuel assembly location could be left unfilled inadvertently. It was of interest, therefore, to ascertain the effect on reactivity of inserting a fuel assembly into an otherwise full core array. The actual experiment consisted of measuring the change in the critical position of rod No. 9 upon removal of a fuel assembly from the forty-assembly core, as a function of the radial location of the assembly. The results are given in Table 9, uncorrected for the effect of assembly removal on the calibration of the control system.

Table 9

REACTIVITY LOSS UPON REMOVAL OF A FUEL ASSEMBLY FROM THE OTHERWISE FULL INITIAL REFERENCE FORTY-ASSEMBLY CORE (Loading No. 19)

Location of Assembly	Change in Indicated Critical Position of Central Control Rod ^a (in.)	Estimated Reactivity Effect ^b (dollars)
47	1.1	-1.4
36	1.55	-1.9
46	3.2	-3.6
45	6.7	-6.6

^a Initial critical rod configuration: Rods No. 1, 3, 5, and 7 at zero; rod No. 9 at 12.90 in.

^b By positive-period measurements, the differential reactivity effect of withdrawing rod No. 9 with the other four rods at zero was determined to be ~ 1.3 dollars/in. at 12.9 in., and 0.66 dollar/in. at 19.6 in. The net differential reactivity effect was assumed to vary linearly between these two values in obtaining the numerical values tabulated in this column.

D. Preliminary Calibrations of Burnable-poison Strips

Normally, curves entitled "differential control rod worth" are presented, based on data of changes in reactivity following changes in rod position, for example as determined by measurements of the asymptotic positive period resulting from incremental rod withdrawal. Frequently, much of the reactivity change arises from changes in neutron flux distribution, resulting in changes in neutron leakage or in neutron absorption by

other parasitic absorbers relative to fuel absorption. Perhaps it would be more appropriate to label these curves "differential reactivity effect of moving rods" than to identify them as "differential rod worth." This distinction is quite pronounced in the case of ALPR, where the relative neutron absorption rate in the burnable-poison strips is a strongly varying function of the vertical penetration of the core by the control rods. The reference configuration of greatest interest for calibrations of the poison strips is the operating core at design power, pressure, and temperature, and with equilibrium conditions of xenon-135 and samarium-149. In lieu of such measurements, calibrations of the strips were undertaken in the cold fresh reactor with enough boric acid dissolved in the reactor water to permit withdrawal of the control rods as a bank to the vicinity of the banked rod position anticipated for the operating reactor (cf. Section X-C).

Preliminary calibrations of the poison strips were made in the cold reactor, with the off-center rods fully inserted, both to refine the calibration technique for the detailed experiments and to obtain data of strip worth in an environment close to that of full rod insertion. The experimental procedure consisted of determining the critical indicated positions of rod No. 9 with all strips in place and with the one poison strip removed from a single fuel assembly in each of the various representative locations. An average differential reactivity effect of 1.25 dollars/in. was assumed for rod No. 9 in computing the reactivity effects of strip modifications. When poison strips in outlying core locations were removed, only a small change in position of rod No. 9 was required to maintain criticality, with large attendant uncertainties in the differential rod motion conceivable. In the final calibrations these uncertainties were reduced by working with representative units of four fuel assemblies (cf. Section X-C).

The major uncertainty in the interpretation of the preliminary calibration data arose from a seemingly anomalous datum noted in Table 10A, namely, the measurement of a critical position of 12.75 in. for rod No. 9 after all poison strips were in place. This may be compared with the value 12.90 in. measured early that morning and also early in the following day. In Table 10B the inferred reactivity worths are stated for base critical positions of 12.75 in. and 12.90 in., assumed to be independent of temperature over the range from 96°F to 99°F. The total reactivity worth of the forty 0.026-in.-thick burnable-poison strips is estimated to be $22(\pm 4)$ dollars in the cold fresh reactor with the four off-center cross control rods at zero and rod No. 9 at ~ 12.9 in. The measurements do not appear to be sufficiently precise to permit a narrowing of the stated range of uncertainty.

These particular strip calibrations were not repeated, since a more accurate determination of strip worth in this environment was not required in the specification of the final reference core. However, it was established that a sizeable difference in strip worth existed for the two environments of measurement, as may be seen by comparison of the total worth of $22(\pm 4)$ dollars, given above, with the value 12.7 dollars obtained in the later calibration of the forty burnable-poison strips (cf. X-C).

Table 10
PRELIMINARY REACTIVITY CALIBRATIONS OF
0.5-g-B¹⁰ STRIPS IN LOADING NO. 19

A. Uninterpreted Data

Assembly Location of Poison Strip Removed	Mass of B ¹⁰ in Strip (g)	Indicated Critical Position of Rod No. 9 ^a (in.)	Temperature of Reactor Water (°F)	Time of Measurement
no strip removed	-	12.90	97	morning
45	0.50	12.05	96	} afternoon; within 3-hour period
46	0.50	12.1	96	
36	0.51	12.15	99	
37	0.48	12.55	98	
47	0.48	12.50	97	
48	0.47	12.70	-	
all strips in place	-	12.75	97-98	
all strips in place	-	12.9	98	next morning

B. Interpretations of Data

Assembly Location	Reactivity Worth of Strip ^b (dollars)		Number of Symmetrical Locations
	Base Position of Rod No. 9: 12.75 in.	Base Position of Rod No. 9: 12.90 in.	
45	0.88	1.06	4
46	0.81	1.0	8
36	0.75	0.94	4
47	0.31	0.50	8
37	0.25	0.44	8
48	0.06	0.25	8
Total for			
40 Strips	18.0	25.5	
Best Estimate	22(±4) dollars		

^a Rods No. 1, 3, 5, and 7 were kept at zero throughout.

^b The reactivities listed were computed on the basis of a differential reactivity effect of 1.25 dollars/in. for rod No. 9 in the vicinity of 12.9 in.

VIII. EXPERIMENTS ON THE INITIAL CORE OF FIFTY-NINE FUEL ASSEMBLIES (LOADING NO. 38)

A. Introduction

The initial fifty-nine-assembly core (loading No. 38) was attained by adding a total of nineteen fuel assemblies to the initial reference forty-assembly loading (No. 19); to each of the 19 assemblies was attached one full-length, 0.021-in.-thick burnable-poison strip containing nominally 0.4 g B^{10} . The additional assemblies were placed in the peripheral regions of the core structure. Figure 21 is a loading diagram for this core.

FIG. 21
INITIAL 59 - ASSEMBLY CORE
(LOADING NO. 38)

It was observed that the five cross control rods could not quite control loading No. 38. Even if 0.026-in.-thick strips had been used, instead of the 0.021-in.-thick strips, it is doubtful that such control would have been possible. (At best, the margin of control with the five cross rods alone would have been inadequate.) The decision to make provision for four tee rods had been made early in the core design precisely because there was a real question as to whether the five cross rods could provide a shutdown margin of at least 3 or 4 dollars in the cold 59-assembly core unless still more B^{10} was added in the core.

On the basis of experimental results described in this section, a revised boron loading was selected, and it was possible to dispense with the tee rods. However, an additional margin of safety would be available if all nine control rods were used. Unlike the situation in the final reference 40-assembly core, it was doubtful that any four-rod configuration could control the cold final 59-assembly loading. Therefore, in the event of sticking of one rod in its out position in a five-rod core, additional control would be required for cold shutdown, for example by the injection of boric acid solution from the soluble poison tank.

B. Critical Control Rod Configurations

In Table 11 a number of critical control rod configurations are listed for the initial 59-assembly core loading. In some cases positive-period measurements of differential worth were made; these data are included in the table. During this phase of the experimental work, the reactor water temperature was maintained at $94(\pm 1)^{\circ}\text{F}$.

Table 11

CRITICAL CONTROL ROD CONFIGURATIONS FOR LOADING NO. 38

(Indicated Rod Positions are Tabulated in In.)

Cross Rods					Tee Rods				Differential Reactivity Effect of Moving Rod (dollars/in.)
1	3	5	7	9	2	4	6	8	
8.6	8.6	8.6	8.6	8.6	0	0	0	0	0.9 (No. 9)
0	0	0	0	0	24.7	24.7	24.7	24.7	-
15.8	0	0	0	0	0	0	0	0	0.7 (No. 1)
0	0	0	0	12.8	0	0	0	0	1.5 (No. 9)
0	0	0	0	11.1	0	0	0	30	1.2 (No. 9)
0	0	0	0	9.0	0	0	30	30	-
0	0	0	0	6.95	0	30	30	30	-
0	0	0	0	0	19.8	30	30	30	-
10.5	10.5	10.5	10.5	0	0	0	0	0	-
8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	-
8.0	8.6	8.6	8.6	8.8	0	0	0	0	0.3 (No. 1)

Three important observations on control rod capacity should be noted:

1. The partial insertion of one tee rod was required to control loading No. 38.
2. All five cross control rods were required for shutdown of the cold fresh reactor, regardless of the positions of the tee rods.
3. It is estimated that the full insertion of all nine rods would have made loading No. 38 subcritical by $6.0(\pm 0.8)$ dollars.

C. Reactivity Effect of Improper Orientation of a Fuel Assembly

Fuel assembly No. 23 was reoriented in location No. 45 to ascertain the reactivity effect of an accidental improper assembly orientation. The results are tabulated in Table 12.

Table 12

INDICATED CRITICAL POSITION OF ROD NO. 9 VS.
ORIENTATION OF ASSEMBLY IN LOCATION NO. 45
(LOADING NO. 38)

Control Rod Number	Indicated Rod Position (in.)		
	Orientation of Assembly		
	Normal	90° Counterclockwise	180°
1;3;5;7	0	0	0
9	6.9	6.05	5.25
2	0	0	0
4;6;8	30	30	30

Notes: If the control system calibration were unaffected by the 90° counterclockwise rotation of the assembly, the 0.85-in. incremental insertion of rod No. 9 would correspond to ~ 0.3 dollar in reactivity. No such estimate is attempted for the 180° rotation, for the overall worth of rod No. 9 is reduced in the presence of a contiguous poison strip and the net reactivity effect cannot be obtained from the experimental data.

D. Heating Experiments

Primarily to test the non-nuclear heating procedure for later detailed experimentation with the final reference core loadings, and secondarily to determine (approximate) temperature coefficients in loading No. 38, the reactor water temperature was raised incrementally

by passing steam from the boiler into the heating coils positioned in the reflector region around the core structure. Reactivity data obtained from these measurements are listed in Table 13.

Table 13

REACTIVITY DATA FROM HEATING EXPERIMENTS
WITH LOADING NO. 38

Temperature of Reactor Water (°F)	Indicated Critical Rod Position (in.) Rod Number					Differential Reactivity Effect of Moving Rod (dollars/in.)
	1	3;5;7	9	2	4;6;8	
94	0	0	12.8	0	0	1.5 (No. 9)
113	0	0	13.15	0	0	1.4 (No. 9)
146	0	0	13.85	0	0	1.3 (No. 9)
194	0	0	15.0	0	0	1.3 (No. 9)
94	8.0	8.0	8.0	8.0	8.0	-
113	8.2	8.2	8.2	8.2	8.2	1.0 (No. 9); 0.4 (No. 1)
145	8.5	8.5	8.55	8.5	8.5	1.0 (No. 9); 0.4 (No. 1); 0.09 (2 in. pull of No. 2)
189	8.9	8.9	8.95	8.9	8.9	1.1 (No. 9)

It is important to distinguish between: (1) the change in reactivity of a fixed system whose temperature is changed, and (2) the net change in excess reactivity available to the system, e.g., the excess reactivity held by the control system, when the temperature is changed. For example, with control rods left in place it is clear (cf. Table 13) that loading No. 38 has a large negative (average) temperature coefficient in the sense of (1) above, and it is possible to estimate this coefficient with reasonable accuracy ($\sim 3.0 \times 10^{-2}$ dollars/°F from 94°F to 194°F). The assignment of an accurate value to the average temperature coefficient as defined in (2) requires a very accurate calibration of the control system at the two extreme temperatures and over the entire region of penetration of rods. It is conceivable that a cold reactor controlled by neutron-absorbing rods could have a negative temperature coefficient in the sense of (1) and yet gain excess available reactivity upon a rise in temperature. The former definition is related to

questions of reactor safety, for example whether or not a rise in temperature of the system would be autocatalytic if control rods were left in place. The latter concept comes to the fore in specifying the excess reactivity required in the cold reactor to permit a rise in temperature to design conditions.

Whenever sufficient information is available to make such estimates feasible, values of "average temperature coefficient" will be quoted for each of the two concepts.

E. Reactivity Effect of Moving Control Rods in the Cold Reactor

Various methods have been devised for detailed reactivity calibrations of control rods in reactors. An important part of each method is the adoption of some procedure to permit the motion of the rod(s) being calibrated to the point of calibration in a critical medium. Subsequently, some technique is used to determine the reactivity effect of a differential change in rod position, e.g., the measurement of a positive asymptotic reactor period corresponding to a slight withdrawal of the rod(s). In each case the value of differential reactivity measured is affected by the calibration procedure. In the early loadings of ALPR, two methods were used to obtain preliminary information about the effectiveness of control rod motion; these were based on the variation of core size and the relative motion of other rods. Smaller perturbations are encountered if a soluble neutron absorber is added to the reactor water to reduce the requirements of rod control. Boric acid was used in the ALPR, making it possible to calibrate the rods from various banked positions and to estimate the total excess reactivity available in the cold reactor. At the same time, the effectiveness of the boric acid system for auxiliary control was tested.

