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A SUMMARY OF DATA FROM
THE ATRC EXPERIMENTAL PROGRAM

J. W. Henscheid, N. C. Kaufman, and J. L. Durney

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IN NUCLEAR SCIENCE ABSTRACTS

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ABSTRACT

The ATRC first attained criticality in May 1964. However, it was not until October 1965 that the first core of actual ATR fuel elements was installed in the ATRC. This core contains 39 kg U-235 and 165 g natural boron. The most significant results of the subsequent experimental program in the ATRC are the following:

- (1) Shutdown reactivity in the five cadmium safety rods is between 8 and 10 \$ depending upon power distribution and shim configuration. The hafnium safety rods that will be used in the ATR are worth about 35 percent more.
- (2) The kinetic parameter β/Λ ranges from about 100 to 185 sec^{-1} , being a nearly linear function of the outer shim positions.
- (3) The total reactivity worth of all 24 neck shims ranges from 6.4 to 9.9 \$ depending upon outer shim positions. The worth of neck shim NE4 or SW4 (corresponding to ATR reg rods) varies from 0.2 to 0.5 \$ depending upon the configuration of the other neck shims and outer shims.
- (4) The total reactivity worth of all outer shims varies from 8.7 to 11.8 \$ depending upon the positions of the neck shims.
- (5) Total excess reactivity with no experiments in the flux traps is 11.3 \$. An additional 11.1 \$ in reactivity is held down by the burnable poison, boron. The four fixed shims in the center flux trap hold down an additional 1.5 \$.
- (6) The positive reactivity effect of voiding a flux trap is a maximum (approximately 1.7 \$) under the following conditions:
 - (a) the flux trap is in the high-power lobe of a 40-50-60 power division
 - (b) the flux trap contains a safety rod guide tube without filler plates (ie, minimum metal/water ratio)
 - (c) the experiment flow tube contains only water (no fuel, minimum metal/water ratio).
- (7) The void coefficient of reactivity in the fuel elements is generally negative except in water channels 1 and 20 where the coefficient is slightly positive.
- (8) The average fuel coefficient of reactivity is about 0.001 \$/g U-235. The average boron coefficient of reactivity in the fuel annulus is 0.067 \$/g natural boron.
- (9) Detailed azimuthal and radial fission-rate profiles in the fuel elements show severe peaking in several fuel positions in both

50-50-50 and 40-50-60 power divisions with and without simulated xenon poisoning.

- (10) Axial fission-rate profiles in the fuel annulus show local peaking due to vents in the fuel element sideplates. .
- (11) Reversing the direction of rotation of the outer shims generally flattens the overall power distribution in each lobe and reduces the hot-spot fission rate as much as 15 percent in one case. Neither the shape nor the magnitude of the outer shim calibration curves change noticeably. The shutdown reactivity and flux trap void effects remain essentially unchanged.

CONTENTS

ACKNOWLEDGEMENTS	ii
ABSTRACT	iii
I. INTRODUCTION	1
1. INITIAL STARTUP	1
2. COMPARISON OF EXPERIMENT AND COMPUTATION	1
3. 39-kg CORE	3
4. ATRC HOT LOOP	3
II. SHUTDOWN REACTIVITY IN SAFETY RODS	5
1. SAFETY ROD CALIBRATION CURVE	5
2. "STUCK ROD" EXPERIMENT	5
3. CHANGES IN SAFETY ROD WORTH	6
4. ATR-ATRC SAFETY RODS	6
III. MEASUREMENTS OF β/Λ	7
IV. SHIM CALIBRATIONS	9
1. ATR SHIM SYSTEMS	9
2. NECK SHIMS	9
3. ATR REG RODS	10
4. OUTER SHIMS	10
V. EXCESS REACTIVITY	14
1. REACTIVITY HELD DOWN BY BORON	14
2. EXCESS REACTIVITY IN SHIMS	14
3. FUEL ELEMENT BORON CONTENT	16
VI. VOID REACTIVITY EFFECTS	17
1. FLUX TRAP VOIDS	17
2. FUEL ELEMENT VOIDS	19
VII. EXPERIMENT REACTIVITY EFFECTS	21

VIII. NEUTRON FLUX AND POWER DISTRIBUTIONS	23
1. INITIAL MEASUREMENTS	23
2. 50-50-50 POWER DIVISION.....	23
3. 40-50-60 POWER DIVISION	27
4. FISSION-RATE PROFILES IN FUEL ELEMENTS	30
5. THERMAL NEUTRON FLUX IN EXPERIMENT HOLES	34
6. FAST NEUTRON FLUX	36
IX. EFFECTS OF REVERSING OUTER SHIM ROTATION	43
1. INITIAL INVESTIGATIONS	43
2. EFFECT OF OUTER SHIM REVERSAL ON POWER DISTRIBUTION.....	43
2.1 50-50-50 Power Division	43
2.2 40-50-60 Power Division	47
2.3 Fission-Rate Profiles in Fuel Elements.....	51
3. OTHER EFFECTS OF OUTER SHIM REVERSAL	53
X. REFERENCES	55
APPENDIX A -- PHYSICAL CONSTANTS FOR AN ATR FUEL ELEMENT	59
APPENDIX B -- REACTIVITY AS A FUNCTION OF REACTOR PERIODS IN ATRC	63

FIGURES

1. Cross-sectional diagram of the ATRC core.....	2
2. ATRC safety rods reactivity calibration curve	5
3. β/Λ as a function of outer shim position	7
4. Normalized reactivity calibration curve for an ATRC neck shim.....	9
5. Reactivity worth of all neck shims as a function of the position of the outer shims	10
6. Reactivity worth of NE4 and SW4 neck shims as a function of outer shim position and neck shims withdrawn	11
7. Normalized reactivity calibration curves for a single outer shim pair in two different shim configurations	12

8. Outer shim reactivity calibration curves for three neck shim configurations	13
9. Reactivity worth of a single outer shim pair versus adjacent inner flux trap composition.	13
10. Reactivity calibration curves of the outer shims around the NE flux trap showing interaction between adjacent pairs of outer shims	13
11. Core-averaged boron reactivity profile across fuel annulus in 39-kg core	15
12. Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers	18
13. Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers	18
14. Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers	18
15. Void coefficient as a function of radial position in fuel element positions 12, 19, 32, and 39 in a 50-50-50 power division in a 39-kg core	20
16. Reactivity effect of temperature change in the ATRC Hot Loop with no experiment	22
17. Cross-sectional diagram of ATRC core showing shim configuration for a nominally 50-50-50 power division	24
18. Cross-sectional diagram of ATRC core showing the power distribution for a nominally 50-50-50 power division normalized to 250 MW	25
19. Cross-sectional diagram of the ATR-ATRC fuel element	25
20. Peaking factors and element powers in a 50-MW lobe (SW) with and without simulated xenon poisoning	26
21. Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with simulated xenon poisoning	27
22. Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division	28
23. Peaking factors and element powers in a 60-MW lobe (NW) with and without simulated xenon poisoning	29
24. Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with simulated xenon poisoning	30
25. Quarter-core power distribution for a nominally 60-50-40 power division with simulated xenon poisoning and with a flux trap filler in the 60-MW lobe	31

26. Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 26, 27, 28, and 29 in a nominally 50-50-50 power division with and without simulated xenon poisoning	32
27. Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 32, 33, 34, and 35 in a nominally 40-50-60 power division with and without simulated xenon poisoning	32
28. Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 26, 27, 28, and 29 in a nominally 50-50-50 power division with and without simulated xenon poisoning	33
29. Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 32, 33, 34, and 35 in a nominally 40-50-60 power division with and without simulated xenon poisoning	33
30. Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel position 28 showing the effect of the sideplate vents on the fission rate peaking.	34
31. Axial fission-rate profile near the center of fuel position 28	35
32. Axial fission-rate profile near the edge of water channel 11 in fuel position 13 showing the sideplate vent perturbations	35
33. Radial fission-rate profile through plates 1, 6, 11, 15, and 19 in fuel position 13	36
34. Axial fission-rate profile in center flux trap baffle position H4	39
35. Axial fission-rate profile in reflector position I4	39
36. Axial fission-rate profile in the NW flux trap.	40
37. Axial fission-rate profiles in the NE and SW flux traps showing the effect of a withdrawn safety rod,	40
38. Cross-sectional diagram of ATRC core showing fast (>1 MeV) neutron flux at reactor horizontal midplane for a nominally 40-50-60 power division normalized to a reactor power of 250 MW	41
39. Axial fast-neutron flux profiles in B1 and H15	42
40. Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with the outer shims reversed.	44
41. ATRC core diagram showing the shim positions for the nominally 50-50-50 power division before and after outer shim reversal	45
42. Peaking factors and element powers in a 50-MW lobe (SW) before and after outer shim reversal.	46
43. Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with simulated xenon poisoning and reversed outer shims	47
44. Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with the outer shims reversed.	48

45. Peaking factors and element powers in a 60-MW lobe (NW) before and after outer shim reversal	49
46. Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with simulated xenon poisoning and reversed outer shims.	50
47. Peaking factors and element powers in a 60-MW lobe with a flux trap filler before and after outer shim reversal	50
48. Quarter-core power distribution normalized to 250 MW for a nominally 60-50-40 power division with simulated xenon poisoning and reversed outer shims	51
49. Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 26, 27, 28, and 29 with and without simulated xenon poisoning in a nominally 50-50-50 power division	52
50. Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 32, 33, 34, and 35 with and without simulated xenon poisoning in a nominally 40-50-60 power division.	52
51. Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 26, 27, 28, and 29 with and without simulated xenon poisoning in a nominally 50-50-50 power division.	53
52. Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 32, 33, 34, and 35 with and without simulated xenon poisoning in a nominally 40-50-60 power division	53
53. Reactivity calibration curve of outer shim pair N34 before and after reversing outer shims	54
B-1 Reactivity equivalent of stable positive periods in ATRC	65

TABLES

I. Core Conditions for ATRC 35-kg and 39-kg Cores	4
II. Reactivity Worths for NE4 and SW4 Neck Shims	11
III. Excess Reactivities and Shim Configurations in 39-kg Core	15
IV. Reactivity Effect of Removing 88 Percent of the Water from a Flux Trap	19
V. Reactivity Effects in a 50-50-50 Power Division	21
VI. Horizontal Midplane Fission-Rate and Thermal Neutron Flux Values at 250 MW	37
VII. Flux Trap Fission Rates at Reactor Horizontal Midplane	38
VIII. Reactivity Effects Before and After Outer Shim Reversal	54
A-I. Physical Constants for an ATR Fuel Element	61

ATR CRITICAL FACILITY (ATRC)

A SUMMARY OF DATA FROM THE ATRC EXPERIMENTAL PROGRAM

I. INTRODUCTION

The ATR Critical Facility (ATRC), a low-power, full-scale, nuclear duplicate of the ATR, has the primary purpose of providing technical support to the ATR. When the decision was made to include the ATRC in the design and construction of the ATR, it was planned and expected that the ATRC would be built and in operation about one year prior to ATR startup. It was assumed that a one-year experimental program in the ATRC would be adequate to investigate and resolve the questions or uncertainties relevant to initial operation of the ATR. However, because of many unforeseen delays in the ATR construction schedule, the ATRC experimental program is now well into its third year, and ATR startup is still several months away. Although this time extension to the ATRC program has permitted much more extensive measurements than were originally planned, it is also evident in light of what has transpired that the one-year program would probably not have been sufficient to provide the requisite data for ATR startup.

