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AEC RESEARCH AND DEVELOPMENT REPORT

THE DETERMINATION OF AXIAL FLUX AND AXIAL EXPOSURE DISTRIBUTIONS IN THE HWCTR

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THE DETERMINATION OF AXIAL FLUX AND AXIAL EXPOSURE DISTRIBUTIONS IN THE HWCTR

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ABSTRACT

Methods were developed to calculate axial flux distributions from a knowledge of control rod position and fuel exposure during full-power nuclear operation of the HWCTR. The maximum specific power, heat flux, and core temperature of each fuel assembly were determined from the flux distributions to ensure safe operation of the reactor. Also, methods were developed to compute the axial exposure distribution of fuel assemblies as required for evaluation of the irradiation performance of each test fuel design.

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THE DETERMINATION OF AXIAL FLUX AND AXIAL EXPOSURE DISTRIBUTIONS IN THE HWCTR

INTRODUCTION

The Heavy Water Components Test Reactor (HWCTR) is a high temperature, pressure vessel reactor, cooled and moderated with D_2O. It was designed and constructed specifically to test candidate fuel elements for power reactors at operating conditions and exposures similar to those in a full-sized D_2O power reactor. The fuel development program was part of the Du Pont program to advance the technology of D_2O power reactors. Operation of the HWCTR was terminated on December 1, 1964, following a decision by the AEC to redirect the heavy water program toward organic-cooled reactors. The HWCTR facility was placed in a standby condition.

A knowledge of the axial neutron flux distribution in the HWCTR was necessary for two major reasons:

1. To ensure that the reactor was being operated below thermal and hydraulic limits.

2. To determine the specific powers and exposures achieved by test fuel pieces.

The absence of in-core neutron flux monitors required that the axial flux distribution be evaluated solely by computation. In this report, computed flux distributions are compared to data obtained from low-power, wire irradiation tests. Also, computed exposure values are compared to data obtained during postirradiation analyses of several fuel assemblies.

SUMMARY

A mathematical method was developed to calculate axial flux distributions in the HWCTR lattice, and was applied during two driver fuel cycles. The uncertainty in the maximum-to-average flux ratio was ±5% during the second driver cycle, as shown from experiments made at the beginning and at the end of the cycle. The maximum-to-average flux ratios calculated during the first cycle were 5 to 10% higher than the actual ratios, averaged over the cycle.

The total exposure in test fuel assemblies was evaluated from analyses for several uranium and plutonium isotopes and ^{137}Cs present after irradiation. With the exception of the CANDU analyses, which appear to contain analytical errors, the chemically determined

and calculated values for ^{235}U depletion agree within ±5%, and the exposures agree within ±2%. If a more complete sampling of irradiated assemblies had been made, the agreement between the chemically-determined and calculated exposures, at the axial layer at which the maximum specific exposure was achieved, would have been ±8%.

DISCUSSION

Facility Description

A detailed description of the HWCTR facility is given in reference 1. Only a brief description of the reactor and its associated equipment will be given here.

The reactor and principal auxiliary equipment are housed in a building designed to confine steam and radioactivity that might be released by accidental rupture of the reactor system. The containment building is constructed of carbon steel and stressed, reinforced concrete. The building is 70 feet in diameter and 125 feet high, with half of the building above grade. The containment building is shown in Figure 1.

An isometric drawing of the reactor vessel is shown in Figure 2. The vessel is approximately 30 feet high. The reactor core is in the lower third of the vessel. Core components are charged or removed through the top of the vessel. Control rods and safety rods are driven by motors and gear assemblies mounted above the reactor head. Primary D_2O coolant, circulated through two identical systems, enters the vessel above the fuel, passes down through the fuel coolant annuli, and enters the bulk moderator region. The D_2O leaves the vessel through two nozzles near the top of the fuel, and passes through two steam generators before returning to the reactor.