Boric acid solution was added to the reactor water in measured increments, the nine control rods were withdrawn to corresponding banked critical positions, and positive-period measurements were made to calibrate the rods. Three curves are shown in Fig. 22, representing the differential reactivity effects of moving the central rod, the four off-center cross control rods, and the four tee rods, from a nine-rod bank. During this series of experiments the reactor water temperature varied between 85°F and 94°F; no corrections have been made for this effect.

The integrals of the differential curves from 8.0 in. to 30 in. have been estimated by numerical integration of the smooth curves shown in Fig. 22, yielding 9.2 dollars for rod No. 9, 14.8 dollars for rods No. 1, 3, 5, and 7, and 2.8 dollars for rods No. 2, 4, 6, and 8. The shut-down margin of the nine control rods is estimated to be 6 dollars.

In appendix E the significance of the procedure of integrating the differential curves is discussed.

FIG. 22
DIFFERENTIAL REACTIVITY EFFECT OF MOVING
CONTROL RODS FROM THE 9-ROD BANK
(LOADING NO. 38 AT $\sim 94^{\circ}\text{F}$)

F. Reactivity Effect of Boric Acid in the Reactor Water

The concentration of H_3BO_3 in the reactor water was determined by analyzing representative samples of the reactor water and also by accounting for the masses of H_3BO_3 added, the volumes of water in the pressure vessel, and dilutions. The curve shown in Fig. 23 was obtained by equating the total reactivity controlled by the boric acid to the integral of the differential reactivity effect of moving the nine control rods from 8.0 in. to the corresponding critical rod bank position. The average worth of the boric acid in solution was computed to be 8.7 dollars per gram H_3BO_3 per liter of reactor water over the range of concentrations used (or ~ 2.3 dollars/(g H_3BO_3 /gal), in terms of the units used in Fig. 23).

FIG. 23
REACTIVITY EFFECT OF BORIC ACID IN THE REACTOR WATER
(LOADING NO. 38 AT $\sim 94^\circ F$)

IX. INTERMEDIATE CORE LOADINGS (48 OR 51 FUEL ASSEMBLIES)

Two core arrays intermediate in size to the 40-assembly and 59-assembly loadings were examined very briefly, to accumulate more information about the hold-down capacity of the five cross control rods.

A 51-assembly loading (No. 42) was achieved by removing eight corner assemblies from loading No. 38. Two critical control rod configurations were noted at a reactor water temperature of $90^\circ F$ and with rods No. 2, 4, 6, and 8 at 30 in.;

- (a) rods No. 1, 3, 5, and 7 at zero;
rod No. 9 at 8.15 in. (0.70 dollar/in. for rod No. 9)
- (b) rods No. 1, 3, 5, 7, and 9 at 5.7 in.;
0.33 dollar/in. for rod No. 9 at 5.7 in.
0.17 dollar/in. for rod No. 1, averaged from 5.7 in. to 7.2 in.

The shutdown margin of the five cross rods was estimated to be $1.5 (\pm 0.3)$ dollars.

The 48-assembly loading was the same as the 51-assembly loading except that dummy fuel assemblies were placed in locations No. 36, 68, and 61 in the stead of fuel assemblies. It is estimated that the 5-rod shutdown margin in this loading was ~ 3 dollars.

X. ADDITIONAL EXPERIMENTS WITH THE INITIAL REFERENCE FORTY-ASSEMBLY CORE

A. Reactivity Effects of Moving Control Rods

A forty-assembly, five-cross-rod core was reconstituted, essentially the same as loading No. 19 but designated loading No. 45 for continuity of the numbering sequence. (The critical 5-rod bank position in loading No. 19 was 9.2 in. at 95°F. With the four off-center cross rods at zero, the reactor was critical with rod No. 9 at ~12.9 in.) In a manner similar to that used in loading No. 38, boric acid was introduced into the reactor water and the differential reactivity effect of moving control rods was measured for each of several rod configurations. The calibration curves are shown in Figs. 24 and 25. The curve for the differential reactivity effect of moving rod No. 9 from the five-rod bank was noted to be quite similar to that for rods No. 1, 3, 5, and 7, and therefore the latter curve is not shown. Instead, in Fig. 26, the differential reactivity effect of moving the five cross control rods from a critical banked position is graphed. In that same figure is shown the curve obtained by integrating the differential curve from the generic vertical position, Z, to 30 in.; this curve was extrapolated to zero inches on the basis of earlier theoretical analyses.

FIG. 24
DIFFERENTIAL REACTIVITY EFFECT OF (A) MOVING CONTROL RODS
NO. 1, 3, 5, AND 7 AS A BANK, AND (B) MOVING CONTROL ROD NO. 9
(LOADING NO. 45 AT ROOM TEMPERATURE;)

FIG. 25
DIFFERENTIAL REACTIVITY EFFECT OF MOVING
CONTROL ROD NO. 9 FROM 5-ROD BANK
(LOADING NO. 45 AT ROOM TEMPERATURE)

During the experimental program, a continuing investigation failed to reduce the range of uncertainty in rod position below ± 0.05 in. In most cases, in which it was clear that such conditions as reactor water temperature and soluble-poison concentration were unaltered, reproducibility of the critical indicated rod position was possible to within ± 0.03 in. Nevertheless, to show the possible errors in the calibration curves, the effect of a ± 0.05 -in. uncertainty at each position is plotted at each point of measurement in Figs. 24 and 25. The solid lines in those figures are simply smooth-curve fittings to the data.

In the majority of the positive-period measurements, several periods were timed over two or three decades of neutron flux rise and well above the region of interference by the antimony-beryllium source. The uncertainty in asymptotic period results in an additional uncertainty in differential worth, estimated to be less than six per cent of the differential worth inferred.

FIG. 26
REACTIVITY EFFECTS OF MOVING CONTROL RODS
AS A 5-ROD BANK
(LOADING NO. 45 AT ROOM TEMPERATURE)

B. Reactivity Effects of Boric Acid in the Reactor Water

For indicated rod positions greater than ~ 25 in. an effect peculiar to the calibration procedure became prominent. When the rod was moved for a positive-period measurement, the movement of the leading edge of the rod follower left a gap in a statistically important region of the core that was filled by poisoned water, leading to a negative reactivity contribution. As a consequence, in some cases it was observed that the withdrawal of a rod to 30 in. resulted in an asymptotic period longer than that resulting from the rod withdrawal to only 28 in.! An attempt has been made to take this into account in drawing smooth curves of differential reactivity effect, but it is believed that these curves tend to underestimate the data that would be obtained if the calibration could be performed without introducing this special distortion. If one considered these curves to be representative of the actual circumstances of the calibration, including the aforementioned effect of the boric acid poisoning, then the differential curve for the four-rod bank should be slightly negative in the neighborhood of 30 in.

An effect of a similar nature is present in all such calibrations. The neutron poisoning of the fuel zone by the boric acid reduces the mean distance a thermal neutron travels before absorption, and therefore fewer of the thermalized neutrons reach the control rods. No corrections have been attempted for these perturbations. They are believed to be small except possibly in the neighborhood of 30 in., a region that contributes little to the total effectiveness of the control rods.

In Fig. 27 are two curves relating to the boric acid poisoning. One shows the correspondence between the critical indicated position of the 5-rod bank and the concentration of H_3BO_3 in the reactor water. The other curve is a linear fit to the data of boric acid effectiveness versus boric acid concentration [~ 2.3 dollars/(g H_3BO_3 /gal)]; the datum points were computed from the integral curve of Fig. 26.

FIG. 27
EFFECT OF BORIC ACID CONCENTRATION ON REACTIVITY
AND ON CRITICAL POSITION OF 5-ROD BANK
(LOADING NO. 45 AT ROOM TEMPERATURE)

C. Reactivity Controlled by Burnable-poison Strips with Control Rods Banked at ~ 20 in.

In Section VII-D the preliminary poison-strip calibration in loading No. 19 was described. During those experiments the four off-center cross rods were fully inserted (at indicated zero) and the calibration data were inferred from the calibrations of rod No. 9 in the vicinity of 12.9 in. It was estimated that with that rod configuration the forty 0.5-g B^{10} -strips controlled $22(\pm 4)$ dollars.

The strip calibration summarized in Table 14 was performed with the five cross rods banked in the proximity of their anticipated position in the operating system, and with enough boric acid dissolved in the reactor water to compensate for the reduction in rod control. To reduce the uncertainties of the change in indicated critical positions of the control rods, in every experiment one strip was removed from each of four essentially symmetrical locations; unless otherwise indicated, the strips were replaced before the next experiments. However, a different uncertainty was introduced in this calibration, for at this position the differential reactivity effect of rod withdrawal drops sharply.

It may be seen that the forty 0.5-g-B^{10} strips were much more effective in the reactor configuration in which most of the control rods were fully inserted. A somewhat simplified but accurate explanation for this difference is that the average thermal neutron flux in these poison strips is increased relative to the average flux in the fuel when control rods are inserted. This is illustrated in Fig. 28, in terms of the results of one-group, one-dimensional, infinite-slab, multiregion (DSN) S_{16} calculations of the thermal neutron flux distribution in a horizontal traverse of a central control cell. In these computations a step-function source of thermal neutrons was assumed; the source level in each region was made proportional to the volume fraction of water in the region. Two cases are described in Fig. 28; in each case a central 0.5-g-B^{10} strip and a side 0.4-g-B^{10} strip are present, and the two curves differ only because of the presence or absence of the control rods. Aluminum-nickel rod followers are in place, in the latter case. The relative rates of thermal neutron absorption in the fuel zones and in the poison strips are listed in Table 15. The influence of the rod followers is shown in that table.

Note: The reactivity effect of the strips cannot be inferred from Table 15 since the computations were for slab geometry, with approximately half of the relative strip volume "included."

Table 14

REACTIVITY CALIBRATION OF BURNABLE-POISON STRIPS
"IN" LOADING NO. 45

(Base Rod Bank: 20.12 in.)

Fuel Assembly Locations	Strip Modifications ^a	Estimated Reactivity Change (dollars)
44; 45; 54; 55	0.5-g-B ¹⁰ → 0.4-g-B ¹⁰	+0.38
44; 45; 54; 55	0.5-g-B ¹⁰ strips removed.	+2.8
44; 45; 54; 55	Half-length poison strip containing ~0.2 g B ¹⁰ added to bottom half of the other side plate. ^b	-1.5
43; 46; 53; 56	0.5-g-B ¹⁰ strips removed.	+2.2
33; 36; 63; 66	0.5-g-B ¹⁰ strips removed.	+2.0
42; 47; 52; 57	0.5-g-B ¹⁰ strips removed.	+0.85
32; 37; 62; 67	0.5-g-B ¹⁰ strips removed.	+0.55
	Twelve more half-length strips each containing ~0.2 g B ¹⁰ added to remainder of the central sixteen assemblies. ^c	-3.5 to -3.9

^aStrip-equivalence relationships may be observed in terms of the following assembly locations:

43 = 35 = 56 = 64	42 = 74 = 57 = 25
34 = 53 = 46 = 65	41 = 84 = 58 = 15
44 = 45 = 54 = 55	51 = 85 = 48 = 14
33 = 36 = 63 = 66	23 = 62 = 76 = 37
32 = 73 = 67 = 26	24 = 52 = 75 = 47

^bThese four half-length strips were left in position for the remainder of the strip calibrations. Boric acid solution was added to lower the critical rod bank to ~20.2 in.

^cThis configuration is loading No. 57, the core composition of the final reference forty-assembly reactor.

<u>Summary:</u>	Forty full-length 0.5-g-B ¹⁰ strips	12.7 dollars
	Sixteen half-length 0.4-g-B ¹⁰ strips	5.0 to 5.4 dollars
	Total	~17.9 dollars

FIG. 28

ONE-GROUP, ONE-DIMENSIONAL (DSN) S_{16} COMPUTATIONS
OF THERMAL NEUTRON FLUX IN SLAB GEOMETRY: CENTRAL
CONTROL CELL OF LOADING NO. 57 AT ROOM TEMPERATURE

Table 15

ONE-GROUP, ONE-DIMENSIONAL CALCULATIONS IN SLAB GEOMETRY^a
 OF RELATIVE ABSORPTION OF THERMAL NEUTRONS BY BURNABLE-
 POISON STRIPS AND BY EFFECTIVE FUEL ZONES IN A COLD CONTROL
 CELL (REPRESENTATIVE OF CENTRAL CONTROL CELL
 IN LOADING NO. 57)

Control-system Configuration	Fractional Absorption of Source Neutrons				Change in Fractional Absorption by Effective Fuel Zones ^b
	Effective Fuel Zones	Burnable-poison Strips			
		0.5-g-B ¹⁰	0.4-g-B ¹⁰	Total	
Control rods in; strips in ^c	0.8025	0.0475	0.0217	0.0692	} 0.0519
strips out	0.8544				
Control rods out and rod followers in;					
strips in	0.8754	0.0477	0.0386	0.0863	} 0.0758
strips out	0.9512				
Control rods out and rod followers out;					
strips in	0.8654	0.0460	0.0441	0.0901	} 0.0787
strips out	0.9441				

^aOne-group (DSN) S_{16} computations; the neutron source distribution was chosen to be uniform in each region and proportional to the volume fraction of water in that region.