The purpose of this report is to summarize the results of the ATRC experimental program insofar as they relate to the operation of the ATR. Preliminary data or data obtained solely to evaluate the ATRC have generally been excluded from this report.

1. INITIAL STARTUP

On May 19, 1964, the ATRC attained criticality for the first time. Because of a delay in the delivery of the neck shim housing and center flux trap baffle, the housing and baffle from the initial ATR critical experiment in Lynchburg, Virginia, were used temporarily in the ATRC in order to proceed with the experimental program. The fuel loading consisted of forty 875-g fuel elements for a total of 35 kg U-235. The subsequent experimental program verified that the ATRC could be operated safely and within the limits established in the ATRC Safety Analysis Report[1]. The results of the measurements made during this initial period of operation have been reported in Reference 2.

With the arrival of the permanent neck shim housing in July 1964, the ATRC was shut down for nearly six weeks while the new neck shim housing, the 24 neck shim drives, and the center flux trap baffle were installed. During the subsequent startup[3], it was found that these changes reduced the excess reactivity approximately 0.5 \$. (A cross-sectional diagram of the ATRC core is shown in Figure 1.)

2. COMPARISON OF EXPERIMENT AND COMPUTATION

Following the startup program, all of the experiment irradiation holes were filled with mock-ups of the experiments used in the comprehensive fuel depletion calculations by the Babcock and Wilcox Company[4]. This was done in order that the ATRC measurements would provide a means for comparing experimental and computational models. The core conditions for the ATRC

Fig. 1 Cross-sectional diagram of the ATRC core.

(listed in Table I) were very similar to those in the fuel depletion calculations by Babcock and Wilcox. However, the ATRC core differed from the core postulated in the depletion calculations in one major respect: the U-235 loading in the ATRC was 35 kg compared with 39 kg in the depletion calculations. In spite of this difference, the shim positions required to attain a nominally 40-50-60 power division [a] in the ATRC were nearly the same as the shim positions for the zero-time step in the fuel depletion calculations[5]. In fact, the difference between the excess reactivity obtained in the depletion calculation and in the ATRC measurements was only about one dollar. Based upon the average U-235 coefficient of reactivity measured in the ATRC, a 4-kg increase in U-235 content would increase excess reactivity approximately 4\$.

A comprehensive experimental program was undertaken in this 35-kg core which included detailed measurements of power distribution, flux trap voiding effects, shutdown reactivity, and shim calibrations. The ATRC also was used to evaluate some of the shim components from the ATR as well as all of the ion chambers and fission counters for the ATR control system.

3. 39-kg CORE

In October 1965, the first fuel elements actually designated for the ATR arrived. These elements each contained 975 g U-235 (100 g more than those in the ATRC) and had vents in the sideplates (see Reference 7, page 141). The first 40 of these new 975-g fuel elements, containing a total of 39 kg U-235 and 165 g natural boron, were designated as the startup core for the ATR. This 39-kg core was then installed in the ATRC in place of the 35-kg core in order to perform a "dress rehearsal" of the low-power measurements planned for the ATR startup program. In order to duplicate the conditions planned in the ATR during startup, all of the loop and capsule experiment mock-ups and two of the nine pressure tube mock-ups were removed from the ATRC. The core conditions for the measurements made in the 39-kg core are shown in Table I along with the conditions prevailing with the 35-kg core.

4. ATRC HOT LOOP

The experiment pressure tubes in the ATRC simulate only the in-pile portion of the actual pressure tubes in the ATR; consequently, the water temperature in these pressure tube mock-ups is at the same ambient temperature as the reactor water. However, in order for the ATRC to provide an even closer duplication of a typical loop in the ATR, a high-temperature pressurized water loop was installed in the NE flux trap of the ATRC. This loop, designated ATRC Hot Loop, was designed to provide operating temperatures as high as 550°F, typical of the operating temperature of the ATR water loops. Measurements in this loop include neutron flux distributions in prototype experiments and reactivity changes due to spectrum hardening and Doppler broadening in materials with nuclear resonances. A more detailed description of this loop is given in Reference 8.

[a] In a 40-50-60 power division, the power generated in the NE and SW lobes is proportional to 40 MW each, the power in the center lobe is proportional to 50 MW, and the power in the NW and SE lobes is proportional to 60 MW each. In a 50-50-50 power division, equal power is generated in all five lobes. A lobe is defined as the eight fuel elements surrounding an inner flux trap.

TABLE I
CORE CONDITIONS FOR ATRC 35-kg AND 39-kg CORES

Core Position	Composition in 35-kg Core ^[a]	Composition in 39-kg Core ^[a]
NW flux trap	A1 experiment ^[b] , flow tube, standard NR pressure tube, and safety rod guide tube.	Same as the 35-kg core but without the A1 experiment.
N flux trap	A5 experiment ^[b] , flow tube, standard NR pressure tube, and safety rod guide tube with flux trap fillers.	Same as the 35-kg core but without the A5 experiment.
NE flux trap	A5 experiment ^[b] , flow tube, standard NR pressure tube, and dummy flux trap filler.	Same as the 35-kg core but without the A5 experiment.
W flux trap	Same as NE flux trap.	Dummy pressure tube, center tube, and dummy flux trap filler.
Center flux trap	A3 experiment ^[b] , flow tube, and standard NR pressure tube.	Same as the 35-kg core but without the A3 experiment.
E flux trap	Same as NE flux trap.	Dummy pressure tube and dummy flux trap filler.
SW flux trap	Same as N flux trap.	Same as N flux trap.
S flux trap	Same as N flux trap.	Same as N flux trap.
SE flux trap	Same as NW flux trap.	Same as NW flux trap.
H holes	Quarter-inch hafnium rods in holes 1, 5, 9, and 13; N16 monitor tubes in holes 3 and 11; water in the ten remaining holes.	Standard ATR Hf shims in holes 1, 5, 9, and 13; the remaining holes identical to 35-kg core.
A holes	Half-inch steel rods.	Solid aluminum fillers.
B holes	5/8-inch steel rods.	Solid aluminum fillers except B1 which has steel rod.
I holes	Aluminum fillers.	Aluminum fillers.

[a] For greater detail in descriptions, dimensions, and material composition, see References 1 and 6.

[b] All flux trap experiments contain 64 aluminum rods and have a metal-to-water ratio of 2.77. The rods in the A1 experiment contain 26.8 g U-235; in the A3 experiment, 53.6 g U-235; and in the A5 experiment, 80.3 g U-235.

II. SHUTDOWN REACTIVITY IN SAFETY RODS

Since the emergency shutdown reactivity in the ATR and ATRC is held down entirely by the five safety rods, the reactivity worth of these rods has been measured extensively. The techniques of poison substitution and reactor periods, rod drop, and pulsed neutrons have been used in these measurements. The reactivity worth of a safety rod has been found to depend upon the reactor power division, the location of the rod, and the composition of the flux trap in which it is located. Additionally, measurements have been made to relate the reactivity worth of the ATRC cadmium rods to the worth of the ATR hafnium rods.

1. SAFETY ROD CALIBRATION CURVE

Shown in Figure 2 is a measured curve for the reactivity worth of the entire set of five ATRC safety rods as a function of rod position. These data were obtained with the pulsed-neutron technique[9] in a 39-kg core with the reactor core conditions as described in Table I and with a nominally 50-50-50 power division. This curve is slightly but clearly asymmetrical due to the perturbation of the axial flux distribution resulting from the partial insertion of the rods[a]. The total safety rod worth indicated in Figure 2 (8.8 ± 0.3 \$) agrees well with rod drop reactivity measurements[10] made with the same core conditions.

Fig. 2 ATRC safety rods reactivity calibration curve.

2. "STUCK ROD" EXPERIMENT

In support of the ATR safety analysis, the so-called "stuck rod" situation was evaluated by measuring the shutdown reactivity of only four of the five safety rods, selecting as the "stuck rod" the safety rod with the greatest worth. In order to determine which rod has the greatest worth, measurements were made to determine how the total shutdown reactivity was distributed among the five rods[11]. The measurements indicated that a safety rod in an outer flux trap was worth 10 percent more than a rod in an inner flux trap of identical composition. Furthermore, the addition of a flux trap filler to a safety rod guide tube enhanced its worth by 35 to 40 percent. Thus, the safety rod with a maximum reactivity effect would be in an outer flux trap and have flux trap filler plates on its follower section. The minimum worth of four safety rods was then determined in a rod drop experiment, dropping all but the "greatest worth" rod. The shutdown reactivity obtained was 6.7 ± 0.2 \$. The rod worth the most (about 2.1 \$) was located in the S flux trap.

[a] The poison section of a safety rod is only 36 inches long, whereas the core height is 48 inches. In the scrambled position, the bottom of the poison section is 9 inches above the bottom of the core. When the rod is withdrawn 36 inches to its upper limit, the bottom of the poison section is still 3 inches below the top of the core.

3. CHANGES IN SAFETY ROD WORTH

When a fueled experiment train is inserted into a flux trap which previously contained water only, the reactivity worth of a safety rod in that flux trap can be expected to increase. Measurements of this effect indicate that the rod will increase by 0.22 percent of its worth for each gram of U-235 uniformly distributed over the full length of the experiment^[11]. However, these measurements also indicated that, if the fueled train were enclosed with hafnium shrouds, the effect of the fuel addition would be neutralized.

The reactivity worth of the safety rods has been found to change as the reactor power division is changed from the nominally 50-50-50 power division used for the measurements above^[9, 10]. The worth of the rods in a 60-50-40 power division (two rods in lower and one in higher power regions) was measured to be 8.2 ± 0.3 \$ using the pulsed-neutron technique. This value is in reasonable agreement with a value of 8.5 \$ derived from the measurement in the 50-50-50 power division assuming (a) that there is no safety rod interaction, (b) that the worths can be apportioned as discussed above, and (c) that the worths change in proportion to the change in the square of the power in the adjacent fuel elements. Using this same semi-calculational method, the worth of the rods in a 40-50-60 power division (two rods in higher power and one in lower power regions) is estimated to be 9.4 \$. This is in good agreement with a value of 9.6 ± 0.4 \$ obtained by summing values obtained in this power division with period calibrations for each rod. The minimum worth of the five rods was estimated to occur with the power peaked toward the center of the reactor by fully inserting outer shims and withdrawing most of the neck shims. A rod drop measurement with this shim configuration indicated that the rods were worth 8.2 ± 0.2 \$.

4. ATR-ATRC SAFETY RODS

The safety rods in ATR have hafnium absorber plates rather than the cadmium plates used in ATRC. Measurements have been made in which the worth of a single rod was measured with cadmium plates and again with hafnium plates^[11]. The hafnium plates were found to increase the worth of the rod by 30 to 40 percent depending on the flux trap and its composition. Consequently, reasonable estimates for the ATR equivalent for each of the values measured in ATRC may be obtained by increasing the measured values by 35 percent.

III. MEASUREMENTS OF β/Λ

The ratio of the effective delayed neutron fraction to the neutron generation time^[a], or β/Λ , is an important kinetic parameter. Consequently, the value for this parameter has been measured in ATRC under several reactor conditions. The measurements have used the techniques of both auto-power and cross-power spectral density analyses^[12] and of pulsed-neutron analysis^[13] (both when critical and subcritical) with the Garelis-Russell method^[14]. Because it had been calculated that β/Λ would vary by as much as 40 percent as the outer shim position was changed^[15], measurements have been made in the ATRC with several different outer shim positions to determine the change experimentally.