A cross section of the reactor core is shown in Figure 3. The driver fuel elements, in six groups of four elements each, surround the test region. The driver fuel provides the necessary lattice reactivity for operation at full power and temperature. There are 12 test fuel positions inside the driver ring, arranged on a seven-inch triangular spacing. Isolated pressure tubes occupy two of the test positions, and serve to separate their fuel coolant from the main system moderator. Each pressure tube is connected to a separate flow loop having its own pumps and heat removal system.

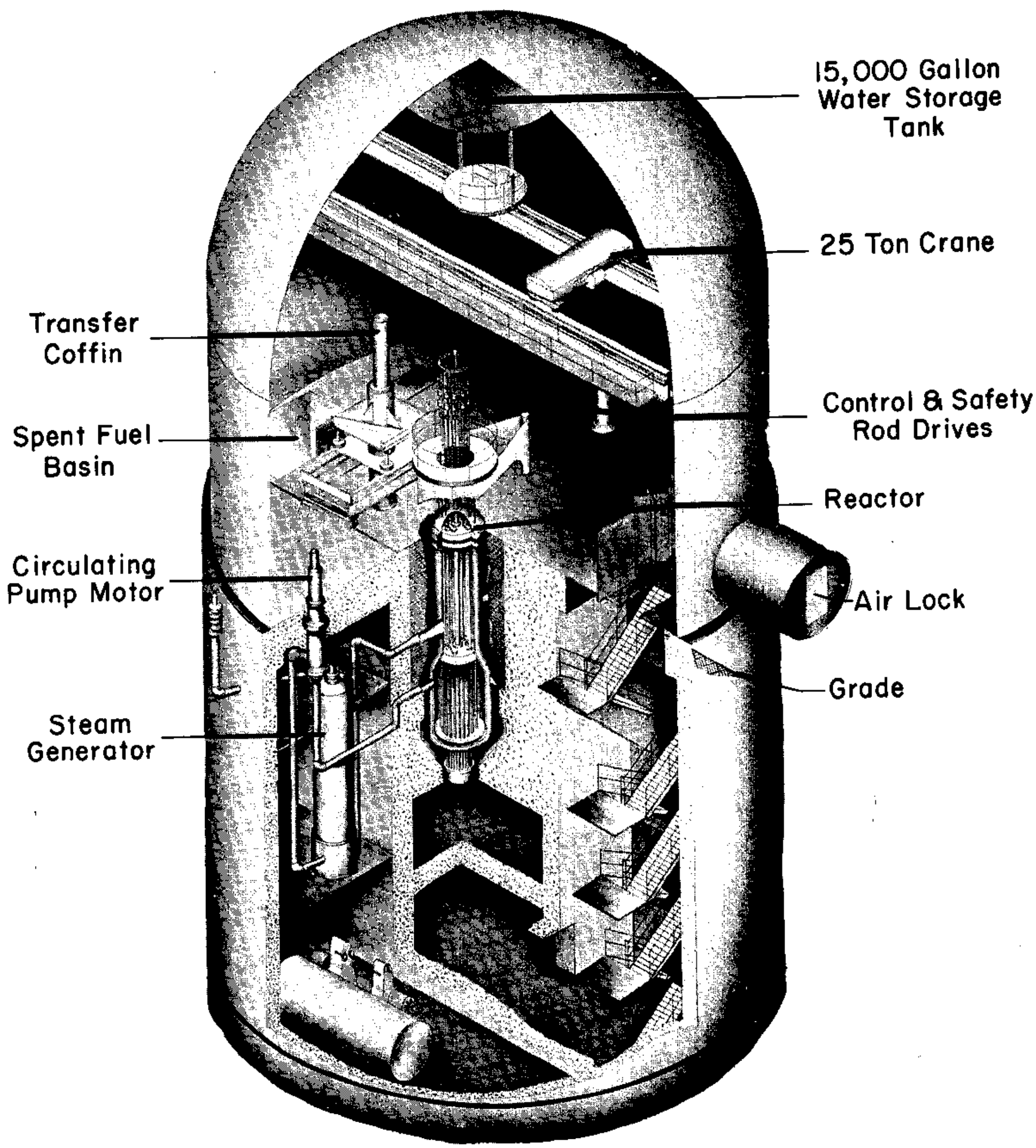

FIG. 1 HWCTR CONTAINMENT BUILDING

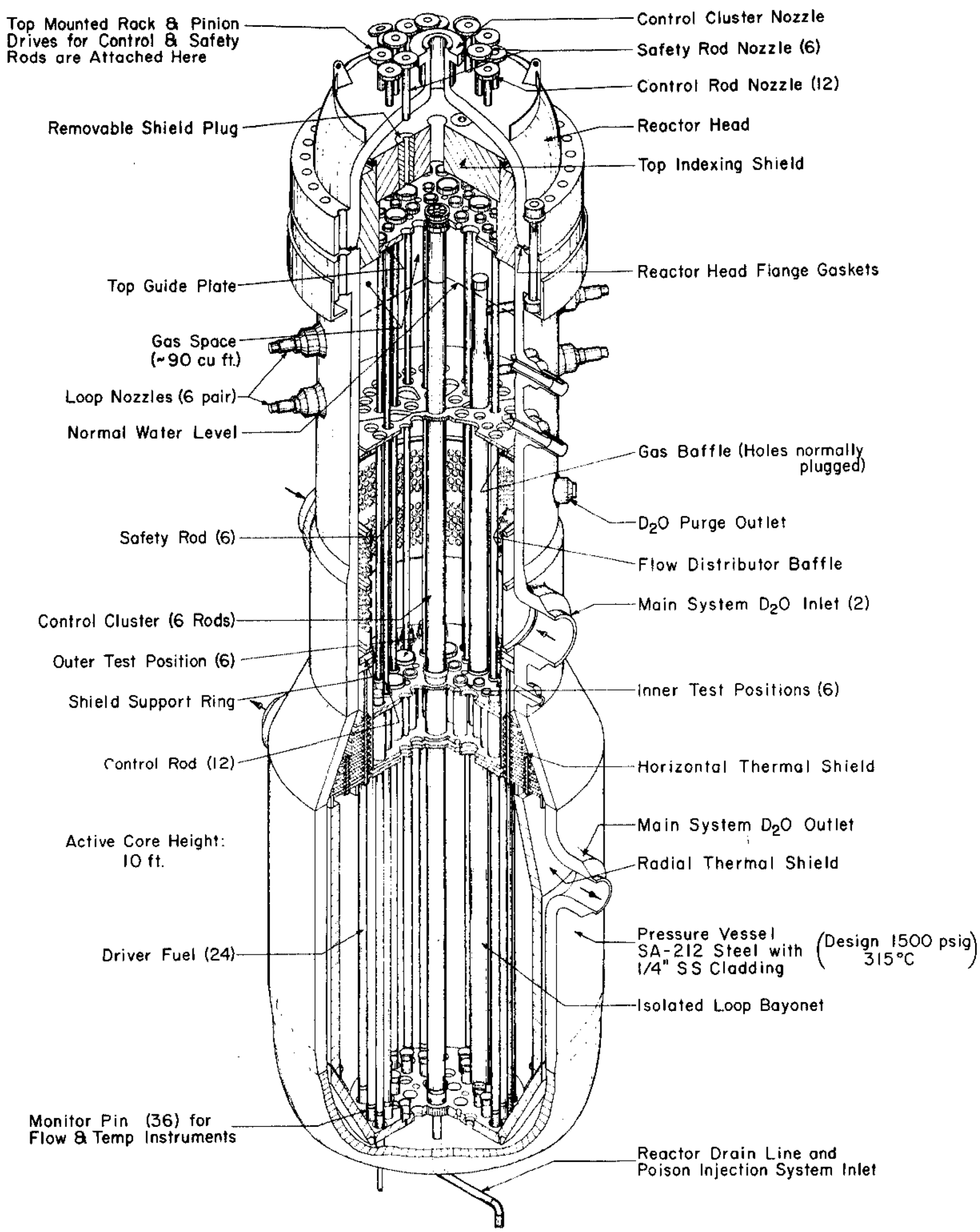

FIG. 2 REACTOR VESSEL AND COMPONENTS ARRANGEMENT