^bWhen burnable-poison strips are removed, the average relative thermal neutron flux is raised in adjacent parasitic absorbers. Therefore, the thermal utilization of the fuel zones does not increase by the full amount of the strip absorption.

^cOne centrally located 0.5-g-B¹⁰ strip; one side 0.4-g-B¹⁰ strip.

D. Experimental "Neutron Flux" Distributions in a Central Control Cell (Loading No. 46)

Nine bare gold wires were inserted for measurements of the axial neutron flux shape under various conditions of control rod penetration in loading No. 46 (essentially the initial reference forty-assembly core). The reactor water was poisoned with boric acid to permit the withdrawal of control rod No. 9 to an indicated position of 21.9 in. and the four off-center cross rods to 14.0 in. In Fig. 29 are shown radial distributions of wire activation through the center of fuel assemblies in locations No. 54 and 64, normal to a control channel, at three different vertical levels. Since most of the activity arises from the capture of thermal neutrons, the figure has been labeled "Horizontal Flux Plots." The top curve is representative of the thermal neutron flux shape six inches above the bottom of the active core. Note that this shape and that in the plane sixteen inches above are quite similar in the neighborhood of the central control rod, differing primarily because of the presence or absence of absorber of control rod No. 3. At the vertical level of 20 in. above the bottom of the active core, the flux is depressed in the vicinity of rods No. 3 and 9 because both rods are inserted beyond this point.

INDICATED ROD POSITIONS: RODS NO. 1, 3, 5, AND 7 AT 14.0 in.; ROD NO. 9 AT 21.9 in.

FIG. 29
HORIZONTAL FLUX PLOTS
(LOADING NO. 46 AT ROOM TEMPERATURE)

XI. EXPERIMENTS WITH THE FINAL REFERENCE 3-Mwt CORE OF FORTY FUEL ASSEMBLIES (LOADING NO. 57)

A. Introduction and Summary

From analysis of the control system calibrations in loadings No. 45 and 46, it was clear that the shutdown margin in the cold fresh initial reference 40-assembly reactor was adequate (approximately 5 dollars), and that the forty 0.5-g-B¹⁰ strips would control less than the ~16 dollars (~11%) in reactivity (stipulated by the design computations) when the control rods were at their operating banked position. In anticipation of this, and rather than use still "grayer" poison strips in place of the 0.5-g-B¹⁰ strips, a supply of 0.4-g-B¹⁰ strips was available for revision of the initial core. By adding one half-length burnable-poison strip (containing ~0.2 g B¹⁰) to each of the central sixteen fuel assemblies, on the bottom half of the other side plate, it was possible to retain the shutdown margin of rod control and, at the same time, to reduce the reactivity controlled by the rods in the operating reactor. The final strip modification shown in Table 14 resulted in a core that appeared to be suitable for operation at 3 Mwt, and this loading (No. 57) was selected as the revised reference system for operation at design conditions, subject to verification of its adequacy by additional testing. Among the experiments performed later were calibrations of the control system in the cold reactor, "neutron flux" mappings at design temperature and, finally, tests of the stability of power production by the core under demand control. At the conclusion of these tests the core loading exemplified by loading No. 57 was considered to be the final reference core for operation at ~3 Mwt.

To summarize, the final reference 3-Mwt core (loading No. 57) was comprised of five cross control rods, one source assembly, and forty fuel assemblies containing a total mass of 14.01 kg U²³⁵. To one side plate of each fuel assembly was attached a full-length burnable-poison strip containing (nominally) 0.5 g B¹⁰. To the bottom half of the other side plate of each fuel assembly in the center 4 x 4 array was attached one half-length poison strip containing (nominally) 0.2 g B¹⁰. Figure 30 is a loading diagram for this core. (Later, the source assembly was moved again, this time to its operating position in assembly location No. 22. Dummy fuel assemblies were added in locations No. 12, 21, and 31.)

At 94°F, loading No. 57 was critical with the five control rods banked at an indicated position of ~12.6 in., or with rod No. 9 at 19.1 in. and the other rods at zero (fully inserted). At room temperature, loading No. 57 was subcritical with any one off-center cross rod at 30 in. and the other rods at zero. (In the cold fresh core, without poisoning by Sm¹⁴⁹ or Xe¹³⁵, it is doubtful that shutdown would be possible with two off-center control rods at 30 in.)

FIG. 30
FINAL REFERENCE 40-ASSEMBLY CORE
(LOADING NO. 57)

Integration of the control-system calibration curve (Fig. 31) yielded an estimate of 16.5 dollars ($\sim 11.5\%$) available excess reactivity in the cold fresh reactor.

Using the value $\beta = 0.0064$ in an infinite medium fueled with U^{235} ,⁽⁷⁾ it has been computed that the effective delayed neutron fraction β_{eff} is ~ 0.007 . Thus, the forty 0.5-g-B¹⁰ strips control $\sim 8.9\%$ reactivity, and the sixteen half-length strips control $\sim 3.6\%$ in the cold fresh reactor with the five cross control rods banked at 20 in. (cf. Table 14).

B. Reactivity Effect of Moving Control Rods in the Cold Reactor

The control system was calibrated, in terms of the motion of control rods from a critical banked position, by positive-period measurements. The requisite additional control was obtained by the incremental addition of boric acid to the reactor water.

FIG. 31
DIFFERENTIAL REACTIVITY EFFECT OF MOVING
CONTROL RODS FROM 5-ROD BANK
(LOADING NO. 57 AT ROOM TEMPERATURE)

The differential "worth" curves for rod No. 9, and for rods No. 1, 3, 5, and 7, are shown in Fig. 31. As is evidenced by Table 15, there is a substantial reduction in the reactivity worth of a poison strip adjacent to a rod channel when the control rod is inserted. Similarly, the presence of a neutron absorber in the immediate vicinity of the arms of the central cross control rod leads to a reduction in the net reactivity gain resulting from a differential withdrawal of that rod, and to a shift in the position of the peak of the differential curve. The off-center cross control rods were not affected so strongly, but a shift in the position of the peak differential reactivity effect may be discerned by comparison of Fig. 31 with Fig. 25. An alternative fit to the curve for rod No. 9 in the region of 15 in. to 20 in. indicated position is shown by a dashed line in Fig. 31. It is believed that the solid line is a better fit, but the alternative curve is admissible.

Integration of the curves in Fig. 31 from the generic indicated position Z , to 30 in. yields the curves shown in Fig. 32.

C. Reactivity Effect of Boric Acid in the Reactor Water

In loading No. 57, the boric acid was observed to be worth approximately 9.5 dollars per gram of H_3BO_3 per liter of reactor water [~ 2.5 dollars/(g H_3BO_3 /gal)]. In contrast with the excellent fit to the total worth of the boric acid in loading No. 19 by a function linear in the H_3BO_3 concentration, it was more difficult to assign an average differential worth to the boric acid in loading No. 57. Possibly there was a real variation in the worth of H_3BO_3 because of the effect of the half-length strips. Possibly the use of a more indirect water-sampling procedure, required during these experiments, introduced a spurious variation. The measured worths varied between 9.1 and 10.2 dollars per gram H_3BO_3 per liter of reactor water over the range of boric acid concentrations encountered, with the reactor water at room temperature.

D. Heating Experiments

The reactor water was heated by passing steam from the external boiler to the steam coils in the downcomer region. Enough boric acid had been dissolved in the reactor water in the cold system to attain a critical rod bank position of ~ 21 in. At higher temperatures it was necessary to withdraw rods farther to achieve criticality. Thus, even with boric acid in solution, if the rods had been left in place there would have been a loss in reactivity upon heating the water. (Indeed, the loss in reactivity upon heating a stationary system would have been quite large.) From time to time, clean water was added to reduce the boric acid concentration so as to maintain a critical rod bank position of ~ 21 in.

It was discovered later that the procedure of taking samples of the reactor water at elevated temperatures was inadequate. Therefore the variations in concentration of boric acid in the reactor water were computed from recorded data of dilutions plus reductions in water density corresponding to measured changes in the saturation temperature. The analysis of reactor water samples taken at room temperature provided the base point of the computation.

It is estimated that upon heating the reactor water from 90°F to 418°F , and with the control rods banked at 21 in., the net loss in available excess reactivity was $3.3 (\pm 0.5)$ dollars.

The differential reactivity effect of moving rods from the five-rod bank was measured at various temperatures and at various banked rod positions. The procedure followed was to heat the reactor water to a specified temperature, wait until the water temperatures measured above and below the reactor core were equalized, determine the critical rod bank position, and then "bump" the rods in the usual fashion for positive-period measurements. Reproducibility was checked by repeating the positive-period calibration. The positive reactivity effect of reactor cooling was evident, for usually the experiments at a "given" temperature required between 15 minutes and 30 minutes, during which time the reactor water temperature dropped 2°F to 5°F . Nevertheless, it was clear that the differential reactivity effect of rod motion increased substantially as the water temperature was raised. For example, data are available at a critical rod bank position of ~ 21.8 in. and at various temperatures:

$$\text{Rod No. 9: } \begin{cases} 0.43 \text{ dollar/in. at } 95^{\circ}\text{F} \\ 0.54 \text{ dollar/in. at } 225^{\circ}\text{F} \\ 0.65 \text{ dollar/in. at } 285^{\circ}\text{F} . \end{cases}$$

(Rod No. 9 was chosen to illustrate this temperature dependence because the perturbation effect of the poisoned water filling the gap left by the rod withdrawal should be small in view of the long (17 in.) rod follower.)

It is not implied that the overall rod worth increases by ~50% from room temperature to 300°F. The measured effectiveness of rod No. 9 at a water temperature of 397°F and at a banked rod position of 24.5 in. was 0.35 dollar/in., which may be compared with the value of 0.21 dollar/in. indicated by the curve shown in Fig. 30 for the average position of 24.9 in. However, it may simply be that the curve is not sufficiently accurate at that point to permit the obvious inference. The purpose of the comparisons made above is to point up the distinction made earlier (see Section VIII-D) between the different values one may assign to the so-called "temperature coefficient of reactivity." In this case, if it were possible to make the hot reactor critical at a rod bank position of 21 in. with an auxiliary control system whose reactivity worth would be unaffected by changes in temperature, the cooling down to room temperature would result in a reactivity gain of perhaps 8 dollars if the control rods were left at 21 in.

E. Neutron Flux Mapping of the Core at 412°F

Before the heating experiments discussed in Section XI-D were begun, bare gold wires were positioned vertically in the core for activation at design temperature and with control rods banked at a position close to that anticipated for the reactor operating at 3 Mwt and with equilibrium xenon. The sequence of events of the reactor heating resulted in a control rod bank position of 22.2 in. at a water temperature of 412°F, and this was considered to be a reasonable environment for the irradiation of the gold wires. Figure 33A shows the approximate locations of the wires and the numbers assigned to them. The axial distributions of gold activation are given in Figs. 33B to 33D. A curve through the datum points from wire No. 0B would be close to the curve shown for wire No. 4B. Similarly, the curve through data from wire No. 8 also fits the normalized activation plot of wire No. 9.

FIG. 33A
POSITIONS OF GOLD WIRES FOR "FLUX PLOT" IN LOADING NO. 57 ($\sim 420^\circ\text{F}$)

FIG. 33B
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 ($\sim 420^\circ\text{F}$)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

FIG. 33C
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 ($\sim 420^\circ\text{F}$)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

FIG. 33D
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX"
IN LOADING NO. 57 ($\sim 420^\circ\text{F}$)
(5-ROD BANK AT INDICATED POSITION OF 22.2 in.)

XII. FINAL CORE LOADING OF FIFTY-NINE FUEL ASSEMBLIES

At this time 35 temporary burnable-poison strips were available for use in zero-power experiments with a final core loading of 59 fuel assemblies. (The strips were unsuitable for use at power, and they were removed from the assemblies and returned to Argonne National Laboratory at the conclusion of these experiments.) Each strip was ~ 0.020 in. thick and contained $\sim 0.8\text{-g B}^{10}$.

The assemblies in locations No. 35, 36, 45, and 46 were moved to locations No. 25, 26, 47, and 37, respectively, in place of the assemblies occupying those positions in loading No. 57. Assemblies No. 24, 36, 21, and 23 were loaded into the four empty locations; to each one was taped $1\frac{1}{2}$ 0.8-g- B^{10} strips, the full-length strips forming a cross at the center of the cell, and the half-length strips attached to the bottom half of the other side plates. It is estimated that these substitutions resulted in a reactivity loss of ~ 1.3 dollars, and that the corresponding substitutions in the remainder of the central 4×4 array of fuel assemblies essentially compensated for the reactivity gained when nineteen assemblies were added to generate the final 59-assembly loading. By so compensating for the increase in core size, it was possible to control the 59-assembly core with the five cross control rods alone. It is extremely unlikely that this cold loading with five cross rods could have been controlled with any one cross rod at 30 in., though the actual test was not performed. Therefore it may be preferable to use all nine control rods.