A summary of the values obtained for β/Λ in the 39-kg core is presented in Figure 3. Only values obtained in cross-power analyses and with the pulsed-neutron technique are shown in the figure since auto-power results did not

Fig. 3 β/Λ as a function of outer shim position.

prove to be reproducible within what was believed to be the experimental error. (However, they differed by no more than six percent from the values shown.) The values presented in Figure 3 for the 39-kg core are approximately 10 percent greater than those measured in the 35-kg core at the same outer shim positions^[16]. This difference is due to the increase in fuel loading as well as to the addition of neutron absorbers to compensate for the resultant increase in reactivity. All of these values are well within the range of values considered in the preparation of safety analyses for both ATR and ATRC.

[a] Neutron generation time: $\Lambda \doteq \ell/k$ where ℓ is the neutron lifetime and k is the effective multiplication factor.

Preliminary pulsed-neutron results indicate that, in a nominally 40-50-60 power division, the value for β/Λ is that shown in Figure 3 for the outer shim position corresponding to the position of the most withdrawn outer shims. Another notable observation is that the value for β/Λ , measured by pulsed neutrons with the reactor subcritical, is the same for all positions of the safety rods. This latter observation was unexpected in view of the heterogeneous configuration of the ATRC.

IV. SHIM CALIBRATIONS

1. ATR SHIM SYSTEMS

There are two shim systems in the ATR: neck shims and outer shims. The neck shim system is comprised of 24 hafnium rods within the central aluminum support housing, two of which (NE4 and SW4) function as reg rods in ATR. The 24 rods are positioned in sets of six in the extensions or necks of the central housing and each set is identified by its orientation about the center of the reactor (NE, SW, SE, NW). The members of each set are identified by numbering from one to six outwardly from the center. The outer shim system comprises 16 beryllium cylinders with curved hafnium plates attached. These cylinders are located in the reflector and are driven in pairs. Individual cylinders are identified by their general orientation about the center (N, S, E, W), and each of the four cylinders on a side is numbered clockwise from 1 to 4. Since they are operated in pairs, each pair is designated by a compass direction and the numbers of the two cylinders in a pair, eg, N12 (north-one-two). The shim numbering system may be further clarified by referring to Figure 1.

The two shim systems are used for both the maintenance of criticality and the establishment and maintenance of a given reactor power distribution. In performing this latter design function, each shim system has the ability to shift the reactor power distribution as its configuration is varied; and, in doing so, it can change the reactivity worth of both the members of the other shim system and other members of its own system. This shim interaction is very pronounced in ATRC, and it requires that each shim worth be qualified by a specification of the position of the other shims.

2. NECK SHIMS

Since the neck shims move vertically in a neutron distribution well described by a cosine, the curve of the reactivity as a function of position has the familiar lazy S shape. Such a measured curve, normalized to unit reactivity, is shown in Figure 4. Since the outer shim absorber sections are full-core length, the shape of a neck shim curve is invariant with outer shim position. However, the total reactivity worth of a neck shim is strongly dependent upon the outer shim positions.

As may be seen in Figure 5, the variation of the total worth of all neck shims over the range shown is nearly a linear function of the position of the ganged set of outer shims. The total reactivity worth of all neck shims is estimated to range from 9.9 \$, with all outer shims inserted, to 6.4 \$, with all outer shims fully withdrawn. In addition to these variations in neck shim worth due to outer shim position, the worth of any given neck shim

Fig. 4 Normalized reactivity calibration curve for an ATRC neck shim.

Fig. 5. Reactivity worth of all neck shims as a function of the position of the outer shims.

depends on whether the adjacent neck shims are inserted or withdrawn. For example, with all other neck shims inserted, the worth of the NE4 neck shim can be increased 25 percent by fully withdrawing the adjacent NE3 neck shim.

3. ATR REG RODS

The reactivity worths of the NE4 and SW4 neck shims in ATRC are of particular interest since they correspond to the ATR reg rods. The worth of these rods is shown in Table II for a cold, clean reactor^[17]. In this table, a measured worth is shown for these rods when in a nominally 50-50-50 power division and when adjacent to the low-power lobe of a nominally 40-50-60 power division. An estimated value is also shown for these rods when adjacent to the high-power lobe of a 40-50-60 power division. As the operating cycle of the ATR progresses, these worths will increase to 0.4 \$ or 0.5 \$, due to withdrawal of the other neck shims. Summarized in Figure 6 are the results of many measurements of the reactivity worths of NE4 and SW4 under various neck shim and outer shim configurations.

4. OUTER SHIMS

The shape of the curve for the reactivity worth of an outer shim pair as a function of its position depends on the position of the neck shims and the other outer shims. This effect is illustrated in Figure 7, which shows a calibration curve for a single outer shim pair under two different shim configurations. The difference between these curves, both of which have been

TABLE II

REACTIVITY WORTHS FOR NE4 AND SW4 NECK SHIMS
(Correspond to ATR reg rods)

Shim Positions	Worth of One Reg Rod (\$)
Positions for beginning-of-cycle 50-50-50 power division	0.217 ± 0.002
Positions for beginning-of-cycle ^o 40-50-60 power division	
(a) If reg rod in 40-MW lobe	0.203 ± 0.003
(b) If reg rod in 60-MW lobe	$0.24[a]$

[a] This value is an estimation based on the square of the average flux in the four most nearly adjacent fuel elements.

Fig. 6 Reactivity worth of NE4 and SW4 neck shims as a function of outer shim position and neck shims withdrawn.

Fig. 7 Normalized reactivity calibration curves for a single outer shim pair in two different shim configurations.

normalized to unit reactivity, is due to a shifting of the neutron distribution within the reflector. In addition to these shape changes, there are changes in magnitude as the core composition and shim positions are changed.

Presented in Figure 8 are curves describing the reactivity worth of the full set of outer shims as a function of their position. There are three curves: one for neck shims inserted, one for neck shims withdrawn, and one for half of the neck shims withdrawn. These curves were measured with the reactor core conditions described in Table I during an excess reactivity determination using a 50-50-50 power division.

The reactivity worth of an outer shim pair has been found to change as the composition of the most nearly adjacent flux trap is changed[18]. These changes have been measured for three flux trap metal-to-water ratios with three different fuel loadings in the experiment flow tube. However, the metal-to-water ratio of the contents of the flow tube was the same in every case. The data obtained in these measurements is summarized in Figure 9. In addition, the worth of an outer shim pair was found to increase by three percent when the temperature within the water-filled tube (no experiment) of the adjacent flux trap (with filler) was increased from 70 to 500° F[8].

The worth of one outer shim pair has been found to be significantly influenced by the position of the most nearly adjacent outer shim pair[18]. This influence is consistent with the concept of rod shadowing; that is, the worth of an outer shim pair is depressed by inserting the adjacent pair. The curves in Figure 10 illustrate this by comparing the sum of the worths of two shim pairs, each measured with the other inserted, to the worth of the two pairs operated in unison.

Fig. 8 Outer shim reactivity calibration curves for three neck shim configurations.

Fig. 9 Reactivity worth of a single outer shim pair versus adjacent inner flux trap composition.

Fig. 10 Reactivity calibration curves of the outer shims around the NE flux trap showing interaction between adjacent pairs of outer shims.

V. EXCESS REACTIVITY

The length of time a reactor will operate with a given fuel and experiment loading, commonly called charge life or lifetime, is determined largely by the reactivity held down by control rods and burnable poisons. Consequently, it is very important in any reactor that this hold-down reactivity be known. In the ATR and ATRC the reactivity of the cold, clean core is held down by the movable shims (neck shims and outer shims) and by a burnable poison in the form of boron carbide particles in the fuel plates. The reactivity held down by the shims (called excess reactivity) is affected by the flux trap experiments and by the desired lobe-power distribution. Of particular current interest is the excess reactivity in a core containing no flux trap experiments operating in a nominally 50-50-50 power division.

A previous measurement of excess reactivity was made on the 35-kg core[2], but no attempt was made to maintain a given power distribution with different shim positions. Those results were then extrapolated to the 39-kg core with fueled experiments in the flux traps[19]. However, since the core conditions were changed significantly after the 39-kg fuel loading was installed, it was deemed necessary to remeasure the reactivity held down by the boron and the shims[a]. The ATRC core conditions for the measurements were identical to those anticipated for ATR startup with the 39-kg core described in Table I.

1. REACTIVITY HELD DOWN BY BORON

To permit shim withdrawal in the ATRC beyond the cold, clean critical point, plastic strips containing amorphous natural boron were added stepwise to the water channels of the fuel elements. Simultaneous shim calibrations provided a measurement of the worth of the boron added in each step. In each poison step the boron strips were inserted in the same water channel in each fuel element to obtain the boron reactivity profile radially across the fuel annulus as shown in Figure 11. This profile was then averaged across the fuel annulus to obtain a core-averaged boron coefficient of reactivity, 0.067 \$/g. The value 11.1 \$ for the reactivity held down by the burnable poison in the fuel elements is the product of this coefficient and the 165 g of natural boron in the fuel. Points obtained on this profile after reaching simulated xenon equilibrium conditions were not used in the averaging since they were probably influenced by self-shielding.

2. EXCESS REACTIVITY IN SHIMS

The excess reactivity held down by the movable shims in the ATRC has been somewhat arbitrarily divided into three parts: (a) that required to compensate for the buildup of equilibrium xenon in the ATR, (b) that required to reach the point in ATR core life at which there is insufficient shimming left to maintain the 50-50-50 power division, and (c) that held down by the shims remaining in the core after the 50-50-50 power division can no longer be

[a] The results of these excess reactivity measurements have already been reported[20]. However, the results given here differ slightly in that they include a small correction to account for a small amount of neutron absorbing material other than boron in the plastic strips used in the measurements.

Fig. 11 Core-averaged boron reactivity profile across fuel annulus in 39-kg core.

maintained. These partial excess reactivities, obtained by summing the reactivity increments of the poisoning sequence, along with the corresponding shim configurations are shown in Table III. The sum of all reactivity increments, 11.3 \$, is the total excess reactivity held down by the movable shims. This is nearly equal to the 11.1 \$ held down by the burnable poison. The four fixed shims in the center flux trap baffle held down an additional 1.5 \$.

TABLE III

EXCESS REACTIVITIES AND SHIM CONFIGURATIONS IN 39-kg CORE

Core Conditions	Shim Configuration		Excess Reactivity of Step[a]
	Outer Shims	Neck Shims	
Cold, clean	All at 77.5°	NE4 and SW4 withdrawn; all others inserted	0
Xenon equilibrium conditions in 50-50-50 power division	All at 106°	All No. 1 and No. 6 shims and NE4 and SW4 withdrawn; all others inserted	5.1 ± 0.1 \$
End of 50-50-50 power division	All at 160°	All No. 3 shims inserted; NE4 and SW4 at 39 in.; all others withdrawn	4.5 ± 0.1 \$
All shims withdrawn	All at 160°	All withdrawn	1.7 ± 0.05 \$
Total			11.3 ± 0.2 \$

[a] Error limits are estimates of one standard deviation based primarily on uncertainties in the shim calibrations.