Twelve control rods and six safety rods are arranged in concentric rings inside the driver fuel. A cluster of six control rods occupies the core center. The control and safety rods are tubes of boronated stainless steel having an outside diameter of 1.25 inches and an inside diameter of 1.00 inch. The natural boron content is 1.0 wt %. The control rods are driven in or out at a maximum speed of 2.5 feet per minute. The safety rods are driven out at a speed of 2.5 feet per minute, and can be rapidly inserted from their full out position to 90% insertion in less than two seconds, following a scram signal.

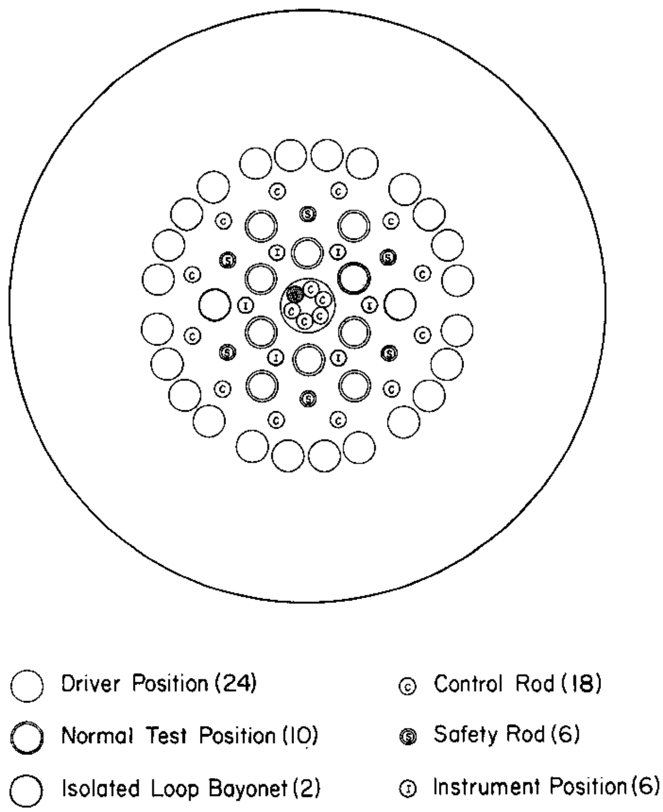

FIG. 3 LATTICE ARRANGEMENT (RADIAL)

Normal operating conditions for the HWCTR were achieved at a reactor power of approximately 50 MW and a moderator temperature of 250°C. Pressurization was provided by a helium gas system, including a small volume of helium in the top of the vessel. Energy was removed from the system by boiling H_2O on the shell side of two vertical U-tube steam generators. The steam was vented to the atmosphere.

The driver assembly and one type of test fuel assembly are shown in Figure 4. The driver fuel tube was Zircaloy clad and contained a ^{235}U loading of 108 g/ft, alloyed with zirconium. The fuel was fabricated by Nuclear Metals, Inc. The target elements, or burnable poison components, consisted of a column of foot-long pairs of boronated stainless steel plates arranged in the shape of a cross.