In summary, the final 59-assembly core loading was comprised of five cross control rods, one source assembly, and the fifty nine fuel assemblies. To each assembly in the central 4×4 zone of the core was attached one full-length 0.8-g- B^{10} strip plus a half-length strip (containing $\sim 0.4\text{ g B}^{10}$) on the lower half of the other side plate. Sixteen fuel assemblies, each containing one full-length 0.5-g- B^{10} strip, and one half-length strip with $\sim 0.2\text{ g B}^{10}$, were positioned in assembly locations No. 32 to 62, 23 to 26, 37 to 67, and 73 to 76. Fuel assemblies without burnable-poison strips occupied locations No. 17, 71, and 87. The source assembly was in location No. 31. To each of the remaining 24 fuel assemblies was attached one full-length 0.5-g- B^{10} strip. This loading is shown in Fig. 34.

The program of experiments on this core loading was limited essentially to a "neutron flux" mapping of the core, at $\sim 365^\circ\text{F}$, accomplished with the aid of vertically positioned copper wires. During the irradiation of the wires, rods No. 1, 3, 5, and 7 were banked at 21.6 in. and rod No. 9 was positioned at 21.2 in.; boric acid was used for auxiliary control. The flux mapping was considered to be the major single experiment that should be carried out with this loading before taking the reference 3-Mwt loading to power, for it was more convenient to place the wires and to remove them before high-intensity irradiation rendered the assemblies accessible only

by remote handling. It was expected that any control system calibrations, etc., considered to be necessary for a better understanding of the larger core could be undertaken at a later date. The urgency of the scheduling for power tests with the design plant and for personnel training precluded a lengthy experimental program on the 59-assembly loading at this time.

FIG. 34
FINAL 59-ASSEMBLY CORE
(LOADING NO. 60)

In Fig. 35-A the locations and numbers of the copper wires are shown. The axial activation distributions are presented in Figs. 35-B to 35-P, with the exception of anomalous data from wires No. 44, 45, and 46 in assembly location No. 58. In each case, an attempt was made to fit the data with a reasonable curve. In some cases there were flat spots in the neighborhood of 7 in. to 11 in. from the bottom of the active core. This flux variation is recorded in the figures but, in some cases, it is difficult to account for the magnitude of the effect.

- • — FULL-LENGTH 0.8gB¹⁰-STRIP
- • • — HALF-LENGTH 0.8gB¹⁰-STRIP
- • — FULL-LENGTH 0.5gB¹⁰-STRIP
- • • — HALF-LENGTH 0.4gB¹⁰-STRIP
- • • • • NO POISON STRIP

FIG. 35A
POSITIONS OF COPPER WIRES FOR "FLUX PLOT" IN THE HOT 59-ASSEMBLY CORE
(LOADING NO. 60 AT ~385°F)

FIG. 35B
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 (~385°F)

FIG. 35C
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 (~385°F)

FIG. 35D
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 (~385°F)

FIG. 35E
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 (~385°F)

FIG. 35L
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 ($\sim 365^\circ\text{F}$)

FIG. 35M
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 ($\sim 365^\circ\text{F}$)

FIG. 35N
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 ($\sim 365^\circ\text{F}$)

FIG. 35P
AXIAL DISTRIBUTIONS OF "THERMAL NEUTRON FLUX" IN LOADING NO. 60 ($\sim 365^\circ\text{F}$)

Appendix A

REMARKS ON THE EVOLUTION OF THE REACTOR CORE DESIGN:
DESIGN MOTIVATIONS AND INFLUENCE
OF ZERO-POWER EXPERIMENTS1. The Fuel Assembly

In a significant sense, the core design of the ALPR owes much to the design of the early BORAX cores and to the operating experience with those reactors.⁽⁹⁾ In turn, the fuel assembly used in the BORAX-I loading was quite similar to the aluminum-clad-plate assembly of the MTR. An important innovation in ALPR arose from the decision to use, instead of 2S aluminum, an aluminum-nickel alloy (M-388) under development at Argonne National Laboratory.⁽¹⁰⁾ This alloy of aluminum with approximately one weight-per cent nickel, now designated Alcoa X 8001, exhibited much greater resistance to corrosion in boiling water at elevated temperatures than did 2S; in particular, the X8001 alloy appeared to be suitable for use in boiling water at 300 psig and $\sim 421^{\circ}\text{F}$ for extended periods of time. All structural aluminum in the core is alloyed with a small percentage of nickel, to take advantage of the increased corrosion resistance of the alloy.

Like the BORAX-I, II, and III assemblies, the ALPR fuel assembly is constituted of fuel plates containing a clad "meat" of an alloy of highly enriched uranium ($\sim 93 \text{ w/o } \text{U}^{235}$) with aluminum. A thicker clad (0.035 in., rather than 0.020 in.) is used in ALPR in the hope of lengthening core life. Consequently a much thicker fuel-plate core, or "meat" (0.050 in. vs. 0.020 in.) was required to keep down the uranium percentage, and therefore the fractional atom burnup, and to yield an overall metal-to-water volume ratio in the reactor core intermediate to the extremes encountered in the first three BORAX reactor cores. The cladding material is X 8001; the aluminum-nickel alloy used in the fuel "meat" contained $\sim 2 \text{ w/o}$ nickel. With the much sturdier fuel plate (0.120 in. thick), it was possible to avoid a center web structure, and, instead, the fuel plates were flanged and spot-welded to two side plates to form a durable element. The spacing of the fuel plates (0.310 in.) was chosen in accordance with the goal of an intermediate metal-to-water ratio in the reactor core; this coolant channel thickness is comparable to that of BORAX-II (0.264 in.). An assembly of nine fuel plates resulted, with a plan cross section $3\frac{7}{8} \text{ in.} \times 3\frac{7}{8} \text{ in.}$

To summarize, the fuel assembly structure of ALPR was designed so as not to deviate greatly from the tested structure of the BORAX-II and III assemblies, to retain desirable irradiation and reactor physics characteristics, and to present a thick corrosion-resistant clad for longer fuel-assembly life.

2. Size of the Core

A 3-Mwt reference core with forty fuel assemblies was chosen. With this loading, at 3 Mwt, an average power density of ~ 17.5 kwt per liter of coolant fluid in the core was obtained. This level appeared to be conservative in comparison with known stable power densities achieved in BORAX-III.⁽⁹⁾ However, for a variety of reasons, provision was made for a maximum core loading of fifty-nine fuel assemblies plus one source assembly. Of these motivations, the most important were: (a) operation with thicker fuel plates and larger coolant channels than had been used in BORAX, and the anticipation that an automatic demand-control system would be utilized to adjust reactor heat output to load needs, with the attendant uncertain effects on the stable power output of the reactor core; (b) a flexibility in core loading to permit reactivity adjustments shown to be desirable as a result of zero-power or power experiments; and (c) the desirability of having available a reactor core from which significantly higher power could be got, for possible use in another plant. The average fuel loading of each assembly ($\sim 350\text{g U}^{235}$) was determined by theoretical analysis for the forty-assembly 3-Mwt core, and for a nominal core life of three years at average power. In Fig. 2 is shown the forty-assembly reference core loading configuration - the array of assembly locations marked "R" ("reference"); the full core of fifty-nine fuel assemblies and one source assembly is the totality of locations "R" and "A" ("additional"). More details and some dimensions are provided in Fig. 5 (Core Plan, Section) and Fig. 6 (Core, Vertical Section).

3. Control Rods

The locations, the number, and the size of the cross control rods were determined by the combination of the size of the drive mechanisms, the limited effective (radial) area of the forty-assembly reference core, the possibility that a still larger core might be used with only the cross rods for control, and the reactivity hold-down requirements of the cold reactor without equilibrium xenon. After a review of alternative absorbers, cadmium was selected as the neutron absorber and an aluminum-nickel container was devised. A design of five cross control rods resulted.

Aluminum-clad cadmium rods had been used successfully in BORAX at similar operating temperatures. A control rod of this type was relatively inexpensive. There were no problems of gas formation. And finally, in view of the high thermal neutron-absorption cross section of the Cd^{113} , it was possible to keep the rod thermally black for at least two core cycles - even in the region of highest neutron current into the rod (assuming that otherwise the rod would be usable for that length of time).

Among other control materials considered were hafnium and boron-stainless steel. Hafnium was expensive and in short supply. At the time of fabrication of the control rods, the problems of fabrication of control rods

using stainless steel with 2 w/o boron had not been fully resolved. With the decision made to use an aluminum-nickel container for the absorbing material, the cadmium was a reasonable choice as a neutron absorber.

For additional control in a fifty-nine-assembly core, control rod channels were provided for a maximum of four tee rods at the periphery of the core. The tee rods were fabricated by the same technique used for the cross control rods. The locations of the nine control rods are shown in Fig. 2.

4. Burnable Poison

It had been intended to supplement the reactivity control of the system of absorbing rods by the incorporation of a burnable poison in the fuel plate "meat," in the form of boron essentially fully enriched in B^{10} . Theoretical analyses indicated that 16 grams of B^{10} would control ~11% in the cold fresh reactor, when uniformly mixed with the U^{235} in the core. Pursuant to a request by the Army Reactors Branch that the fuel assemblies be suppliable commercially, research and development contracts were let for a technique of fabricating fuel plates containing B^{10} in the "meat." These contracts did not lead to a method for producing fuel plates of the specified quality of an integral bond of cladding to the "meat." Finally, to expedite the procurement of fuel plates from a commercial fabricator, it was decided to resort to the use of thin strips containing an aggregate of powders of aluminum-nickel and enriched boron. As had been done in BORAX-III, where similar strips were used to assist rod control, these strips were to be tack-welded to one or both side plates of designated fuel assemblies. Two lots of burnable-poison strips were obtained, containing (nominally) 0.4 g B^{10} and 0.5-g- B^{10} per full-length strip, respectively. In the initial reference forty-fuel-assembly core, one 0.5-g B^{10} strip was attached to each assembly. In the second, final, reference 40-assembly core, half-length strips, each containing nominally 0.2 g B^{10} , were added to the other side plate of the central sixteen fuel assemblies in the lower half of the active core region. Later, in the course of the zero-power work, similar strips, each containing 0.8 g B^{10} , were fabricated, purely for zero-power reactor physics experiments with 59-assembly core loadings. (Subsequently, all of the 0.8-g- B^{10} strips were returned to Argonne National Laboratory.)

5. Interplay of Core Design and the Zero-power Reactor Physics Program

The zero-power program of reactor physics experimentation was formulated to: (a) determine the adequacy of the fuel loading in the cold 3-Mwt reference core, in terms of the reactivity available for taking the system to design conditions of power operation; (b) to assign a final loading of burnable-poison strips such that there would be a reactivity margin of approximately $5\frac{1}{2}$ dollars when the reactor was first taken to 3 Mwt and an equilibrium xenon distribution had been attained (of this, one dollar would be offset by the Sm^{149} that builds up slowly over the first few months of

effective power production); and (c) to ascertain the control effectiveness of the system of movable control rods, including the margin of shutdown in the cold reactor in the absence of xenon, and the possibility of control with one or more off-center rods stuck in the position of full rod withdrawal (30 in.).

In the initial reference 3-Mwt core, one burnable-poison strip containing (nominally) 0.5 g B^{10} was attached to each fuel assembly. With five cross control rods banked in the vicinity of the position anticipated for 3-Mwt operation, the poison strips were calibrated, and one half-length strip containing (nominally) 0.2 g B^{10} was added to the bottom half of the other side plate of each of the central sixteen assemblies. This loading of (forty) fuel assemblies and burnable-poison strips was designated the final reference core for operation at design power. The shutdown margin was estimated to be approximately 5 dollars in the cold fresh reactor, without xenon. It was shown that this cold system could be shut down with one off-center cross control rod fully withdrawn and a second off-center rod almost fully withdrawn. The system was critical with the center rod at 19.1 in. and the other cross rods fully inserted.

The average U^{235} content of the fuel assembly was determined for the 40-assembly reference core loading. If the reference system had included a 59-assembly core, doubtless a different uranium loading would have been chosen. The proving of a final 40-assembly core dominated the reactor physics programs, and the selection of a 59 fuel-assembly core was relegated to a position of lesser importance. The final 59-fuel-assembly core was comprised of a central 4 x 4 array of fuel assemblies each containing one full-length 0.8-g- B^{10} strip, plus one half-length strip, with ~ 0.4 g B^{10} , on the lower half of the other side plate; a ring of sixteen fuel assemblies, each with one full-length strip containing 0.5 g B^{10} and one half-length strip with 0.2 g B^{10} ; a second ring of twenty-four assemblies, each with one full-length 0.5-g- B^{10} strip; and three fuel assemblies without poison strips. This loading could be controlled by the five cross control rods alone, although it is unlikely that the cold reactor would be subcritical with one of the five rods at 30 in.