The excess reactivity held down by the movable shims (or by the boron strips added to the core when the shims were withdrawn) was also calculated using one-group perturbation theory. The change in the macroscopic absorption cross section of the core can be related to reactivity by the approximate equation^[a]

$$\rho(\text{dollars}) = \frac{1}{\beta} \left(\frac{\Delta\Sigma}{\Sigma + \frac{1}{2} \Delta\Sigma} \right)$$

where Σ is the macroscopic neutron absorption cross section of the active core (ie, the fuel elements), $\Delta\Sigma$ is the change in the cross section due to the addition of the borated plastic strips, and β is the effective delayed neutron fraction, assumed to be 0.0075 in this calculation. This method of calculating excess reactivity gives consistently greater values than the aforementioned method by 8 percent. Macroscopic cross sections for an ATR-ATRC fuel element are given in Appendix A.

3. FUEL ELEMENT BORON CONTENT

In order to provide a check on the boron content of each ATR fuel element, a nondestructive assay is performed in the ATRC on individual fuel elements. The boron contents are determined by relating the reactivity effect of each element in a standard fuel element position to a reactivity-effect versus boron-content calibration curve for that fuel element position. The calibration curve is constructed from reactivity measurements with several different precisely known amounts of boron in plastic strips inserted in a special nonborated fuel element. The borated plastic strips are the same length as the fuel plates (48 inches) and are uniformly distributed in the water channels in order to simulate the boron distribution in a standard borated element. This assaying technique presumes that the U-235 content of all fuel elements is accurately known (provided by the fabricator) so that any deviations from the special nonborated standard are known and can be corrected for.

To date, forty-eight 875-g (U-235) fuel elements and eighty-five 975-g fuel elements have been assayed. The natural boron content of these elements ranges from 3.80 to 4.40 g per element with an average of 4.12 g. The standard deviation of the distribution is about 0.1 g; whereas, the error in a measurement is estimated to be somewhat less, about 0.05 g for one standard deviation.

[a] The derivation of this equation and a discussion of its application are given in Reference 21.

VI. VOID REACTIVITY EFFECTS

The reactivity effects associated with the introduction of voids in the water moderator and coolant of the ATR were of particular interest in the preparation of the Safety Analysis Report for the ATR[7]. Consequently, these reactivity effects were studied rather extensively in ATRC, particularly those associated with voiding within a flux trap since they formed the basis for studies of a maximum credible accident. In these measurements in ATRC, Teflon strips, solid aluminum pieces, and air-filled cylinders were used to simulate voids. However, in all cases a measured conversion factor was applied to the data so obtained to relate it to an equivalent air void.

1. FLUX TRAP VOIDS

The reactivity effects of voiding an ATRC flux trap were measured with the 35-kg fuel loading described earlier in this report. These measurements were not repeated with the present 39-kg fuel loading (except during the outer shim reversal program described in Chapter IX) because, as will be subsequently discussed, all values are expected to be less than or equal to the results from the 35-kg core. The majority of the measurements were made for an inner flux trap containing a safety rod guide tube but no flux trap filler. This flux trap and composition were selected on the basis of a PDQ-4 prediction[22] that these would provide the largest effects. Throughout these measurements, the metal-to-water ratio within the experiment flow tube was held at 3.05 with three different fuel loadings (80.3, 26.8, and 0.0 g U-235).

Preliminary experimental results had indicated that the reactivity effect associated with a given void was strongly dependent upon the shim configuration. Consequently, these experiments were performed with three typical but diverse shim configurations: (a) all outer shims inserted, and the 18 outermost neck shims withdrawn; (b) all outer shims at approximately 80°, and all neck shims inserted; and (c) all shims in a configuration which produces a 40-50-60 power division with the high-power lobe containing the flux trap of interest. For all three flux trap fuel loadings, the greatest reactivity changes due to voiding occurred with the third shim configuration.

The results of this experimental program[23] are shown in Figures 12, 13, and 14. Each figure shows, for a particular shim configuration, the reactivity effect associated with various void fractions for each of the three fuel loadings in the flow tube. In all three figures, the curves shown are nearly linear from zero to approximately 60 percent void. Apparently, the water for void fractions less than 60 percent acts only as a neutron poison and is not providing significant fast neutron moderation. For larger void fractions, the removal of this neutron absorber begins to be compensated by the associated removal of moderation. As would be expected, the importance of the water as a moderator varies directly with the number of fast neutrons produced in the flux trap and, consequently, with the amount of fuel in the flow tube.

Even with no fuel in the flow tube, the curves shown in the figures tend to rise less steeply as 100 percent void is approached. This effect arises because the void simulations used for the larger voids increased the proximity of the void to the fuel annulus. Consequently, the reactivity effects measured

Fig. 12 Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers. Shims are in a configuration for a 40-50-60 power division.

for the larger voids included progressively more of the effect caused by removing moderator important to the fission process in the fuel annulus. Increasing the fuel loading from 35 to 39 kg will make this moderator even more important and will cause these curves to bend over more for a particular void fraction than that shown. Thus, the flux trap void effects for the 39-kg core should be somewhat reduced in magnitude from those shown.

As stated above, these void measurements were made in an inner flux trap since a PDQ-4 simulation predicted they would be the largest in the core. A measurement was made to confirm that prediction by placing the maximum void considered (88 percent) in an outer flux trap. The results of these measurements, as given in Table IV, show that the effect of a void in the high-power lobe of a nominally 40-50-60 power division was indeed larger than the value obtained in the outer flux trap for either a 40-50-60 or a 50-50-50 power division. However, the void effect in the outer flux trap was 10 percent greater than in an inner flux trap for a 50-50-50 power division. This agrees very well with the observation reported in Chapter II that, with a 50-50-50 power division, a safety rod in an outer flux trap is worth 10 percent more than in an inner flux trap.

Fig. 13 Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers. All neck shims are inserted, and all outer shims are at 80 degrees.

Fig. 14 Reactivity effect of voids in an inner flux trap containing a safety rod guide tube without flux trap fillers. All outer shims are inserted, and the 18 outermost neck shims are withdrawn.

TABLE IV

REACTIVITY EFFECT OF REMOVING 88
PERCENT OF THE WATER FROM A FLUX TRAP^[a]

Power Division (shim configuration)	Reactivity Effects (\$)	
	Inner Flux Trap (NE)	Outer Flux Trap (E)
40-50-60 (outer shims N34, E12, S34, and W12 at 100°; all number 5 and 6 neck shims out; all other shims in- serted)	+1.37	+1.15
50-50-50 (all outer shims at 80°; all neck shims inserted)	+0.94	+1.04

[a] Flux trap composition: aluminum rods but no fuel in the flow tube; flux trap annulus metal-to-water ratio within 5 percent of that for a safety rod guide tube with no flux trap filler.

All void effects reported above are for a flux trap whose flow tube has a metal-to-water ratio of 3.05. If the metal inside the flow tube is removed (zero metal-to-water ratio) and the entire tube voided as part of the experiments reported in Figures 12, 13, and 14 and Table IV, an additional +0.17 \$ in reactivity will be obtained. This reactivity should be added to the effects reported for no fuel in the flow tube.

2. FUEL ELEMENT VOIDS

In addition to the reactivity effects of voiding an ATRC flux trap reported above, the reactivity effects of localized void formation within the coolant channels of the ATRC fuel annulus have been measured. These measurements have been concentrated in fuel element positions 12, 19, 32, and 39 (see core diagram in Figure 1), for it is in these elements that boiling would first be expected during an accident with the core in a 40-50-60 power division.

Preliminary measurements^[24] in the 35-kg core indicated that the void effects in the fuel element positions given above in a 40-50-60 power division were more negative than in a 50-50-50 power division. Thus, in subsequent measurements^[25] in the 39-kg core, conservative data were obtained in these elements by using only the 50-50-50 power division. The measurements were made radially along the azimuthal center of the elements and also radially filling one-half channels with Teflon to obtain channel-averaged void effects. The measured reactivity effects converted to void coefficients are shown in Figure 15. The shape of the curves shown in this figure are in reasonable agreement with those obtained in the 35-kg core. However, the measured value at each point is in every case more negative than the value obtained at that point with the 35-kg core, indicating that the increased fuel loading increased the

importance of the water coolant as a moderator. This effect is consistent with and tends to support the reasoning used to interpret the flux trap void effects, ie, that the increased fuel loading would increase the importance of the water on the periphery of the flux trap and, thus, reduce the magnitude of the positive reactivity addition associated with a given void in the 39-kg core. In all cases the measured reactivities associated with voids in the fuel annulus were more negative than had been estimated during the safety analyses for ATR and ATRC, and thus make the pertinent analyses quite conservative.

Fig. 15 Void coefficient as a function of radial position in fuel element positions 12, 19, 32, and 39 in a 50-50-50 power division in a 39-kg core.

VII. EXPERIMENT REACTIVITY EFFECTS

During the study of various core conditions used to date, the reactivity effects associated with the insertion of various materials into the different core regions have been measured. A summary of values obtained is presented in Table V. These values are for a cold, clean reactor having a composition

TABLE V
REACTIVITY EFFECTS IN A 50-50-50 POWER DIVISION

Core Position and Component	Material Inserted Uniformly over Entire Core Length	Reactivity Effect of Insertion Including Displacing Water
Flux Trap:		
Dummy flux trap filler	Aluminum	+0.55 \$
Guide tube mock-up	Aluminum	+0.18 \$
Pressure tube	Stainless Steel	-0.51 \$
Dummy pressure tube	Stainless Steel	-0.41 \$
Experiment (within flow tube)	Aluminum	$+3.6 \times 10^{-3}$ \$/cu.in.
	Zirconium	$+3.6 \times 10^{-3}$ \$/cu.in.
	U-235	$+4.5 \times 10^{-3}$ \$/g
Experiment shroud:		
(1) With flux trap filler:		
(a) With 36 g U-235 in experiment	Hafnium	-0.47 \$
(b) Without U-235 in experiment	Hafnium	-0.36 \$
(2) Without flux trap filler:		
(a) With 36 g U-235 in experiment	Hafnium	-0.30 \$
(b) Without U-235 in experiment	Hafnium	-0.20 \$
I-Hole:		
Large	Aluminum	-0.008 \$/filler
Small	Aluminum	+0.0008 \$/filler
A-Hole	Stainless Steel	-0.0042 \$/cu.in.
H-Hole:		
Filler piece	Aluminum	+0.03 \$/rod
ATR fixed shim	Hafnium	-0.25 \$/rod
B-Hole	Stainless Steel	-0.01 \$/cu.in.
Fuel Annulus:		
Fuel (core average)	U-235	+0.001 \$/g
Burnable poison (core average)	Natural Boron	-0.067 \$/g

as described in Table I for the 39-kg core and operated in a nominally 50-50-50 power division. When these values are used to estimate reactivity effects, they are treated as guides, accurate to ± 10 percent for any particular experiment change in a 50-50-50 power division.

The values listed in Table V, particularly those for compositional changes within flux traps, are sensitive to changes in the reactor power distribution. Good results have been achieved in estimating flux trap effects by correcting these values on the basis of the change in the square of the average power in the adjacent fuel elements. The values for changes outside the flux traps are not as sensitive to the power changes but are corrected when local power values are available.

When the water temperature within a flux trap flow tube is increased, the resulting reactivity effects depend on the materials inside. With only water in the flow tube, the measured reactivity effects are as shown in Figure 16. For aluminum and zirconium, the reactivity changes can be estimated from the density change. For hafnium, spectrum hardening may be a factor. With a flux trap filler surrounding the pressure tube, the reactivity worth of a 100 percent hafnium shroud increased about 10 percent when the loop water was heated from 70 to 500°F. However, it is believed that if the pressure tube were surrounded by a guide tube only, the hafnium would increase in worth by 40 to 60 percent as the loop water is heated from 70 to 500°F.