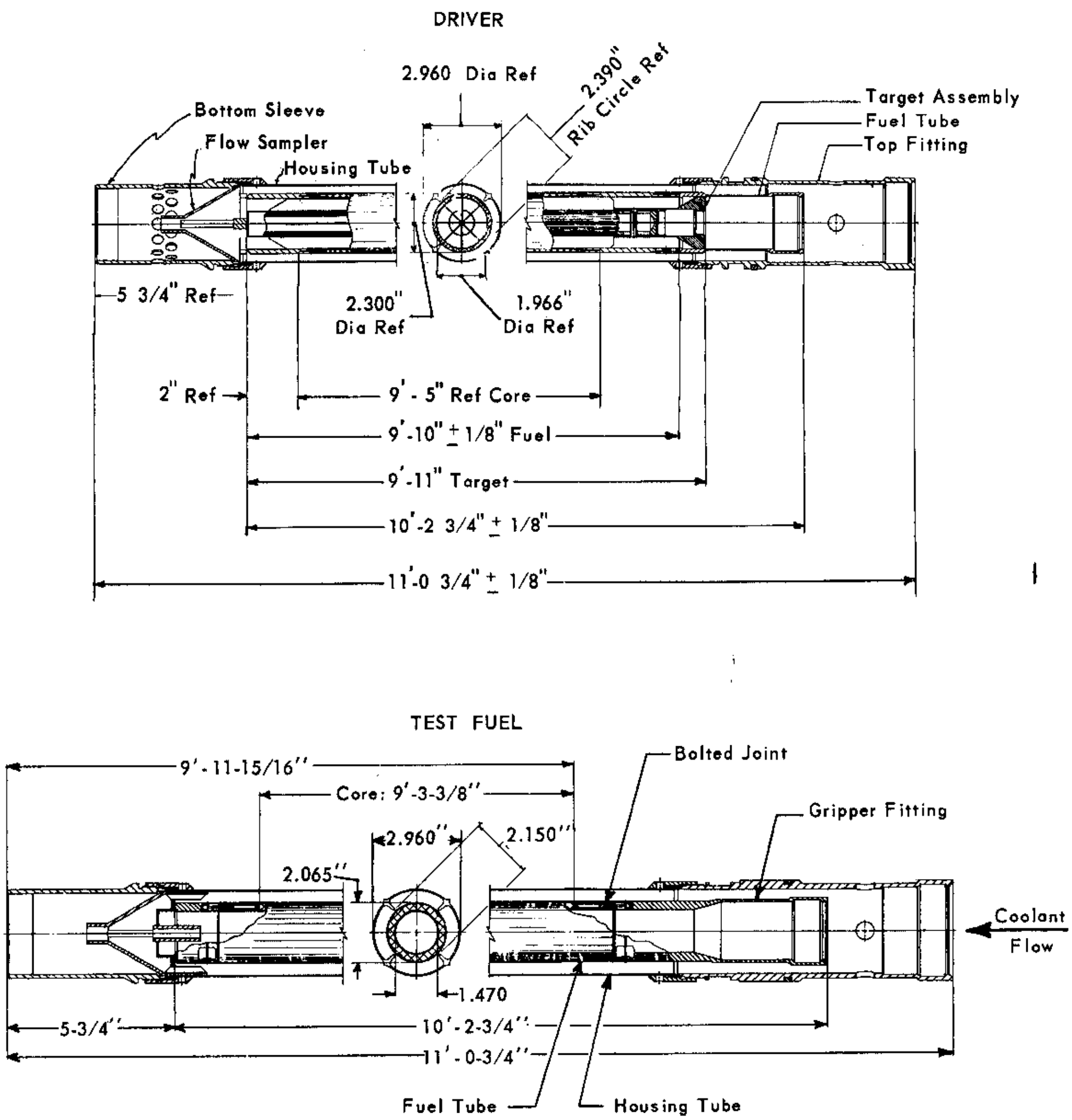

FIG. 4 DRIVER AND TEST FUEL ASSEMBLIES

A variety of test fuel assemblies was irradiated in the HWCTR. The uranium was either metal or oxide. The ^{235}U content varied from 0.71 wt % (natural) to 3.0 wt %. The assemblies were continuous 10-foot tubes, or short slugs stacked in a column.

AXIAL FLUX DISTRIBUTIONS

Computational Methods

The RZ version of the TURBO CODE[2] was used to compute axial flux distributions. TURBO is a two-dimensional depletion code that uses the same neutron diffusion equations as the PDQ-2 code.[3] Figure 5 shows the 2-dimensional representation of the lattice. The mathematical representation consisted of concentric cylinders of finite height that represented the tank wall, driver fuel, control rods, outer test fuel, and inner test fuel. The detailed, pointwise orientation of core components is shown in Figure 3.