Appendix B

COMPOSITION OF THE FUEL "MEAT" IN THE
ALPR FUEL PLATES

In Table B-1 are listed the U^{235} contents of the individual, numbered fuel plates in the 66 fuel assemblies fabricated for possible use in the zero-power reactor physics experiments with cores ranging in size to a maximum of 59 fuel assemblies. The U^{235} contents tabulated were obtained by gamma-emission counting of each core melt, with the exception of plates No. E4, 94, and 499, whose U^{235} contents were estimated from other data.^(11,12) The w/o uranium listed was calculated from the U^{235} content and the isotopic percentage in the core.⁽¹³⁾

The U^{235} content of some of the fuel plates lies somewhat outside the originally specified limits of $39(\pm 1)$ g U^{235} per plate. However, these plates were considered to be acceptable provided that the fuel assemblies contained $351(\pm 9)$ g U^{235} . All of the assemblies meet this criterion. The U^{235} enrichments ranged from 93.20% to 93.30%.

The total amount of U^{235} in the final reference forty-fuel-assembly core loading (loading No. 57) was 14.01 kg.

Table B-2 gives the weight percentages of Ni, Fe, and some other elements in each rolled plate from which cores were punched.⁽¹⁴⁾ It is presented here to indicate the isotopic contents of elements other than uranium in the alloy, as determined by emission spectrochemical analysis. The average Ni content is 1.60 w/o, and the average Fe content is 0.40 w/o.

Table B-1

URANIUM CONTENT OF FUEL PLATES IN ALPR FUEL ASSEMBLIES

<u>U²³⁵</u>			<u>U²³⁵</u>			<u>U²³⁵</u>		
<u>Plate</u> <u>No.</u>	<u>Content</u> <u>(g)</u>	<u>w/o U in</u> <u>the Meat</u>	<u>Plate</u> <u>No.</u>	<u>Content</u> <u>(g)</u>	<u>w/o U in</u> <u>the Meat</u>	<u>Plate</u> <u>No.</u>	<u>Content</u> <u>(g)</u>	<u>w/o U in</u> <u>the Meat</u>
<u>Assembly No. 1</u>			<u>Assembly No. 5</u>			<u>Assembly No. 9</u>		
E 9	39.3	17.58	E 8	38.6	17.31	E247	38.9	17.35
82	39.1	17.41	88	39.9	17.74	249	38.9	17.32
11	39.2	17.48	253	39.8	17.81	246	39.2	17.43
80	39.1	17.47	21	39.6	17.65	259	39.0	17.44
69	39.4	17.56	280	39.0	17.62	276	39.1	17.34
52	38.8	17.29	31	39.5	17.57	248	39.5	17.65
25	39.0	17.37	16	39.3	17.51	138	39.1	17.41
48	38.8	17.34	7	38.4	17.27	142	38.4	17.21
24	38.8	17.35	57	39.2	17.51	159	38.9	17.33
	<u>351.5</u>			<u>353.3</u>			<u>351.0</u>	
<u>Assembly No. 2</u>			<u>Assembly No. 6</u>			<u>Assembly No. 10</u>		
E 4	39.6*	17.69*	E166	39.0	17.49	E255	38.8	17.35
13	39.0	17.51	155	38.5	17.30	278	38.7	17.22
79	39.5	17.71	83	39.4	17.53	279	39.1	17.37
46	39.1	17.43	165	39.1	17.42	272	39.0	17.38
6	37.8	17.09	32	39.0	17.44	261	39.9	17.84
81	40.4	17.95	160	39.1	17.37	257	39.2	17.57
59	39.5	17.53	28	39.5	17.63	244	39.8	17.73
26	39.3	17.50	163	39.5	17.70	277	39.4	17.54
72	38.8	17.33	22	38.8	17.29	281	39.0	17.63
	<u>353.0</u>			<u>351.9</u>			<u>352.9</u>	
<u>Assembly No. 3</u>			<u>Assembly No. 7</u>			<u>Assembly No. 11</u>		
E68	39.1	17.56	E164	39.5	17.57	E227	38.6	17.37
47	39.2	17.52	140	38.4	17.12	193	39.2	17.53
56	39.1	17.41	162	39.9	17.76	226	38.5	17.46
66	38.4	17.30	158	39.5	17.57	254	39.8	17.87
30	40.2	17.82	141	38.7	17.22	221	39.5	17.79
74	38.2	17.19	273	39.0	17.37	275	39.3	17.53
18	38.0	17.13	139	38.7	17.22	251	39.9	17.81
34	38.2	17.03	157	39.2	17.53	197	39.2	17.54
45	39.2	17.52	137	38.1	17.08	195	39.9	17.80
	<u>349.6</u>			<u>351.0</u>			<u>353.9</u>	
<u>Assembly No. 4</u>			<u>Assembly No. 8</u>			<u>Assembly No. 12</u>		
E78	38.9	17.42	E250	38.5	17.15	E192	39.5	17.76
33	39.7	17.69	252	39.3	17.55	190	39.4	17.50
75	38.9	17.40	256	38.9	17.42	176	39.1	17.56
85	39.0	17.39	260	39.4	17.63	183	39.4	17.61
35	38.5	17.17	264	39.3	17.58	182	39.3	17.47
62	38.8	17.24	265	39.2	17.53	184	38.5	17.23
44	38.8	17.28	269	39.3	17.64	186	38.9	17.37
73	39.1	17.49	270	38.8	17.17	191	38.6	17.43
39	38.6	17.15	274	39.0	17.43	189	38.6	17.45
	<u>350.3</u>			<u>351.7</u>			<u>351.5</u>	

*Contents were computed by using data other than those obtained by gamma-emission counting.

Table B-1 (Cont'd.)

U ²³⁵			U ²³⁵			U ²³⁵		
Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat
<u>Assembly No. 13</u>			<u>Assembly No. 17</u>			<u>Assembly No. 21</u>		
E167	39.7	17.71	E 93	38.3	17.27	E204	39.3	17.59
245	38.8	17.40	90	38.4	17.26	203	39.4	17.67
123	39.1	17.58	232	39.3	17.67	207	39.0	17.44
121	39.4	17.90	105	38.6	17.35	210	39.6	17.75
143	38.7	17.32	127	38.7	17.27	212	38.6	17.34
146	38.4	17.23	233	38.6	17.38	216	39.3	17.56
145	38.7	17.29	125	38.2	17.19	239	39.2	17.54
153	38.6	17.24	169	38.8	17.28	230	39.2	17.55
147	39.0	17.36	128	39.2	17.51	284	39.9	17.88
	<u>350.4</u>			<u>348.1</u>			<u>353.5</u>	
<u>Assembly No. 14</u>			<u>Assembly No. 18</u>			<u>Assembly No. 22</u>		
E 99	38.6	17.45	E211	39.4	17.62	E339	38.0	17.08
96	39.2	17.67	218	39.8	17.90	338	39.1	17.42
110	39.7	17.69	173	38.5	17.26	342	38.3	17.12
119	38.7	17.40	234	38.9	17.59	340	38.0	17.12
130	39.6	17.62	100	38.4	17.34	344	38.8	17.36
134	38.9	17.42	89	38.5	17.34	201	39.5	17.82
171	38.6	17.36	104	39.6	17.80	346	38.1	17.18
131	38.6	17.26	91	39.9	17.97	314	39.2	17.41
129	39.7	17.67	103	38.3	17.33	320	38.5	17.27
	<u>351.6</u>			<u>351.3</u>			<u>347.5</u>	
<u>Assembly No. 15</u>			<u>Assembly No. 19</u>			<u>Assembly No. 23</u>		
E236	38.4	17.23	E120	38.1	17.33	E323	39.0	17.34
229	38.5	17.25	126	39.6	17.62	358	39.4	17.80
228	38.9	17.48	341	38.3	17.17	353	38.7	17.26
240	39.2	17.57	324	38.8	17.31	354	39.4	17.52
243	39.2	17.46	109	40.0	17.85	356	39.1	17.53
94	39.5*	17.60*	231	38.2	17.25	309	38.9	17.32
92	40.4	18.14	205	39.6	17.62	325	39.0	17.41
238	39.3	17.66	235	38.3	17.22	337	38.9	17.36
95	39.4	17.71	115	39.2	17.63	349	38.4	17.33
	<u>352.8</u>			<u>350.1</u>			<u>350.8</u>	
<u>Assembly No. 16</u>			<u>Assembly No. 20</u>			<u>Assembly No. 24</u>		
E222	39.4	17.69	E297	38.5	17.32	E286	38.6	17.20
174	38.0	17.29	290	38.7	17.34	217	39.0	17.41
177	38.6	17.29	287	39.0	17.39	296	39.5	17.59
179	38.9	17.35	298	39.1	17.60	288	39.4	17.53
188	38.3	17.28	299	38.6	17.38	293	39.9	17.69
237	38.5	17.27	302L	38.1	17.15	301	38.7	17.56
223	39.0	17.52	307	39.8	17.70	308	39.5	17.41
187	38.6	17.26	303L	39.0	17.49	306	38.3	17.17
224	39.2	17.58	304L	39.0	17.45	350	39.8	17.82
	<u>348.5</u>			<u>349.8</u>			<u>352.7</u>	

* Contents were computed by using data other than those obtained by gamma-emission counting.

Table B-1 (Cont'd.)

U ²³⁵			U ²³⁵			U ²³⁵		
Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat
<u>Assembly No. 25</u>			<u>Assembly No. 29</u>			<u>Assembly No. 33</u>		
E333	39.5	17.65	E363	39.6	17.64	E248	38.0	17.08
327	38.5	17.20	374	38.7	17.25	372	39.5	17.53
321	38.7	17.33	369	39.0	17.35	317	39.0	17.46
347	38.5	17.17	461	40.4	17.95	420	38.7	17.29
319	39.2	17.50	436	39.6	17.73	443	40.0	17.74
318	39.6	17.69	385	39.4	17.52	379	39.9	17.67
348	38.0	17.06	430	38.6	17.18	450	38.4	17.11
213	39.0	17.55	367	39.9	17.71	424	39.6	17.64
285	38.9	17.59	316	39.0	17.41	322	39.0	17.38
	<u>349.9</u>			<u>354.2</u>			<u>352.1</u>	
<u>Assembly No. 26</u>			<u>Assembly No. 30</u>			<u>Assembly No. 34</u>		
E395	39.6	17.70	E389	39.8	17.69	E409	39.6	17.70
370	39.0	17.38	451	39.7	17.66	391	39.0	17.22
361	39.0	17.35	396	39.4	17.57	393	38.4	17.22
357	39.5	17.66	411	38.1	17.14	444	40.0	17.75
362	39.6	17.66	388	39.3	17.51	425	39.7	17.71
360	38.9	17.35	400	38.9	17.41	422	39.8	17.60
364	38.6	17.18	413	38.5	17.19	447	38.3	17.05
332	38.3	17.14	408	39.4	17.56	449	38.7	17.28
355	39.2	17.50	404	39.2	17.48	406	39.3	17.44
	<u>351.7</u>			<u>352.3</u>			<u>352.8</u>	
<u>Assembly No. 27</u>			<u>Assembly No. 31</u>			<u>Assembly No. 35</u>		
E328	38.5	17.16	E433	39.2	17.55	E 15	40.2	17.89
305	38.7	17.37	435	39.3	17.61	458	40.3	17.94
295	39.5	17.59	418	39.9	17.73	472	37.7	16.91
300	39.1	17.53	397	39.8	17.71	439	40.1	17.78
291	38.4	17.33	427	39.0	17.33	442	40.1	17.94
194	39.0	17.38	438	39.1	17.50	441	40.2	17.75
313	38.7	17.38	437	39.1	17.49	445	40.1	17.90
410	39.3	17.51	329	38.9	17.37	27	37.3	16.83
412	38.1	17.06	335	39.2	17.37	464	39.9	17.84
	<u>349.3</u>			<u>353.5</u>			<u>355.9</u>	
<u>Assembly No. 28</u>			<u>Assembly No. 32</u>			<u>Assembly No. 36</u>		
E331	38.1	17.07	E432	38.1	17.11	E487	39.7	18.02
380	39.1	17.44	452	39.3	17.54	478	38.3	17.48
365	38.5	17.17	423	39.1	17.35	505	38.4	17.38
417	39.1	17.41	334	38.6	17.21	481	38.3	17.33
384	39.3	17.46	453	38.6	17.43	499	39.1*	17.60*
386	39.6	17.56	454	39.4	17.69	484	38.8	17.59
336	39.6	17.55	401	39.8	17.75	501	38.7	17.42
382	39.6	17.55	459	40.2	17.94	502	38.2	17.36
381	39.1	17.46	446	40.2	17.96	493	38.4	17.27
	<u>352.0</u>			<u>353.3</u>			<u>347.9</u>	

* Contents were computed by using data other than those obtained by gamma-emission counting.

Table B-1 (Cont'd.)