The reactivity worth of the hafnium H-hole inserts (also called fixed shims) varies as the outer shim and neck shim positions vary. Since their worth is estimated to change in the same manner as the neck shims, the remarks in Chapter IV regarding neck shims will pertain. These rods have been observed to have a maximum value of 0.35 \$ per rod and a minimum value of 0.20 \$ per rod.

Fig. 16 Reactivity effect of temperature change in the ATRC Hot Loop with no experiment.

VIII. NEUTRON FLUX AND POWER DISTRIBUTIONS

One of the more significant features of the ATR is that the neutron flux level in the in-pile tube of a given flux trap may be adjusted without adversely affecting the other experimental loops in the reactor. This is accomplished using a combination of outer shims and neck shims (mostly the former) to adjust the power density in the eight fuel elements surrounding the given flux trap and in-pile tube. It is important, therefore, that the neutron flux and power distributions throughout the experiment positions and fuel annulus be measured in considerable detail with several shim configurations. Most of these measurements in the ATRC are made using quarter-inch, 40-mil, uranium-aluminum wires as fission-rate monitors. (A detailed description of power calibration techniques in the ATRC is given in Reference 26.)

1. INITIAL MEASUREMENTS

Early measurements of the power distribution in the ATRC were made in the 35-kg core with the shims positioned to give a 40-50-60 power division[5]; ie, the power generated in the NE and SW lobes is proportional to 40 MW in each lobe, the power in the center lobe is proportional to 50 MW, and the power in the NW and SE lobes is proportional to 60 MW in each lobe[a]. Additional power distribution and thermal flux measurements were made with the shims in a configuration to give a 50-50-50 power division[27], ie, all five lobes generating the same power.

After the 35-kg core was replaced with the 39-kg core (see Chapter I), more power distribution measurements were made (and in greater detail) to determine the nominally 50-50-50 and 40-50-60 power divisions, both with and without simulated xenon poisoning. The core conditions for these measurements are those listed in Table I for the 39-kg core.

2. 50-50-50 POWER DIVISION

Most of the ATRC power distribution measurements to date have been made with the reactor in a 50-50-50 power division[28, 29], a particularly convenient reference distribution to which all other power distributions may be compared for evaluation. Since all flux traps (except the center, of course) have nearly the same compositions, a nominally 50-50-50 power division is obtained with all outer shims withdrawn the same amount, as shown in Figure 17. The power distribution measured in this nominally 50-50-50 power division is depicted in the core diagram of Figure 18. This diagram shows that there is actually a slight tilt in the power distribution. The power in the SW lobe is about seven percent greater than the power in the NE lobe, whereas the other two lobes (NW and SE) agree rather closely. This asymmetry is actually quite small, and if it were not so consistent it might well be neglected. Although considerable effort has been expended to resolve this unexpected asymmetry[30], no cause has yet been determined.

[a] A lobe is defined as the eight fuel elements surrounding an inner flux trap.

Fig. 17 Cross-sectional diagram of ATRC core showing shim configuration for a nominally 50-50-50 power division.

The power generated in the eight fuel elements comprising the center lobe was slightly greater than in the other lobes. This difference between the center and outer lobes could be decreased by inserting additional fixed shims into the H-holes. Also, finer (and unbalanced) adjustment of the outer shims would lead to a more even power distribution for the outer lobes.

The fission rate in the fuel elements is cusp shaped in both radial and azimuthal directions because of increased moderation and reduced absorption around the edges of the fuel. This peaking is typically most severe in plate 19

Fig. 18 Cross-sectional diagram of ATRC core showing the power distribution for a nominally 50-50-50 power division normalized to 250 MW. The number in each fuel element is the element power in MW. The number in each flux trap is the lobe power in MW. All outer shims are at 78.5 degrees, and both NE4 and SW4 neck shims are withdrawn.

and only slightly less in plates 1, 2, and 18 (see Figure 19 for a cross-sectional view of an ATR-ATRC fuel element). To illustrate the extent of this peaking in plate 19, peaking factors (peak-to-core average fission-rate ratios) for several power divisions are included in this report. These peaking factors are obtained from fission-rate measurements in water channel 19 and are extrapolated into plate 19 by increasing them seven percent (see Section 4 below).

Peaking factors and element powers for a typical 50-MW lobe (SW) in the nominally 50-50-50 power division are shown in Figure 20. The peaking is most severe in fuel position 28 at the 27-28 interface where the specific power at the nominal edge of the fuel in plate 19 would be 19.9 kW/g U-235 (peaking factor, 3.11) at 250 MW. The estimated

Fig. 19 Cross-sectional diagram of the ATR-ATRC fuel element.

Fig. 20 Peaking factors and element powers in a 50-MW lobe (SW) with and without simulated xenon poisoning. The numbers in the fuel elements are element powers in MW.

error (one standard deviation) for the peaking factors in positions 27 and 28 is four percent. In positions 26 and 29 the fission-rate mapping to obtain the peaking factors lacked detail, resulting in an estimated standard deviation of seven percent.

In order to assess the effect of the equilibrium xenon poison in the ATR, boron was uniformly added to all fuel elements by the use of boron impregnated plastic strips[a]. The power distribution normalized to 250 MW in a quarter core with this simulated poisoning is shown in Figure 21. (Since each quarter of the core is nearly a mirror image of the adjacent quarter, ie, so-called quarter-core symmetry, the power distribution in only one quarter of the core is usually mapped in any detail.) Peaking factors and element powers for a typical 50-MW lobe (SW) are shown in Figure 20. For the reason given in the paragraph above, the estimated

[a] Approximately 1.85 g natural boron per fuel element for a total reactivity effect of approximately -5 \$.

Fig. 21 Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with simulated xenon poisoning. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims NE4, SW4, all number 1's, and all number 6's are withdrawn; all other neck shims are inserted; outer shims, N12, E34, S12, and W34 are at 111.1 degrees, and N34, E12, S34, and W12 are at 103.2 degrees.

error (1σ) in the peaking factors in position 28 is four percent, and that in positions 26, 27, and 29 is seven percent.

It may be noted from Figure 20 that, with simulated xenon poisoning, each fuel element of the lobe generates a more nearly equal share of the lobe power. This change in the power distribution is the result of the shim movement to compensate for the addition of the poison.

3. 40-50-60 POWER DIVISION

The 40-50-60 power division has particular significance since, in the original ATR design, two high-power (60-MW) lobes and two low-power (40-MW) lobes were postulated to meet design specifications for some of the Navy in-pile loops. The quarter-core diagram in Figure 22 shows the power distribution measured in a nominally 40-50-60 power division in the ATRC. Peaking factors and element powers for a typical 60-MW lobe (NW) are shown in Figure 23. The peaking factors in those elements of the inner half of the lobe are all large,

Fig. 22 Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division. The numbers in the flux trap are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims NE4, SW4, and all number 6's are withdrawn; all other neck shims are inserted; outer shims N12, E34, S12, and W34 are at 108.0 degrees, and N34, E12, S34, and W12 are at 25.0 degrees.

with the largest value occurring in fuel position 33 at the 33-34 interface. The specific power at the nominal edge of the fuel in plate 19 would be 23.1 kW/g U-235 (peaking factor, 3.61) at 250 MW. The estimated error (1σ) for the peaking factors is four percent.

The power distribution in a 40-50-60 power division also was measured with boron uniformly added to simulate equilibrium xenon poisoning in the ATR. The resulting quarter-core power distribution normalized to 250 MW is shown in Figure 24. The peaking factors and element powers for a typical 60-MW lobe (NW) are shown in Figure 23. The change in shim positions to compensate for the added poison leads to a more nearly equal power distribution among the fuel elements in the high-power lobes. As a result, there will apparently be a slight reduction in the largest peaking factors in the ATR during the buildup of the xenon poisoning.

In order to determine the effect of the flux trap filler on the power distribution in the high-power lobe, the above measurement was repeated with the

Fig. 23 Peaking factors and element powers in a 60-MW lobe (NW) with and without simulated xenon poisoning. The numbers in the fuel elements are element powers in MW.

power division rotated 90° about the center flux trap. With this rotation, the NE and SW lobes containing aluminum flux trap fillers became high-power (60-MW) lobes, and the NW and SE lobes containing water in the flux trap annuli became low-power (40-MW) lobes. The resulting power distribution normalized to 250 MW is shown in Figure 25. These data may be reflected about an east-west axis for comparison purposes since the core is nearly symmetrical. With this reflection, the peaking factors and element powers shown in Figure 23 were obtained for the high-power lobe with an aluminum flux trap filler.

Fig. 24 Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with simulated xenon poisoning. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. All number 1, 4, 5, and 6 neck shims are withdrawn; all other neck shims are inserted; outer shims N12, E34, S12, and W34 are at 160.0 degrees, and N34, E12, S34, and W12 are at 46.2 degrees.

The use of an aluminum flux trap filler results in two apparent improvements in the high-power lobe. First, the peaking in those fuel plates most nearly adjacent to the flux-trap-filler region is reduced. In the case of plate 19 position 32, this reduction is about 12 percent. Second, the positive effect of the aluminum filler relative to water plus the increased worth of the adjacent outer shims (see Section 4 in Chapter IV) results in additional excess reactivity in the shims around the high-power lobes. Although the amount of movable reactivity remaining around these high-power lobes is not large, it will increase the charge life.

4. FISSION-RATE PROFILES IN FUEL ELEMENTS

In order to obtain a reasonably accurate picture of the power distribution throughout the fuel annulus of the ATRC, detailed measurements are made of the azimuthal, radial, and axial fission-rate profiles in the fuel elements. Combinations of these data are used to calculate fuel element power as well as peaking factors.

Fig. 25 Quarter-core power distribution for a nominally 60-50-40 power division with simulated xenon poisoning and with a flux trap filler in the 60-MW lobe. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. All number 4, 5, and 6 neck shims are withdrawn; all other neck shims are inserted; outer shims N34, E12, S34, and W12 are at 155.0 degrees, and N12, E34, S12, and W34 are at 54.2 degrees.

The azimuthal and radial fission rate profiles are generally obtained at the reactor horizontal midplane. Typical azimuthal profiles in water channel 19 for 50-50-50 and 40-50-60 power divisions, both with and without simulated xenon poisoning, are shown in Figures 26 and 27. These profiles are used to pinpoint the hot-spot fission rate (which usually occurs in channel 19); consequently, they were obtained with somewhat more detail than the profiles in the other channels. The peaking factors shown in Figures 20 and 23 were obtained from these azimuthal profiles. In order to illustrate the midplane fission-rate distribution more fully, radial profiles at the fuel element azimuthal centers are shown in Figures 28 and 29. (Internormalization of all radial and azimuthal profiles is based on lobe power.)

The peaking factors quoted in this report apply in each case to the nominal edge of the fuel in plate 19. The estimated errors in the peaking factors reflect the precision with which the fission rate or specific power can be measured at that point and does not include the error associated with the uncertainty in the location of the edge of the fuel.

Fig. 26 Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 26, 27, 28, and 29 in a nominally 50-50-50 power division with and without simulated xenon poisoning.

Fig. 27 Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 32, 33, 34, and 35 in a nominally 40-50-60 power division with and without simulated xenon poisoning.