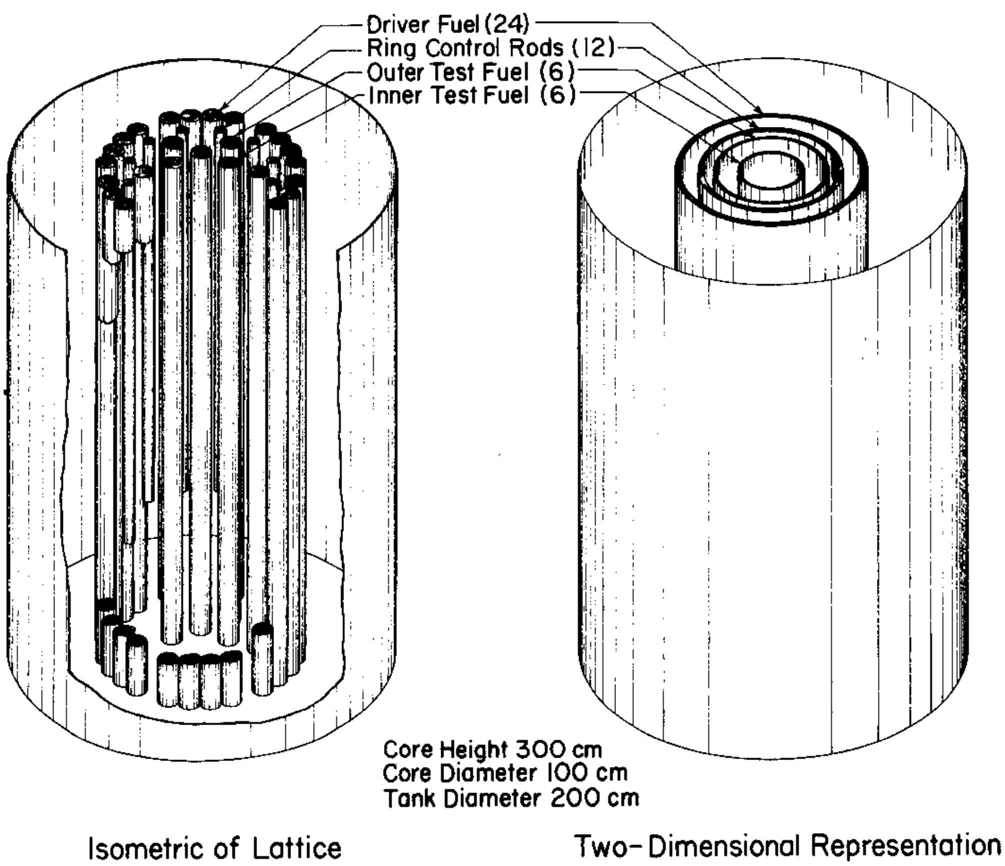

FIG. 5 REPRESENTATION OF HWCTR LATTICE

The representation of the driver fuel as a cylinder surrounding the other core components is a good approximation of the actual driver fuel arrangement in the HWCTR lattice. The ^{235}U depletion experienced by the individual driver elements was within ±10% of the average depletion, which made it possible to treat the driver fuel as a single radial component. TURBO code computations showed that the axial flux distributions in the test fuel were dependent on the driver ^{235}U distribution and the control rod position rather than on the exposure of the test fuel itself. Thus, the two-dimensional representation was used to calculate axial flux distributions at any lattice position as a function of control rod position, and driver fuel and control rod exposure.

Measurements

Prior to initial power operation of the reactor, low-power critical runs were conducted to irradiate 10-foot lengths of copper wire positioned in selected fuel elements. The axial flux profiles were obtained by scanning the wires with a scintillation probe. Input parameters to TURBO for the control rod region were selected empirically to give agreement between calculated and measured distributions.

Depletion of the ^{235}U in the driver fuel was accounted for in the computations as the first driver cycle proceeded. No control rod depletion was allowed because no method had been developed to represent properly the change in control rod parameters with exposure. The real axial flux distributions were known to be less peaked than the calculated distributions. Because specific powers in the fuel were directly proportional to the axial flux values, the fuel elements that limited reactor power were actually operating at specific powers somewhat less than the real limiting values.