U ²³⁵			U ²³⁵			U ²³⁵		
Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat
<u>Assembly No. 37</u>			<u>Assembly No. 41</u>			<u>Assembly No. 45</u>		
E508	38.3	17.51	E540	39.1	17.50	E571	39.2	17.54
521	39.0	17.49	474	37.9	16.97	608	38.5	17.51
510	37.9	17.31	470	38.0	17.05	586	39.5	17.64
512	38.0	17.46	473	37.8	16.94	568	38.9	17.54
496	39.3	17.78	465	37.5	16.93	558	38.2	17.55
483	38.2	17.39	463	40.0	17.87	534	39.3	17.25
523	38.5	17.24	455	39.6	17.70	576	38.5	17.47
490	38.9	17.61	466	40.0	17.73	592	39.0	17.37
482	38.1	17.32	519	38.5	17.23	581	38.5	17.31
	<u>346.2</u>			<u>348.4</u>			<u>349.6</u>	
<u>Assembly No. 38</u>			<u>Assembly No. 42</u>			<u>Assembly No. 46</u>		
E514	38.2	17.39	E561	39.2	17.54	E589	38.4	17.14
541	39.0	17.48	560	39.6	17.69	588	38.6	17.27
511	38.4	17.43	552	38.6	17.41	609	39.6	17.66
520	38.3	17.15	572	39.2	17.50	597	39.8	17.89
542	39.0	17.45	517	38.9	17.55	593	39.3	17.57
515	38.2	17.25	530	39.8	17.88	594	39.3	17.53
522	39.0	17.56	567	39.8	17.71	598	38.0	17.05
547	39.5	17.77	531	40.0	17.92	603	39.5	17.71
479	38.5	17.35	535	39.5	17.67	610	39.2	17.53
	<u>348.1</u>			<u>354.6</u>			<u>351.7</u>	
<u>Assembly No. 39</u>			<u>Assembly No. 43</u>			<u>Assembly No. 47</u>		
E550	39.7	17.60	E569	39.1	17.63	E612	40.0	17.92
516	38.1	17.28	533	39.4	17.65	602	39.5	17.70
551	39.8	17.76	575	38.7	17.46	601	38.9	17.43
525	39.2	17.57	577	38.3	17.16	595	39.5	17.59
553	39.4	17.61	563	39.7	17.71	600	39.1	17.43
546	39.2	17.59	574	39.5	17.73	599	38.4	17.23
555	39.5	17.65	532	39.2	17.58	604	39.8	17.85
527	39.2	17.76	566	39.9	17.77	591	38.6	17.29
528	39.9	17.83	518	40.1	17.95	606	39.0	17.44
	<u>354.0</u>			<u>353.9</u>			<u>352.8</u>	
<u>Assembly No. 40</u>			<u>Assembly No. 44</u>			<u>Assembly No. 48</u>		
E565	39.1	17.51	E579	38.4	17.18	E573	39.3	17.45
524	39.0	17.54	562	39.7	17.70	587	40.0	17.90
549	39.2	17.52	584	39.2	17.59	607	39.3	17.55
536	39.4	17.66	529	39.4	17.74	605	39.4	17.58
544	38.9	17.40	570	39.7	17.76	613	39.7	17.86
537	39.8	17.86	582	38.2	17.11	559	38.4	17.53
539	39.1	17.48	580	39.6	17.72	611	39.3	17.60
543	39.2	17.50	578	38.3	17.17	585	38.7	17.47
564	38.9	17.49	583	38.1	17.13	590	38.6	17.27
	<u>352.6</u>			<u>350.6</u>			<u>352.7</u>	

Table B-1 (Cont'd.)

U ²³⁵			U ²³⁵			U ²³⁵		
Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat	Plate No.	Content (g)	w/o U in the Meat
<u>Assembly No. 49</u>			<u>Assembly No. 53</u>			<u>Assembly No. 57</u>		
E616	38.5	17.45	E794	38.6	17.34	E766	38.6	17.34
670	39.2	17.57	762	38.1	17.23	814	39.1	17.57
675	38.0	17.11	683	38.0	17.05	772	37.8	17.19
650	39.2	17.51	665	38.1	17.06	660	38.9	17.41
637	38.1	17.24	711	39.3	17.57	652	39.5	17.70
614	38.3	17.40	682	38.2	17.26	787	38.1	17.16
671	39.6	17.69	666	38.0	17.05	703	38.3	17.29
668	38.8	17.64	715	38.5	17.34	708	39.6	18.85
651	39.2	17.65	724	39.0	17.48	692	38.7	17.40
	<u>348.9</u>			<u>345.8</u>			<u>348.6</u>	
<u>Assembly No. 50</u>			<u>Assembly No. 54</u>			<u>Assembly No. 58</u>		
E618	38.0	17.16	E771	37.9	17.01	E693	38.0	17.08
625	38.4	17.26	757	38.1	17.10	705	39.0	17.46
631	38.2	17.36	662	37.9	17.03	783	38.3	17.14
628	38.0	17.50	758	38.4	17.12	690	39.3	17.64
676	38.4	17.24	764	38.2	17.16	742	39.2	17.59
672	38.0	17.32	706	38.8	17.44	743	39.4	17.67
619	38.5	17.40	700	38.6	17.45	816	38.6	17.50
677	38.4	17.34	658	38.8	17.37	615	38.9	17.62
674	38.2	17.17	793	38.3	17.22	720	39.0	17.51
	<u>344.1</u>			<u>345.0</u>			<u>349.7</u>	
<u>Assembly No. 51</u>			<u>Assembly No. 55</u>			<u>Assembly No. 59</u>		
E632	38.2	17.41	E734	39.5	17.69	E777	38.3	17.22
622	38.2	17.24	730	39.9	17.88	695	38.3	17.17
620	38.3	17.49	745	38.3	17.20	811	39.0	17.14
653	39.6	17.67	688	37.9	17.08	815	39.8	17.43
635	38.8	17.48	717	38.4	17.28	788	38.2	17.86
617	37.9	17.28	791	38.6	17.22	790	38.5	17.20
629	38.2	17.30	680	38.1	17.10	813	38.9	17.48
633	38.3	17.35	799	38.3	17.19	809	38.1	17.40
636	38.5	17.46	770	38.0	17.05	749	39.4	17.71
	<u>346.0</u>			<u>347.0</u>			<u>348.5</u>	
<u>Assembly No. 52</u>			<u>Assembly No. 56</u>			<u>Assembly No. 60</u>		
E673	38.1	17.22	E723	39.4	17.62	E781	38.4	17.22
634	38.2	17.30	741	39.4	17.60	761	38.2	17.26
669	38.8	17.75	722	39.9	17.88	801	39.1	17.43
627	38.2	17.29	808	38.7	17.39	786	38.1	17.14
624	38.7	17.56	789	38.5	17.27	721	38.7	17.47
718	39.1	17.59	803	39.2	17.54	754	38.3	17.21
663	38.6	17.35	806	39.0	17.47	807	38.6	17.38
626	38.5	17.43	804	38.6	17.40	810	38.7	17.34
689	38.4	17.29	812	39.1	17.57	785	38.2	17.13
	<u>346.6</u>			<u>351.8</u>			<u>346.3</u>	

Table B-1 (Cont'd.)

Plate No.	U ²³⁵		Plate No.	U ²³⁵		Plate No.	U ²³⁵	
	Content (g)	w/o U in the Meat		Content (g)	w/o U in the Meat		Content (g)	w/o U in the Meat
<u>Assembly No. 61</u>			<u>Assembly No. 63</u>			<u>Assembly No. 65</u>		
E725	39.9	17.83	E691	38.5	17.35	E750	38.4	17.37
696	38.6	17.25	697	38.2	17.22	698	39.0	17.53
755	38.2	17.56	716	38.7	17.41	768	38.3	17.26
775	37.8	17.58	779	38.3	17.02	782	38.5	17.25
756	38.8	17.34	737	39.1	17.51	729	39.9	17.85
694	39.2	17.07	738	39.3	17.72	740	39.3	17.61
713	38.7	17.03	773	37.9	17.01	667	38.0	17.30
759	38.5	17.93	796	39.0	17.52	778	38.7	17.23
701	39.6	17.26	739	38.9	17.46	763	38.2	17.12
	<u>349.3</u>			<u>347.9</u>			<u>348.3</u>	
<u>Assembly No. 62</u>			<u>Assembly No. 64</u>			<u>Assembly No. 66</u>		
E704	39.2	17.62	E767	38.2	17.16	E731	38.9	17.51
765	38.6	17.30	727	39.6	17.75	687	38.0	17.12
699	38.9	17.51	747	38.4	17.30	792	38.8	17.31
728	39.5	17.67	744	39.8	17.85	760	38.1	17.12
709	40.1	17.91	710	40.0	17.82	795	38.3	17.35
702	37.9	17.37	769	39.4	17.69	797	38.9	17.45
661	38.7	17.34	748	38.0	17.18	678	38.2	17.28
732	38.2	17.45	735	39.2	17.54	681	38.0	17.11
774	37.8	16.98	753	38.8	17.40	800	38.9	17.48
	<u>348.9</u>			<u>351.4</u>			<u>346.1</u>	

Table B-2
ISOTOPIC COMPOSITION OF ROLLED PLATES FOR ALPR FUEL-PLATE CORES
(Uranium Percentages Are Not Listed)

Melt No.	Alloy Content, w/o				Impurities, w/o ^(b,c)					
	Ni ^(a)		Fe ^(a)		Cr	Cu	K	Mg	Mn	Pb
	Top	Bottom	Top	Bottom						
A-9	1.58	1.56	0.37	0.42	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-10	1.73	1.82	0.37	0.42	< 0.05	0.3	< 0.1	< 0.01	0.01	0.04
A-11	1.83	1.58	0.40	0.39	< 0.05	0.3	< 0.1	< 0.01	0.01	0.01
A-12	1.64	1.77	0.40	0.39	< 0.05	0.4	< 0.1	< 0.01	0.01	< 0.01
A-13	1.75	1.82	0.38	0.46	< 0.05	0.4	< 0.1	< 0.01	0.01	0.04
A-14	1.87	2.04	0.38	0.44	< 0.05	0.3	< 0.1	< 0.01	0.01	0.04
A-15	1.72	1.80	0.41	0.42	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-16	1.63	1.70	0.43	0.43	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-17	1.62	1.68	0.38	0.38	< 0.05	0.3	< 0.1	< 0.01	0.01	0.03
A-18	1.68	1.57	0.39	0.37	< 0.05	0.3	< 0.1	< 0.01	0.01	0.02
A-19	1.53	1.98	0.34	0.42	< 0.05	0.4	< 0.1	< 0.01	0.01	0.01
A-20	1.75	1.85	0.41	0.43	< 0.05	0.4	< 0.1	< 0.01	0.01	< 0.01
A-21	1.72	1.80	0.38	0.42	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-22	1.58	1.50	0.39	0.39	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-23	1.73	1.83	0.39	0.43	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-24	1.44	1.69	0.37	0.40	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-25	(d)	1.80	(d)	0.40	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-26	1.74	1.82	0.36	0.44	< 0.05	0.3	< 0.1	< 0.01	0.01	0.03
A-27	1.80	1.60	0.43	0.41	< 0.05	0.3	< 0.1	< 0.01	0.01	0.05
A-28	1.65	1.87	0.46	0.48	< 0.05	0.3	< 0.1	< 0.01	0.01	0.03
A-29	1.75	1.80	0.40	0.43	< 0.05	0.3	< 0.1	< 0.01	0.01	< 0.01
A-30	1.45	1.58	0.36	0.39	< 0.05	0.4	< 0.1	< 0.01	0.01	0.02
A-31	1.73	1.95	0.45	0.51	< 0.05	0.08	< 0.1	< 0.01	0.01	0.02
A-32	1.78	2.08	0.51	0.54	< 0.05	0.02	< 0.1	< 0.01	0.01	< 0.01
A-33	1.82	1.88	0.52	0.53	< 0.05	0.01	< 0.1	0.01	0.03	< 0.01
A-34	(d)	1.80	(d)	0.49	< 0.05	0.03	< 0.1	< 0.01	0.03	< 0.01
A-35	(d)	1.58	(d)	0.40	< 0.05	0.01	< 0.1	0.01	0.03	< 0.01
A-36	(d)	1.63	(d)	0.44	< 0.05	0.01	< 0.1	< 0.01	0.03	< 0.01
A-37	1.60	1.54	0.41	0.42	< 0.05	0.01	< 0.1	< 0.01	0.03	< 0.01
A-38	1.58	1.65	0.39	0.44	< 0.05	0.01	< 0.1	< 0.01	0.03	< 0.01
A-39	1.60	1.62	0.43	0.45	< 0.05	0.01	< 0.1	< 0.01	0.03	< 0.01
A-40	1.62	1.65	0.40	0.42	< 0.05	0.01	< 0.1	< 0.01	0.01	0.08
A-41	1.63	1.53	0.38	0.39	< 0.05	0.03	< 0.1	0.03	0.01	0.05
A-42	1.62	1.63	0.39	0.40	< 0.05	0.01	< 0.1	0.01	0.01	0.01
A-43	Remelted									
A-44	1.73	2.00	0.15	0.18	0.1	0.2	0.3	0.3	0.2	0.03
A-45	1.55	1.58	0.37	0.38	0.5	0.2	< 0.1	0.2	0.08	0.02
A-46	2.00	1.75	0.46	0.43	0.02	0.01	< 0.1	< 0.01	0.01	< 0.01
A-47	1.68	1.82	0.33	0.34	0.1	0.4	< 0.1	< 0.01	0.03	< 0.01
A-48	1.41	1.69	0.26	0.32	0.05	0.2	< 0.1	< 0.01	0.03	< 0.01
A-49	1.43	1.00	0.33	0.31	0.2	0.2	0.1	0.01	0.03	< 0.01
A-50	1.30	1.22	0.30	0.32	0.08	0.2	< 0.1	< 0.01	0.02	< 0.01
A-51	1.30	1.34	0.31	0.33	< 0.05	0.2	< 0.1	< 0.01	0.02	< 0.01
A-52	1.73	1.68	0.37	0.42	< 0.05	0.2	< 0.1	0.01	0.02	< 0.01
A-53	1.48	1.57	0.38	0.43	< 0.05	0.08	< 0.1	< 0.01	0.02	< 0.01
A-54	1.43	1.68	0.44	0.45	< 0.05	0.02	< 0.1	0.01	0.02	< 0.01
A-55	1.63	1.42	0.44	0.37	0.05	0.08	< 0.1	< 0.01	0.02	< 0.01
A-56	1.34	1.37	0.32	0.36	< 0.05	0.08	< 0.1	< 0.01	0.02	< 0.01
A-57	1.30	1.25	0.34	0.33	< 0.05	0.08	< 0.1	< 0.01	0.02	< 0.01
A-58	1.30	1.13	0.33	0.37	< 0.05	0.3	< 0.01	0.02	0.02	< 0.01
A-59	1.25	1.32	0.33	0.31	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-60	1.28	1.16	0.32	0.32	0.3	0.3	< 0.01	< 0.01	0.02	< 0.01
A-61	1.45	1.20	0.31	0.31	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-62	1.22	1.22	0.31	0.34	< 0.05	0.3	< 0.01	< 0.01	0.02	0.4
A-63	1.23	1.16	0.30	0.35	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-64	1.40	1.46	0.40	0.45	0.1	0.3	< 0.01	0.02	0.02	< 0.01
A-65	1.82	1.65	0.43	0.41	0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-66	1.63	1.63	0.52	0.42	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-67	1.55	1.81	0.35	0.54	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01
A-68	1.63	1.32	0.41	0.41	< 0.05	0.3	< 0.01	< 0.01	0.02	< 0.01