In addition to the fission-rate profiles at the horizontal midplane, several azimuthal fission-rate profiles were measured at the same levels as the sideplate vents 1.75 inch above and 2.25 inches below the reactor midplane to determine the extent of the peaking at the sideplate vents. The results of these measurements in channel 19 of fuel position 28 indicate that the hot-spot fission rate in the 50-50-50 power division is about 10 percent higher than the midplane value where channel 19 is vented (1.75 inch above midplane) and about

Fig. 28 Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 26, 27, 28, and 29 in a nominally 50-50-50 power division with and without simulated xenon poisoning.

Fig. 29 Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 32, 33, 34, and 35 in a nominally 40-50-60 power division with and without simulated xenon poisoning.

8 percent higher where adjacent channels are vented (2.25 inches below midplane). The azimuthal fission-rate profiles at these three elevations are compared in Figure 30.

A typical axial fission-rate distribution near the center of a fuel element is shown in Figure 31. The solid curve is a least-squares computer fit to the data. This distribution also shows peaking near both ends of the fuel element due to the water reflector. Since this peaking occurs so near the end of the element, it is neglected in the core power calculation. The shape of the axial fission-rate profile is a chopped cosine whose peak-to-average ratio is 1.42. (Although Figure 31 shows 1.44, the average over the core is 1.42.)

The axial fission-rate profile near the fuel element sideplate is perturbed slightly due to the sideplate vents. The effects of these vents were originally investigated in the 35-kg core[31]. The results of a more recent measurement

Fig. 30 Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel position 28 showing the effect of the sideplate vents on the fission-rate peaking.

in a 39-kg core are shown in Figure 32. This axial profile was obtained in water channel 11 about 0.1 inch from the nominal edge of the fuel in the adjacent fuel plates.

A number of detailed measurements were made to determine the fission-rate depression in the fuel plates of an ATRC fuel element [32]. The data obtained are shown in Figure 33. Due to the error in these measurements, it is virtually impossible to ascribe a correction factor for fission-rate depression in the fuel plates. However, the data show that care must be exercised in applying fission-rate values measured in the water channels to the values in the adjacent fuel plates. From Figure 33 it is clear that one could underestimate the fission rate in fuel plates 1 or 19 by applying values routinely obtained in water channels 2 and 19, respectively, to these fuel plates. Therefore, in determining the core hot spot, the channel 19 fission rate is increased by seven percent. Although these data were obtained in the 35-kg core (875-g fuel elements), they are assumed to be equally applicable in the 39-kg core (975-g elements).

5. THERMAL NEUTRON FLUX IN EXPERIMENT HOLES

The results of the thermal neutron flux and fission-rate measurements in the experiment irradiation holes are summarized in Table VI. These values were obtained in a power distribution proportional to that shown in Figure 18 and, therefore, reflect the slight asymmetry of this power distribution. The thermal neutron flux values in the flux traps can be normalized to 50-MW lobe

Fig. 31 Axial fission-rate profile near the center of fuel position 28.

Fig. 32 Axial fission-rate profile near the edge of water channel 11 in fuel position 13 showing the sideplate vent perturbations.

powers to obtain $8.6, 8.8,$ and 8.5×10^{14} n/cm²-sec for the C, SW, and NW flux traps, respectively, instead of the flux values shown in the table. The differences in these three numbers reflect the differences in the material compositions of the three flux traps as indicated for the 39-kg core in Table I. In a flux trap containing water only (ie, flowtube, pressure tube, and flux trap filler removed), the peak thermal flux will nearly double. The estimated error (one standard deviation) for the values shown in Table VI is three percent.

Thermal neutron flux measurements also have been obtained in a typical flux trap with various experiment loadings. The results of these measurements are summarized in Table VII. The KAPL mock-up used in these measurements consisted of a nonfueled experiment train with aluminum strips in aluminum holders and was intended to represent a typical experiment train without fuel. The relatively high metal-to-water ratio (3.68) in the flow tube of this flux trap accounts for the much lower thermal flux values compared with the thermal flux values in the flux traps in Table VI, which were obtained with no metal whatsoever in the flow tubes.

Typical axial fission-rate distributions measured in several experiment holes are shown in Figures 34 through 36. The curves shown in these figures are least-square computer fits of the data points. It should be noted that the peak-to-average ratios of these axial fission-rate profiles are nearly the same as the average value in the fuel elements, 1.42.

Fig. 33 Radial fission-rate profile through plates 1, 6, 11, 15, and 19 in fuel position 13.

The axial fission-rate distribution shown in Figure 37 illustrates the effects of the lower end of a withdrawn ATRC safety rod on the fission rate along the axis of an experiment loop[27]. During normal operation of both the ATR and ATRC, the lower ends of the safety rod neutron absorber plates extend three inches into the reactor core. As a result, the fission rate at the top of the core is reduced slightly, but the depression extends no more than five or six inches into the core.

6. FAST NEUTRON FLUX

Before the 35-kg core was replaced with the 39-kg core, the fast (> 1 MeV) neutron flux was measured in several of the experiment irradiation holes[33]. The fast neutron flux at the reactor midplane is shown normalized to 250 MW in Figure 38. Typical axial distributions obtained in the reflector (B1) and in the center flux trap baffle (H15) are shown in Figure 39. The data were obtained with the reactor operating in a nominally 40-50-60 power division with the core conditions listed in Table I for the 35-kg core. The fast flux values have an estimated standard deviation of 20 percent, mostly due to the very low activities of the zinc threshold detectors even at the high-power level (for ATRC) of 1000 watts. Increasing the U-235 loading of the core from 35 to 39 kg would not be expected to alter these fast flux values significantly since the fast flux is nearly proportional to fission rate and is insensitive to small changes in the thermal absorption cross section.

TABLE VI
HORIZONTAL MIDPLANE FISSION RATE
AND THERMAL NEUTRON FLUX VALUES AT 250 MW

<u>Core Position</u>	<u>Fission Rate (10^{14} fission/g-sec)</u>	<u>Thermal Neutron Flux (10^{14} neutrons/cm²-sec)</u>
C Flux Trap	14.3 ^[a]	9.2 (53.8-MW lobe)
SW Flux Trap	14.3	9.2 (52.1-MW lobe)
NW Flux Trap	12.6	8.1 (47.8-MW lobe)
H-4	7.5	4.7
H-6	7.4	4.6
H-8	7.7	4.9
H-10	7.7	4.8
B-2	11.7	[b]
B-3	11.2	[b]
B-4	10.9	[b]
B-5	11.1	[b]
A-10	5.3	[b]
A-11	4.9	[b]
I-4 ^[c]	1.0	0.6
I-5	1.1	0.7
I-6	2.4	1.6
I-7	1.1	0.7
I-8	0.8	0.5
I-9	0.9	0.5
I-10	1.1	0.7
I-11	2.3	1.5
I-12	1.1	0.7
I-13	1.0	0.6

[a] Estimated value.

[b] No values available due to lack of space for cadmium ratio measurement.

[c] All I-hole values are at the edge of the hole nearest reactor center.

TABLE VII

FLUX TRAP FISSION RATES AT REACTOR HORIZONTAL MIDPLANE
(Normalized to 50-MW lobe power)

Flux Trap	Flux Trap Composition[a]	Fission Rate (10^{14} fission/g-sec)	Thermal Neutron Flux (10^{14} neutrons/cm ² -sec)
NE	Dummy flux trap filler, stainless steel retainer tube, and KAPL mock-up.	5.4	3.3
NE	Dummy flux trap filler, HF retainer tube, and KAPL mock-up.	2.0	1.15
NE	Safety rod guide tube, HF retainer tube, and KAPL mock-up.	2.3	1.4
NE	Safety rod guide tube, stainless steel retainer tube, and KAPL mock-up.	6.2	3.9
E	Dummy flux trap filler, stainless steel retainer tube, and KAPL mock-up.	6.0	3.75
C	Stainless steel retainer tube and KAPL mock-up.	5.05	3.1

[a] In all cases the flux trap contained a standard NR pressure tube mock-up.

Fig. 34 Axial fission-rate profile in center flux trap baffle position H4.

Fig. 35 Axial fission-rate profile in reflector position I4.

Fig. 36 Axial fission-rate profile in the NW flux trap.

Fig. 37 Axial fission-rate profiles in the NE and SW flux traps showing the effect of a withdrawn safety rod.

Fig. 38 Cross-sectional diagram of ATRC core showing fast (> 1 MeV) neutron flux at reactor horizontal midplane for a nominally 40-50-60 power division normalized to a reactor power of 250 MW.

Fig. 39 Axial fast-neutron flux profiles in B1 and H15.

IX. EFFECTS OF REVERSING OUTER SHIM ROTATION

1. INITIAL INVESTIGATIONS

From the power distribution and fission-rate measurements described in the previous chapter, it seemed probable that the peak fission rate in the ATR would limit the operating power level to something less than its 250-MW design power. In considering possible ways to surmount this problem, it appeared that simply reversing the direction of withdrawal of the outer shim cylinders would reduce the fission-rate peaking. This approach was initially investigated in the ATRC by reversing the withdrawal direction of the two outer shim pairs around the SW lobe (W12 and S34) and then obtaining detailed fission-rate data in the fuel elements in this lobe. These data were so encouraging that all ATRC outer shim drives were then modified to withdraw in this reverse direction, and additional fission-rate data as well as information on the reactor parameters and operating characteristics were obtained.

In the initial experiment it was determined that the position of the zero index for the outer shim pairs was not at a reactivity null^[a]. Additional experiments with shim cylinders W1 and S3 established that the outboard shim (W1) was at a reactivity null but that the inboard shim (S3) had to be rotated 35 degrees toward the center of the reactor to reach a true zero or reactivity null. Consequently, during the changes made to reverse the direction of withdrawal, the inboard shim in each pair was rotated 35 degrees toward the core center. Since the outer shim calibration curve is quite flat between 0 and 35 degrees, the reactivity effect of this change was relatively small, ie, about 0.26 \$ for all eight inboard cylinders.

2. EFFECT OF OUTER SHIM REVERSAL ON POWER DISTRIBUTION

Since the outer shim reversal was expected to have a significant effect on the power distribution and fission-rate peaking, comprehensive fission-rate data were obtained. Both the nominally 50-50-50 and 40-50-60 power divisions with and without simulated xenon poisoning were investigated. Fission-rate peaking factors in the fuel elements for both power distributions were also obtained for comparison with earlier data. These outer shim reversal data show a significant reduction in the fission-rate peaking in the nominally 50-50-50 power division. There is less improvement in the 40-50-60 power division, however, since the outer shims around the high-power lobes are almost fully withdrawn regardless of the direction of withdrawal.

2.1 50-50-50 Power Division

The quarter-core power distribution for the cold, clean core in the nominally 50-50-50 power division (normalized to 250 MW) is shown in Figure 40. The effect of the shim reversal, as is evident from a comparison of Figures 40 and

[a] When an outer shim cylinder is in its zero position (inner limit), a straight line between the center of the cylinder and the center of the adjacent inner flux trap bisects the arc of the hafnium plate on the cylinder.

Fig. 40 Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with the outer shims reversed. The numbers shown in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims NE4 and SW4 are withdrawn; all other neck shims are inserted; all outer shims are at 80.8 degrees.

18, is to increase the power generated in the outer portion of the corner lobes and to decrease the power generated in the inner half of these lobes. This more uniform power distribution around the lobes is the result of redistributing the outer-shim poison by reversing the outer shims. This new outer-shim poison configuration is shown in Figure 41 along with the original configuration.