^(a)Samples were taken from the rolled plates, near the top and near the bottom of the plate, for Ni and Fe analysis. The accuracy of the weight percentage of these two elements is estimated to be 3% of the percentage listed.

^(b)Estimated accuracy is the order of magnitude (factor of ten)

^(c)For all melts.

Ag less than 0.01% with exception of A-26 which is 0.03%

As less than 0.5°.

B less than 0.001%.

Ba, Be, Bi, Li, Na, Sb Sn contents are <0.01%.

Ca, Ti, Zn, P less than 0.1%.

Co less than 0.01% with exception of A-51 which is 0.05°.

Si less than 0.2% for melts A-9 through A-57, less than 0.02% for melts A-58 through A-68.

Zr less than 0.1% with exception of A-45 which is <0.2%.

^(d)High U background interfered with analysis of Ni and Fe.

Appendix C

BURNABLE-POISON STRIPS: B^{10} CONTENTS AND
LOCATIONS IN THE FINAL CORE LOADINGS

Three groups of burnable-poison strips were fabricated for experiments on the various core loadings. One group consisted of forty strips, each 0.026 in. thick and containing nominally 0.5 g B^{10} . A second set of forty strips was fabricated by rolling to a thickness of 0.021 in.; each of these strips contained approximately 0.4 g B^{10} . A third group of 35 poison strips was prepared; each strip was ~ 0.020 in. thick and contained nominally 0.8 g B^{10} (~ 0.93 w/o B^{10}). The 0.8-g- B^{10} strips were blistered, and were not suitable for use in an operating reactor. Therefore, after the heating experiments with the final 59-assembly loading were completed (cf. Section XII), these strips were returned to Argonne National Laboratory.

The weights of the individual numbered strips are listed in Tables C-1 to C-3, and, directly or indirectly, the B^{10} content of each strip is supplied.⁽¹⁵⁾ The assignments of the strips in the two final core loadings are tabulated in Tables C-4 and C-5.

Table C-1
DATA FOR THE FORTY 0.5-g- B^{10} STRIPS

Strip No.	Weight (g)	Weight B^{10} (g)	Strip No.	Weight (g)	Weight B^{10} (g)
2	113.831	0.48	23	119.314	0.50
4	112.630	0.48	24	118.148	0.50
5	114.877	0.49	25	114.176	0.48
6	112.036	0.47	26	116.365	0.49
7	111.433	0.47	27	117.769	0.50
8	111.530	0.47	28	116.046	0.50
9	111.980	0.47	29	111.605	0.47
10	112.112	0.48	30	118.603	0.50
11	112.551	0.48	31	116.524	0.49
12	109.242	0.46	32	117.739	0.50
13	112.384	0.48	33	112.468	0.48
14	112.926	0.48	34	113.960	0.48
15	115.766	0.49	35	116.887	0.50
16	113.606	0.48	36	119.751	0.51
17	110.762	0.47	37	111.520	0.47
18	117.580	0.50	38	120.775	0.51
19	110.180	0.47	39	117.893	0.50
20	116.769	0.49	40	116.031	0.49
21	117.157	0.50	41	112.602	0.48
22	117.726	0.50	42	114.834	0.49

Table C-2

DATA FOR THE FORTY 0.4-g-B¹⁰ STRIPS

Strip No.	Weight (g)	B ¹⁰ Content (g)	Strip No.	Weight (g)	B ¹⁰ Content (g)
1	93.875	0.40	61	95.783	0.40
3	93.608	0.40	62	93.372	0.40
43	92.985	0.39	63	88.912	0.38
44	91.324	0.39	64	95.789	0.40
45	94.324	0.40	65	93.335	0.40
46	96.290	0.41	66	95.379	0.40
47	93.952	0.40	67	94.964	0.40
48	95.948	0.41	68	94.092	0.40
49	94.778	0.40	69	96.137	0.41
50	93.891	0.40	70	92.661	0.39
51	94.298	0.40	71	94.194	0.40
52	94.509	0.40	72	95.498	0.40
53	97.447	0.41	73	96.043	0.41
54	96.808	0.41	74	94.519	0.40
55	94.896	0.40	75	91.015	0.38
56	89.479	0.38	76	97.309	0.41
57	95.325	0.40	77	88.906	0.38
58	91.482	0.39	78	89.512	0.38
59	96.691	0.41	79	92.495	0.39
60	95.106	0.40	80	93.327	0.40

Table C-3

DATA FOR THE THIRTY-FIVE TEMPORARY 0.8-g-B¹⁰ STRIPS^a

Strip No.	Weight (g)	Strip No.	Weight (g)	Strip No.	Weight (g)
X-1	89.17	X-13	89.50	X-25	90.17
X-2	88.90	X-14	89.90	X-26	90.17
X-3	90.05	X-15	90.00	X-27	90.50
X-4	88.55	X-16	89.60	X-28	88.72
X-5	91.65	X-17	89.50	X-29	88.92
X-6	89.00	X-18	91.5	X-30	89.20
X-7	88.85	X-19	89.42	X-31	90.40
X-8	89.40	X-20	89.80	X-32	90.96
X-9	90.40	X-21	90.20	X-33	89.40
X-10	90.30	X-22	90.45	X-34	89.80
X-11	89.90	X-23	90.80	X-35	90.05
X-12	89.82	X-24	90.05		

^aThe nominal content of each strip is 0.93 w/o B¹⁰.

Table C-4

ASSIGNMENTS OF BURNABLE-POISON STRIPS
IN LOADING NO. 57

Assembly Number	Full-length 0.5-g-B ¹⁰ Strip		Half-length 0.4-g-B ¹⁰ Strip	
	Number	B ¹⁰ Content (g)	Number	B ¹⁰ Content ^a (g)
2	29	0.47	80A	0.20
3	28	0.50	56	0.19
4	22	0.50	72A	0.20
5	15	0.49	-	-
6	23	0.50	74A	0.20
7	40	0.49	72	0.20
8	35	0.50	56A	0.19
9	16	0.48	-	-
10	18	0.50	78A	0.19
11	19	0.47	-	-
13	20	0.49	76A	0.205
14	26	0.49	-	-
17	33	0.48	-	-
18	7	0.47	-	-
19	37	0.47	-	-
20	41	0.48	-	-
35	39	0.50	79A	0.195
38	27	0.50	74	0.20
39	36	0.51	63	0.19
41	21	0.50	78	0.19
42	42	0.49	80	0.20
43	34	0.48	-	-
44	25	0.48	-	-
45	30	0.50	-	-
46	13	0.48	-	-
47	6	0.47	-	-
48	38	0.51	79	0.195
49	4	0.48	-	-
50	32	0.50	76	0.205
51	9	0.47	-	-
52	31	0.49	63A	0.19
53	8	0.47	-	-
54	14	0.48	-	-
55	5	0.49	-	-
56	12	0.46	-	-
57	24	0.50	-	-
58	10	0.48	-	-
59	2	0.48	-	-
60	11	0.48	-	-
61	17	0.47	-	-

^a The B¹⁰ contents listed were computed as one-half of the B¹⁰ content of the full-length 0.4-g-B¹⁰ strip. These strips were cut in two to yield sixteen half-length strips, each containing nominally 0.2 g B¹⁰.

Table C-5

ASSIGNMENTS OF TEMPORARY 0.8-g-B¹⁰ STRIPS
IN LOADING NO. 60

Assembly Number	Full-length Strip	Half-length Strip
12	X-13	X-12
15	X-15	X-12
16	X-14	X-16
22	X-22	X-16
24	X-10	X-6
25	X-17	X-25
26	X-26	X-18
27	X-27	X-18
28	X-19	X-20
29	X-23	X-20
30	X-35	X-21
31	X-31	X-21
32	X-32	X-25
33	X-11	X-6

Appendix D

ORGANIZATION CHART AND OPERATING INSTRUCTIONS FOR
ZERO-POWER NUCLEAR TEST PROGRAMS

The following instructions were used during the experimentation.

1. Organization and Responsibilities (cf. Fig. 36)

A coordinator will have overall responsibilities for the nuclear tests to be performed. Responsible staff people will be notified when the approved alternate is temporarily in charge.

FIG. 36
ALPR OPERATIONS ORGANIZATION CHART

A Chief Operator will be responsible for all electrical systems and instrumentation used in performing nuclear tests. He will be in charge of the control room and all operations performed there. Reactor Operators who perform manipulations and take data in the control room will be supervised by the Chief Operator.

A Chief Physicist must be present at all times when alterations are being made to the reactor core or when operations which could affect

reactivity are being performed in the control room or on the reactor. Approved alternates may serve in the absence of the Chief Physicist.

All fuel for the reactor will be under the control of a Chief Loader. The Chief Loader will be responsible for an accurate inventory of the fuel at all times and for making all changes to the reactor core.

An independent check of alterations made to the reactor core will be made by a staff member designated by the Coordinator to verify that all fuel loadings are exactly in accordance with the approved loading plan. This check must be completely independent of the one made by the Chief Loader.

Before any fuel is loaded into the reactor or any alterations made to fuel loadings already in the reactor, a loading plan must be prepared by the Chief Physicist and reviewed by the Coordinator, the Chief Operator, and the Chief Loader. Only after this loading plan has been approved by the Coordinator and Chief Physicist will the Chief Loader take any action on the new experiment.

Approvals of the Chief Operator and Coordinator are required for any changes in the neutron source, neutron detectors, safety and alarm interlocks, or nuclear instrumentation.

(Argonne National) Laboratory personnel will be responsible for insuring that proper precautions are taken against radiation or other reactor hazards. A Health Physicist will be responsible for conformance with all AEC regulations pertaining to radiation safety. He will be responsible for maintaining records for personnel exposure, area surveys, waste disposal, and other matters relating to health physics.

Keys and key switches are used to lock out certain critical items of equipment on the control panel to prevent unauthorized manipulation. The keys will be controlled by the Coordinator.

These keys will be issued by the Coordinator. A record will be made when the keys are issued and to whom. At the termination of activities on a given day, all keys must be returned to the Coordinator.

2. General

The security control of the site will follow the practice of the Idaho Division of the Laboratory. The entire site will be under security control during the period when nuclear tests are being performed. Access to the building will be controlled from the site entrance. An approved list of people who may enter the site at any time will be supplied by the Coordinator and modified as necessary to keep the list as short as possible without

seriously hampering the work. If persons not on this list wish to enter the site, the guard will contact the Coordinator. If the Coordinator feels that conditions in the building at the time are such that it is impossible or inconvenient to admit the person, entry will be denied.

Personnel selected by the Coordinator will be stationed in the Support Facilities Building to prevent entry of personnel to the control room and to the operating floor. Upon authorization of the Coordinator, construction and other personnel, if necessary, will be admitted to do types of work that previously have been reviewed and approved by the Laboratory.