Peaking factors and element powers for a 50-MW lobe (SW) are compared with the data from the previous chapter (ie, before outer shim reversal) in Figure 42. The comparison shows that the smoothing of the power distribution has resulted in a 15 percent reduction of the peaking factor in fuel position 28.

The power distribution in the 50-50-50 power division was remeasured with simulated xenon poisoning to obtain data applicable to the ATR after xenon equilibrium is reached. The quarter-core power distribution is shown in Figure 43 normalized to 250 MW and may be compared directly to Figure 21 to observe the change in power generation with outer shim reversal. Peaking factors and element powers for this poisoned core are shown in Figure 42 where

Chapter VIII data have been repeated for comparison purposes. Although the reduction in peaking is not as pronounced as in the cold, clean core, the hot-spot fission rate is still reduced by reversing the outer shims.

Fig. 41 ATRC core diagram showing the shim positions for the nominally 50-50-50 power division before and after outer shim reversal. The outer shims below the diagonal line are for the shims-reversed case, and those above the line are for the normal case. The NE4 and SW4 neck shims (ATR reg rods) were withdrawn in both cases.

50-50-50 Power Division
Cold, Clean Core

Peaking factors

Outer Shims Reversed

Peaking factors

Outer Shims Normal

50-50-50 Power Division
Xe Equilibrium

Peaking factors

Outer Shims Reversed

Peaking factors

Outer Shims Normal

Fig. 42 Peaking factors and element powers in a 50-MW lobe (SW) before and after outer shim reversal. The numbers in the fuel elements are element powers in MW.

Fig. 43 Quarter-core power distribution normalized to 250 MW for a nominally 50-50-50 power division with simulated xenon poisoning and reversed outer shims. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims NE4 and SW4 and all number 1 and 6 neck shims are withdrawn; all other neck shims are inserted; all outer shims are at 108.7 degrees.

2.2 40-50-60 Power Division

The quarter-core power distribution for the cold, clean core in the nominally 40-50-60 power division is shown in Figure 44 normalized to 250 MW. This power distribution may be compared with Figure 22 of Chapter VIII to observe the effect of reversing the outer shims. Peaking factors and element powers for a 60-MW lobe (NW) are shown in Figure 45 along with data from Chapter VIII (ie, before outer shim reversal). It should be noted from this figure that the maximum peaking factor occurs in different positions for the two cases. Before the outer shims were reversed, the core hot spot in a 40-50-60 power division occurred in plate 19 at the 33-34 interface; whereas, with the shims reversed, the core hot spot is in plate 19 of fuel position 32. This change in peaking is the result of smoothing out the power distribution by reversing the outer shims.

Fig. 44 Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with the outer shims reversed. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims NE4 and SW4 and all number 6 neck shims are withdrawn; all other neck shims are inserted; outer shims N13, E34, S12, and W34 are at 108.0 degrees, and N34, E12, S34, and W12 are at 17.8 degrees

The above measurement was repeated with simulated xenon poisoning in order to obtain data applicable to the ATR after xenon equilibrium has been reached. The resulting quarter-core power distribution is shown in Figure 46 normalized to 250 MW and may be compared to Figure 24 to observe the changes in power generation accompanying the outer shim reversal. Peaking factors and element powers for a 60-MW lobe (NW) are shown in Figure 45 along with data from Chapter VIII (ie, before outer shim reversal). In this comparison there is little difference in the power generated in like elements or in the peaking factors except in fuel position 32.

It has already been shown in the previous chapter that the peaking in position 32 was considerably reduced by including an aluminum flux trap filler in the high-power lobe. This reduction due to the presence of the filler was also observed with the outer shims reversed. Figure 47 shows a comparison of a 60-MW lobe with simulated xenon poisoning and an aluminum flux trap filler before and after the outer shims were reversed. Although these data were obtained in the SW lobe, they were reflected about an east-west symmetry line

40-50-60 Power Division
Cold, Clean Core

Outer Shims Reversed

Outer Shims Normal

40-50-60 Power Division
Xe Equilibrium

Outer Shims Reversed

Outer Shims Normal

Fig. 45 Peaking factors and element powers in a 60-MW lobe (NW) before and after outer shim reversal. The numbers in the fuel elements are element powers in MW.

Fig. 46 Quarter-core power distribution normalized to 250 MW for a nominally 40-50-60 power division with simulated xenon poisoning and reversed outer shims. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. All number 1, 4, 5, and 6 nook shims are withdrawn; all other neck shims are inserted; outer shims N12, E34, S12, and W34 are at 155.0 degrees, and N34, E12, S34, and W12 are at 57.3 degrees.

Fig. 47 Peaking factors and element powers in a 60-MW lobe with a flux trap filler before and after outer shim reversal. The numbers in the fuel elements are element powers in MW.

to the NW lobe in order to compare the data with other 60-MW NW-lobe data. (The quarter-core power distribution obtained with the SW lobe operating as the high-power lobe is shown in Figure 48 normalized to 250 MW.)

Fig. 48 Quarter-core power distribution normalized to 250 MW for a nominally 60-50-40 power division with simulated xenon poisoning and reversed outer shims. The numbers in the flux traps are lobe powers in MW, and those in the fuel elements are element powers in MW. Neck shims SW1 and NE1 and all number 4, 5, and 6 neck shims are withdrawn; all other neck shims are inserted; outer shims N34, E12, S34, and W12 are at 160.0 degrees, and N12, E34, S12, and W34 are at 48.3 degrees.

2.3 Fission-Rate Profiles in Fuel Elements

Detailed fission-rate distributions were obtained in water channel 19 of those fuel elements where extreme fission-rate peaking in each power division was expected. The azimuthal fission-rate profiles shown in Figures 49 and 50 were used to obtain the peaking factors shown in Figures 42 and 45. The estimated error in these values is four percent (1σ).

A comparison of Figure 26 with Figure 49 shows little change in the general shapes of the azimuthal profiles (for the nominally 50-50-50 case) in channel 19 with the exception of fuel position 26. In this position, the change in outer shim poison configuration caused the fission-rate profile to tilt in the opposite

Fig. 49 Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 26, 27, 28, and 29 with and without simulated xenon poisoning in a nominally 50-50-50 power division.

Fig. 50 Azimuthal fission-rate profiles at reactor midplane in water channel 19 of fuel positions 32, 33, 34, and 35 with and without simulated xenon poisoning in a nominally 40-50-60 power division.

direction. Further comparison of the fission-rate distribution before and after the shim reversal can be obtained from Figure 28 and Figure 51. These radial fission-rate profiles also indicate very little change in shape as a result of either the shim reversal or the addition of boron for simulated xenon poison except in position 26 where the shape was again changed by the shim reversal. However, the figures show that the fission rate at the azimuthal center of channel 19 in position 28 has been reduced, as is also shown in the peaking factors in Figure 42. Similar conclusions may be reached for the 40-50-60 power divisions

by comparing Figures 27 and 29 with Figures 50 and 52. (Internormalization of all radial and azimuthal profiles is based on lobe power.)

Fig. 51 Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 26, 27, 28, and 29 with and without simulated xenon poisoning in a nominally 50-50-50 power division.

Fig. 52 Radial fission-rate profiles at reactor midplane near azimuthal centers of fuel positions 32, 33, 34, and 35 with and without simulated xenon poisoning in a nominally 40-50-60 power division.

3. OTHER EFFECTS OF OUTER SHIM REVERSAL

In view of the changes in the power distribution resulting from the reversal of the outer shim rotation, a number of additional measurements were made to determine if the reversal had changed the value of parameters important to reactor operation and safety analysis. The parameters measured were the worth of the outer shims as a function of position, the worth of a reg rod (NE4) in a 50-50-50 power division, the flux trap void effect, safety rod shutdown reactivity, and the ratio of the effective delayed neutron fraction to the neutron generation time (β/Λ). The results of these measurements are summarized in Figure 53 and Table VIII.

Shown in Figure 53 are (a) a detailed calibration curve for a single outer shim pair taken before the reversal and (b) a curve for the same pair taken in the same manner but after the reversal. Clearly, the reversal has not significantly altered the shape of the curve except to cause the worth to increase slightly during the early portion of withdrawal. However, within the experimental

Fig. 53 Reactivity calibration curve of outer shim pair N34 before and after reversing outer shims.

error, there has been no change in the reactivity magnitude of the shim in this configuration as evidenced by the values shown in Table VIII. Also, in this table are values for the reg rod worth, which also was found to be the same as before the reversal.

The reactivity effect due to voiding an outer flux trap was found, as shown in Table VIII, to be the same as that measured in the 35-kg core (as described in Chapter VI). No measurement was made in the inner flux traps since

TABLE VIII
REACTIVITY EFFECTS BEFORE AND AFTER OUTER SHIM REVERSAL [a]

Parameter	Value Before Reversal	Value After Reversal	Experimental Technique Used
Outer shim N34	1.41 ± 0.02 \$	1.41 ± 0.02 \$	Reactor period
Outer shim S34	1.54 ± 0.03 \$	1.55 ± 0.02 \$	Reactor period
Reg rod NE4	0.217 ± 0.002 \$	0.216 ± 0.003 \$	Reactor period
88% void in east flux trap	+1.04 \$ [b] +1.15 \$ [b,c]	+1.02 ± 0.05 \$ +1.13 ± 0.05 \$ [c]	Reactor period Reactor period
Safety rods	8.75 ± 0.31 \$	8.50 ± 0.20 \$	Pulsed neutron
β/Λ	142.9 ± 1.3 sec ⁻¹	----	Cross-spectrum noise analysis
β/Λ	139.9 ± 1.0 sec ⁻¹	157.8 ± 3.5 sec ⁻¹	Subcritical pulsed neutron
β/Λ	----	157.2 ± 2.2 sec ⁻¹	Critical pulsed neutron

[a] These values are for a 39-kg core in a 50-50-50 power division unless otherwise stated.

[b] This value was obtained in the 35-kg core.

[c] This value is for a 40-50-60 power division.

any effect of the reversal would be screened by the fuel annulus.

The reversal of the outer shims was found to have increased β/Λ by 10 percent. Most of this change is probably due to a decrease in Λ resulting from a change in neutron leakage. As also shown in Table VIII, the reactivity worths of the safety rods are seemingly unchanged by the outer shim reversal. This is consistent with the results of the flux trap void effect measurements.

X. REFERENCES

1. E. E. Burdick and J. W. Henscheid (eds.) Advanced Test Reactor Critical Facility Safety Analysis Report, IDO-16950-1 (July 1964).
2. N. C. Kaufman and J. W. Henscheid, "Initial ATRC Experiments", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1964, IDO-17042 (November 1964) pp 7-10.
3. J. W. Henscheid and W. L. Schrader, "ATRC Start-up with Permanent Neck-Shim Housing", Nuclear Technology Branches Quarterly Report, July 1-September 30, 1964, IDO-17052 (February 1965) pp 12-13.
4. R. A. Turner, Advanced Test Reactor Turbo Report, IDO-24459 (January 1964).
5. L. C. Richardson and J. W. Henscheid, "ATRC Power Calibration", Nuclear Technology Branches Quarterly Report, October 1-December 31, 1964, IDO-17081 (July 1965) pp 2-5.
6. J. W. Henscheid and W. D. Gregory, A Physical Comparison of the ATR and ATRC Cores, IDO-17150 (March 1966).
7. D. R. deBoisblanc and S. Cohen, Safety Analysis Report, Advanced Test Reactor, IDO-17021 (Vol. 1 and 2) (April 1965).
8. N. C. Kaufman and W. D. Gregory, "ATRC Hot Loop", Nuclear Technology Branches Quarterly Report, October 1-December 31, 1965, IDO-17191 (June 1966) pp 3-6.
9. N. C. Kaufman, "ATRC Safety Rod Worths Using the Pulsed-Neutron Technique", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202 (November 1966) pp 6-8.
10. N. C. Kaufman, "ATRC Safety Rod Worths from the Rod-Drop Method", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202 (November 1966) pp 1-4.
11. W. L. Schrader and J. W. Henscheid, "ATR-ATRC Safety Rod Worths", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1965, IDO-17140 (March 1966) pp 6-7.
12. N. C. Kaufman and R. P. Morrell, "Measurements of β/Λ in ATRC Using Reactor Noise Analysis", Nuclear Technology Branches Quarterly Report, July 1-September 31, 1966 IN-1048, p-1.
13. N. C. Kaufman, "Critical Facilities Pulsed-Neutron Experimental Procedure and Apparatus", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202 (November 1966) pp 4-6.
14. E. Garelis and J. L. Russell, "Theory of Pulsed Neutron Source Measurements", Nucl. Sci. Eng., 16 (July 1963) pp 263-270.

15. H. M. Jones, R. J. Neuhold, and R. A. Turner, Nuclear Analysis of the Advanced Test Reactor Critical Experiments, IDO-24458 (December 1963) pp 25-32.
16. R. P. Morrell, "Experimental Measurement of β/ℓ for the ATRC Critical Facility" Nuclear Technology Branches Quarterly Report, April 1-June 30, 1964, IDO-17042 (November 1964) p 56.
17. N. C. Kaufman, "ATRC Outer Shim and Regulating Rod Calibrations", Nuclear Technology Branches Quarterly Report, January 1-March 31, 1966, IDO-17192 (August 1966) pp 4-6.
18. N. C. Kaufman, "Interaction Between Outer Shims and Inner Flux Trap in ATRC", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1965, IDO-17140 (March 1966) pp 5-6.
19. N. C. Kaufman, "Experimental and Calculated Reactivity Distribution for the ATRC", Nuclear Technology Branches Quarterly Report, October 1-December 31, 1964, IDO-17081 (July 1965) p 1.
20. W. L. Schrader and J. W. Henscheid, "Excess Reactivity in ATR-ATRC 39-kg Core", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202 (November 1966) pp 8-10.
21. J. B. Willis, "Reactivity Determinations in the ETRC by the Distributed Poison Method", MTR-ETR Technical Branches Quarterly Report, January 1-March 31, 1961, IDO-16695 (June 1961) pp 1-5.
22. F. M. Alcorn, R. A. Mickle, and C. B. Woodhall, Advanced Test Reactor Temperature and Void Coefficients of Reactivity, IDO-24455 (March 1964).
23. N. C. Kaufman, "Reactivity Effects of Voiding an ATRC Flux Trap", Nuclear Technology Branches Quarterly Report, January 1-March 31, 1965, IDO-17104 (December 1965) pp 3-6.
24. N. C. Kaufman "ATRC Fuel Annulus Void Coefficients of Reactivity", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1965, IDO-17140 (March 1966) pp 4-5.
25. N. C. Kaufman, "ATRC Fuel Annulus Void Coefficients of Reactivity", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202, (November 1966) p 1.
26. J. L. Durney and N. C. Kaufman, Calculating Reactor Power from Activation Techniques as Applied to an Unusual Fuel Geometry (ATRC), IN-1047 (January 1967).
27. J. L. Durney and J. W. Henscheid, "Power and Flux Measurements in ATRC", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1965, IDO-17140, (March 1966) pp 1-5.
28. J. L. Durney and J. W. Henscheid, "Power and Flux Measurements in a 39-kg Core in the ATRC", Nuclear Technology Branches Quarterly Report, January 1-March 31, 1966, IDO-17192 (August 1966) pp 1-4.

29. J. L. Durney, "Remeasurement of the 50-50-50 Power Distribution in the ATRC", Nuclear Technology Branches Quarterly Report, April 1-June 30, 1966, IDO-17202 (November 1966) p 1.
30. J. W. Henscheid, "Comments on Apparent Asymmetries in ATRC Power Distribution", Nuclear Technology Branches Quarterly Report, January 1-March 31, 1966, IDO-17192 (August 1966) pp 6-10.
31. N. C. Kaufman, "Effects of Venting ATR Fuel Element Side Plates", Nuclear Technology Branches Quarterly Report, October 1-December 31, 1964, IDO-17081 (July 1965) pp 5-9.
32. J. L. Durney, "Flux Depression Measurements in ATR Fuel Plates", Nuclear Technology Branches Quarterly Report, October 1-December 31, 1965, IDO-17191 (June 1966) p 1.
33. J. L. Durney, "Fast Neutron Flux in the ATRC", Nuclear Technology Branches Quarterly Report, January 1-March 31, 1965, IDO-17104 (December 1965) pp 1-2.
34. G. R. Keepin, T. F. Wimett, and R. K. Zeigler, "Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium, and Thorium", Phys. Rev., **107** (August 1957) pp 1044-1049.

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APPENDIX A
PHYSICAL CONSTANTS FOR AN ATR FUEL ELEMENT

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APPENDIX A -- PHYSICAL CONSTANTS FOR AN ATR FUEL ELEMENT

Table A-1 is a list of physical constants associated with an ATR fuel element. All atom densities and macroscopic cross sections in the table were calculated

TABLE A-I
PHYSICAL CONSTANTS FOR AN ATR FUEL ELEMENT

Uranium Content	975 g in U ₃ O ₈ (93% enriched in U-235)
Natural Boron Content	4.15 g in B ₄ C
Metal Volume	2998 cm ³
Water Volume	3950 cm ³
Total Volume	6948 cm ³
Metal-to-Water Ratio	0.759
Atom Densities:	
N(235)	3.60×10^{20} atoms/cm ³
N(238)	0.36×10^{20} atoms/cm ³
N(B)	0.334×10^{20} atoms/cm ³
N(A1)	246×10^{20} atoms/cm ³
N(H)	380×10^{20} atoms/cm ³
Macroscopic Absorption Cross Sections (at 2200 m/sec):	
$\Sigma_a(235)$	0.2460 cm ⁻¹
$\Sigma_a(238)$	0.0001 cm ⁻¹
$\Sigma_a(B)$	0.0254 cm ⁻¹
$\Sigma_a(A1)$	0.0059 cm ⁻¹
$\Sigma_a(H_2O)$	0.0126 cm ⁻¹
$\Sigma_a(\text{Total})$	0.2900 cm ⁻¹
$\Sigma_f(235)$	0.2071 cm ⁻¹

assuming the fuel element constituents to be homogenized over an entire "fuel cell", the cell being defined as a 45-degree sector of a 48-inch-high cylindrical annulus with a 2.908-inch inside radius and a 5.563-inch outside radius (ie, a 2.655-inch-thick annulus). The following microscopic thermal absorption cross sections were used in computing the cross sections in the table:

$$\begin{aligned}\sigma_a(235) &= 683 \text{ barns} \\ \sigma_f(235) &= 575 \text{ barns} \\ \sigma_a(238) &= 2.70 \text{ barns} \\ \sigma_a(A1) &= 0.240 \text{ barns} \\ \sigma_a(H) &= 0.332 \text{ barns} \\ \sigma_a(B) &= 760 \text{ barns}\end{aligned}$$

All other elements or impurities have been neglected since their cross sections or amounts are negligible when calculating macroscopic cross sections.

APPENDIX B
REACTIVITY AS A FUNCTION OF REACTOR PERIODS IN ATRC

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APPENDIX B --

REACTIVITY AS A FUNCTION OF REACTOR PERIODS IN ATRC

A technique frequently used in reactivity measurements is to infer the reactivity worth of a reactor manipulation from the stable reactor period induced by the manipulation. Shown in Figure B-1 is a curve relating stable positive periods to reactivity expressed in dollar units.

The curve shown in Figure B-1 was calculated from a form of the inhour equation which is particularly well-suited to the ATRC. The standard form of the inhour equation relating period T and reactivity ρ is

$$\rho = \omega\Lambda + \sum_i \frac{\omega \bar{\beta}_i}{\omega + \lambda_i} \quad (1)$$

where $\omega \equiv 1/T$ (T in seconds), Λ is the effective neutron lifetime (in seconds), $\bar{\beta}_i$ is the number of delayed neutrons per fission of the i th energy group which reach thermal energies relative to an equivalent number of prompt fission neutrons which reach thermal energies, and λ_i is the decay constant of the i th delayed group (in inverse seconds).

Defining β_i as the absolute yield of delayed neutrons per fission of the i th group (a number characteristic of the fissile material only) and making the additional definitions that $\gamma_i \equiv \frac{\bar{\beta}_i}{\beta}$, $\gamma \equiv \frac{\bar{\beta}}{\beta}$, and $\beta \equiv \sum_i \beta_i$, Equation (1) may be written

$$\rho = \omega\Lambda + \bar{\beta} \sum_i \frac{\gamma_i}{\gamma} \cdot \frac{\omega \frac{\beta_i}{\beta}}{\omega + \lambda_i} \quad (2)$$

To within one percent, the approximation $\gamma_i = \gamma$ is valid and, thus, Equation (2) may be written

$$\rho = \omega\Lambda + \bar{\beta} \sum_i \frac{\omega \frac{\beta_i}{\beta}}{\omega + \lambda_i}$$

or

$$\frac{\rho}{\bar{\beta}} = \frac{\omega\Lambda}{\bar{\beta}} + \sum_i \frac{\omega \frac{\beta_i}{\beta}}{\omega + \lambda_i} \quad (3)$$

Fig. B-1 Reactivity equivalent of stable positive periods in ATRC.

In ATRC, only positive stable periods longer than 20 seconds are used in reactivity measurements. (Negative periods are adversely influenced by source effects in the beryllium reflector.) When such long periods are used, the term in Equation (3) containing $\Lambda/\bar{\beta}$ contributes less than 0.1 percent of the value for $\rho/\bar{\beta}$; hence, the variations in $\Lambda/\bar{\beta}$ described in Chapter III may be ignored. Consequently, an average or typical value for $\Lambda/\bar{\beta}$ is used, 7.1 msec. Thus, Equation (3) for ATRC is

$$\frac{\rho}{\bar{\beta}} = \omega(7.1 \times 10^{-3}) + \sum_i \frac{\omega \frac{\beta_i}{\bar{\beta}}}{\omega + \lambda_i} \quad (3a)$$

Equation (3a) was selected for use in ATRC because the right-hand side may be calculated using only the measured period and constants which are determined by the fissile material. This was important in ATRC because computer simulations indicated that $\bar{\beta}$ varies by as much as 40 percent as the outer shims rotate over their full travel^[15]. However, in Equation (3a) all $\bar{\beta}$ variation will be implicit in the reactivity in dollars, $\rho/\bar{\beta}$, and a specification of the computed reactivity and outer shim position will be unique even though the actual value for $\bar{\beta}$ is not known.

In evaluating Equation (3a) to obtain the curve in Figure B-1, six delayed neutron groups are used with values as reported by Keepin et al^[34]. The large contribution of neutrons from the reflector, often treated as a seventh delayed neutron group, was considered as a prompt neutron contribution since the decay constant calculated for this "delayed" group is so large ($\approx 5 \times 10^3 \text{ sec}^{-1}$).