A list of approved personnel, Reactor Operators, and alternates for each of the major areas of responsibility will be posted in the control room log. The list will be expanded from time to time as other personnel gain experience in performing tests of this kind.

3. Routine Checks

a. Daily Checks

During the period of nuclear testing the daily checks will include the following:

1. Inspect the control room and reactor building. Report any abnormalities. Maintain good housekeeping and remove all unnecessary equipment.
2. Check the diesel-generator oil supply, start the diesel-generator, operate the unit for a few minutes, and shut it off.
3. Check to see that the building ventilation system is operating.
4. Check the solution level in the boric acid tank. Also, check for leakage.
5. Check the setting of the period meter.
6. Check the setting of the high-flux trip circuits.
7. Pull the antimony rod out of the beryllium block and take readings on the nuclear instruments. A significant change should occur in all readings of nuclear instruments. Restore the antimony rod to the beryllium block and verify that the nuclear instrument readings return to their former values. Care must be exercised to insure that adequate shielding is above the antimony source as it is being withdrawn.

8. Check to see that the emergency door to the outside stairway can be opened easily.
9. All pertinent operations connected with the preparation of experiments are to be recorded in the log book kept in the control room by the Chief Operator.

b. Pre-startup Checks

Before starting any experiment the following steps are to be carried out:

1. Make sure that only approved personnel are in the control room and the reactor building.
2. Verify that power is connected to the plant transformer, and energize all electrical buses.
3. Check to insure that the emergency power system is operating.
4. Check reactor water level to verify that it is at the desired level.
5. Check the radiation monitors to verify that all radiation levels are normal.
6. Check reactor water quality. Determine if boric acid is present, and, if so, the concentration by chemical analysis.
7. Determine that the dummy fuel assembly containing the neutron source is in the proper core location and that the antimony rod is inside the beryllium block.

4. Control Tests

Prior to beginning multiplication or critical experiments on any given day, operation of the control rods is to be checked. Each rod is to be raised to the all-out position* and dropped by pushing the "rod drop" button at the control console. The resulting insertion time should be less than two seconds. If any rod requires more than two seconds, the test should be repeated twice. If the rod drop time still exceeds two seconds for either of these two tests, the rod should be considered defective. If any rod is defective, no multiplication or critical tests are to be performed until the trouble is corrected.

*or to a predesignated safely subcritical position, as amended prior to the zero-power work.

5. Multiplication or Critical Experiments

All experiments will be performed in the following manner:

1. Two qualified Reactor Operators will be present.
2. A fuel-loading plan will be prepared by the Chief Physicist and reviewed by the Coordinator, Chief Operator, and Chief Loader. (It must be approved by the Chief Physicist and by the Coordinator.)
3. The approved fuel-loading plan will be given to the Chief Loader.
4. The Chief Operator will announce on the building communication system that an experiment is about to begin and will turn on power to the control rod drives.
5. Rods will be raised to a predetermined configuration chosen so that enough control is available to shut down the reactor by insertion of the central control rod alone, or by the insertion of two off-center cross control rods.
6. The Chief Loader will proceed to load the fuel assemblies, one at a time, into the water-filled core until the presented loading is achieved. The Chief Operator will be notified over the communication system as each fuel assembly enters the core and when each assembly is in place. The loading will be checked to verify that it is exactly according to the approved loading plan.
7. An announcement will be made over the communication system that all personnel must evacuate the reactor building. The Coordinator will check the reactor building to make certain that evacuation is complete.
8. The Chief Operator will withdraw the control rods slowly, while listening to the counters and watching the linear flux recorder. A Reactor Operator stationed next to the Chief Operator is responsible for changing ranges on nuclear instruments as ordered. The control rod drive motors must be stopped while instrument ranges are being changed. If a significant change in the counting rate occurs, control rod motion is stopped while accurate instrument readings are taken and recorded.
9. When necessary, the source and instrument tube locations will be changed by the Chief Loader in accordance with instructions by the Coordinator.

10. All pertinent data are to be recorded in the log book. The time when the operations were performed are to be recorded together with any significant observation of the results of the operations.

11. Experiments are to be numbered consecutively and are to be described clearly in the log book. Other pertinent data, such as fuel-loading charts, counting rates, and nuclear calculations, are to be neatly filed under the respective experiment or run numbers.

Appendix E

SIGNIFICANCE OF INTEGRATION OF CURVES OF
DIFFERENTIAL REACTIVITY EFFECTS

Two different approximate formulas may be derived from idealized computational models of the neutron removal effects of control rods in a thermal reactor, viewing the rods primarily as controllers of the neutron leakage or as noninteracting, uniformly distributed absorbers of thermal neutrons.

If it were possible to determine the reactivity effect of control rod motion from full insertion to full withdrawal in one step, then the primary effect in the ALPR would be the net removal of a "thermal" neutron absorber, that is, essentially a change in the average macroscopic thermal neutron absorption cross section $\Sigma_{a \text{ thermal}}$ in the core. However, the differential calibration procedure used was to establish a rod-bank critical, "bump" rods, and measure the corresponding positive asymptotic reactor period, and then change the concentration of the soluble poison in the reactor water to alter the critical position of the rod bank. Thus, the control system was calibrated starting with a control rod bank at a position of partial insertion. Two effects intervened during the sequence of incremental rod-bank withdrawals and poison additions: the neutron leakage from the reactor changed, as the effective core size was altered; and the value of $\Sigma_{a \text{ thermal}}$ in both the "rodded" and the "unrodded" regions of the core was changed.

If the value of $\Sigma_{a \text{ thermal}}$ had essentially zero effect on the neutron leakage from the system, then the total reactivity effect (ρ) of withdrawing the rod bank in N steps, from a position of partial insertion at Z_0 to a new position Z_N , would be related to the differential reactivity effects (ρ_j) by the equation⁽¹⁶⁾

$$\rho \approx 1 - \exp \left(- \sum_{j=1}^N \rho_j \right) \quad (\rho_j \text{ in units of } \%)$$

On the other hand, if the reactivity effect of soluble poison addition were unaffected by the positions of the control rods, and, in turn, if the differential worth of rod motion were unaffected by the presence of the soluble poison, then the local calibrations of the reactivity effect of poison addition could be used to infer the total reactivity effect of rod withdrawal, from the formula

$$\rho \approx - \sum_{j=1}^N \bar{\rho}_j$$

where $\bar{\rho}_j$ is the negative reactivity effect of poison addition at the j -th step.

Neither formula is applicable to ALPR, but there are elements of both models in the actual situation. The second formula, corresponding to the integration of the differential calibration curves, was adopted because of its simplicity, in the belief that it is as reasonable an approximation as the exponential formulation. If a total reactivity effect of 14% is determined with the summation formula, then a value of 13% would be computed from the exponential formulation.

REFERENCES

1. M. Treshow, ALPR Preliminary Design Study (Argonne Low Power Reactor) Phase I, ANL-5566 (April 1956).
2. Hazard Summary Report on the Argonne Low Power Reactor (ALPR), ANL-5744, completed October, 1957; Published November, 1958; Edited by A. Smaardyk.
3. Initial Testing and Operation of the Argonne Low Power Reactor (ALPR), Edited by Eberhard E. Hamer, ANL-6084 (Dec. 1959).
4. Ibid., p. 37.
5. R. W. Thiel et al., Early Operating Experience with the Argonne Low Power Reactor, Trans. of the Am. Nuc. Soc., Supplement, Vol. 2, No. 1, p. 160 (June 1959).
6. Ref. 2, p. 17.
7. Reactor Physics Constants, ANL-5800, Tables 1-15.
8. Design of the Argonne Low Power Reactor (ALPR), Edited by Eberhard E. Hamer, ANL-6076 (May 1960).
9. J. R. Dietrich, H. V. Lichtenberger, and W. H. Zinn, Design and Operating Experience of a Prototype Boiling Water Power Reactor, Proceedings of the International Conference on the Peaceful Uses of Atomic Energy (New York: United Nations, 1956), Vol. 3, p. 56.
10. J. E. Draley and W. E. Ruther, Corrosion-resistant Aluminum above 200°C, ANL-5430 (July 1955).
11. W. J. McGonnagle and R. B. Perry, Analyses of Fuel Element Core Blanks for Argonne Low Power Reactor by Gamma Counting, ANL-5944 (Dec. 1959).
12. R. L. Salley and W. R. Burt, Jr., Casting and Fabrication of Core Material for Argonne Low Power Reactor Fuel Elements, ANL-5950 (Dec. 1959).
13. Private Communication from W. R. Burt, Jr., ANL.
14. Ref. 12, p. 42.
15. Private Communication from D. E. Walker, ANL.
16. F. B. Fairbanks and J. G. Gallagher, APPR-I Control Rod Experiments and Calculations, TID-7532 (Pt. 1) (Oct. 1957).

ACKNOWLEDGMENTS

A detailed listing of participants in various phases of the design, construction, and testing of the Argonne Low Power Reactor has been published in ANL-6076.(8) It is the intent of the following paragraphs to acknowledge the contributions of the individuals who were associated more intimately with the actual zero-power program of reactor physics measurements.

Reactor Engineering (ANL)

A. D. Rossin and F. H. Martens collaborated with the writer in the detailed planning and execution of the physics experiments, and acted as Alternate Chief Physicist and (Alternate) Coordinator, respectively. As Coordinator, W. C. Lipinski supervised the overall operations of the groups of reactor physicists and engineers from ANL, and the military cadre. E. E. Hamer was the designated Chief Loader, with responsibility for review of proposed core loading changes and for the supervision of the approved modifications.

Idaho Division (ANL)

W. R. Wallin, the supervisory representative of the Idaho Division, was the Chief Operator. Under Mr. Wallin's direction, a training program for Reactor Operators and for Chief Operator was established, and a number of individuals, military and civilian, qualified in one or both categories. Messrs. Wallin, R. W. Thiel, and J. D. Cerchione had been closely associated with BORAX program experiments, and they played leading roles in various engineering capacities in the ALPR work. In particular, Mr. Thiel acted as Alternate Chief Operator, and Mr. Cerchione as Alternate Chief Loader. P. G. Stoddart fulfilled the responsibilities of Health Physicist, in collaboration with E. D. Graham. C. C. Miles provided assistance by his analysis of reactor water samples.

Military Cadre

It is difficult to single out individuals, in this case, for essentially the entire cadre participated quite actively in a variety of ways during the zero-power experiments. A policy of extensive participation was established to provide additional training in reactor "operation" and insight into the details of the reactor physics of the ALPR. The members of the military cadre demonstrated unusual personal scientific interest and devotion to duty that were of material benefit to the success of the program. The list of Cadre personnel is included below.

We should like to express our particular gratitude to Capt. R. A. Roberge, 1/Lt. J. T. Artman, WO-3 R. E. Embleton, and M/Sgt. M. B. Hobson for their cooperation in the various flux-mapping experiments.

Military Cadre

Maj. D. A. Ayres	Cadre Chief
Capt. R. A. Roberge	Training Officer
1/Lt. J. T. Artman	Operations Officer
WO-3 L. S. Hicks	Plant Superintendent
WO-3 R. E. Embleton	Chief Operator Mechanical Supervisor
WO-1 D. K. Brown	Chief Operator Electronics Supervisor
WO-1 W. C. Mathers	Chief Operator Electrical Supervisor
CPO H. J. Miller	Chief Operator Mechanical Technician
M/Sgt. H. M. Curran	Chief Operator Electrical Technician
M/Sgt. M. B. Hobson	Chief Operator Electrical Technician
M/Sgt. R. C. Lewis	Chief Operator Electronics Technician
M/Sgt. J. A. Parks	Chief Operator Mechanical Technician
M/Sgt. G. E. Paull	Health Physicist
T/Sgt. M. V. Sitton	Reactor Operator Electronic Technician
SFC R. M. Bishop	Reactor Operator Mechanical Technician
SFC P. J. Conlon	Reactor Operator Electronics Technician
SFC L. L. Dole	Health Physicist
SFC R. L. Honeycutt	Reactor Operator Mechanical Technician
SP-5 E. O'Neal	Reactor Operator Electronics Technician

General Acknowledgments

The cooperation of the other staff members of Reactor Engineering associated with the design, construction, and testing of the ALPR as members of the ALPR Group is gratefully acknowledged, notably including C. R. Braun (ALPR Project Engineer), H. H. Hooker, W. J. Kann, G. C. Milak, and A. Smaardyk, all of whom were active participants in the ZPR measurements, at various times.

M. Novick (Director) and F. W. Thalgott (Associate Director) of the Idaho Division cooperated in providing whatever supporting efforts were required from the Idaho Division. On a more personal level, both individuals offered a number of helpful suggestions, of particular value during the first sequence of core loadings. A. W. Solbrig, Jr., assisted in the physics program, principally in the correlation of experimental data.

The direct assistance of the Metallurgy Division (ANL), by the prompt fabrication of the temporary 0.8-g-B^{10} strips, made it possible to continue the ZPR experiments without interruption.

The writer is particularly indebted to E. E. Hamer, Reactor Engineering (ANL), for his technical review of the sections describing the reactor components.