Figure 6 shows the calculated distribution of the ^{235}U remaining in the driver fuel at the end-of-reactivity life of the first driver cycle. Approximately 34% of the total amount of ^{235}U in the driver fuel was depleted. The curve represents an exposure-weighted average of the axial power distributions attained during the cycle. The maximum-to-average ratio of the power distributions varied from 2.1 at the beginning of the driver cycle to 1.6 at the end.

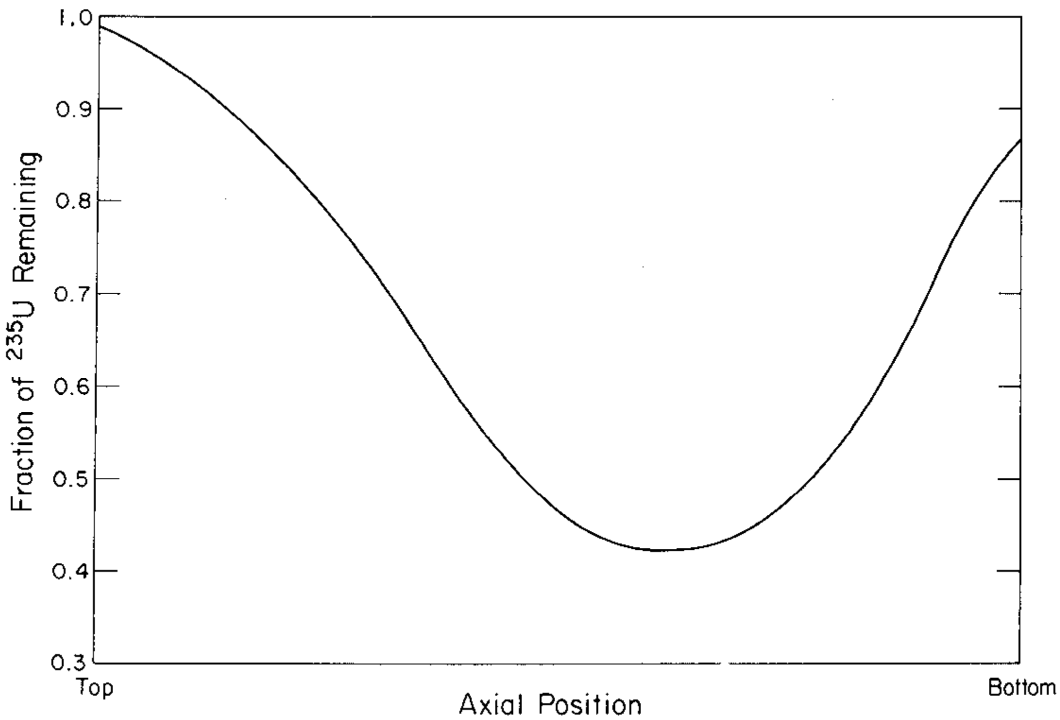

FIG. 6 AXIAL DISTRIBUTION OF ^{235}U REMAINING IN DRIVER FUEL
End of Driver Cycle 1

Measurements were made of axial flux distributions at the end-of-reactivity life of driver cycle 1. Examples of the flux distributions measured when depleted fuel and depleted control rods were present are shown in Figure 7. The calculated maximum-to-average flux ratio was higher than the measured ratio by 11% in the driver fuel and 18% in the test fuel. These differences were never exceeded during operation at power, because the control rod insertion during the tests exceeded the rod insertion during power operation. The control rod insertion during the wire irradiation measurements was 70%; during power operation at the end of the driver cycle, the depleted control rods were inserted only 38% into the core. Uncertainties in the strength of the control rods became less important in calculating axial flux shapes as the rods were withdrawn from the core.

Averaged over the entire cycle, the maximum-to-average ratios of the calculated and measured flux distributions agreed within 10% in all fuel elements.

An empirical method for calculating control rod depletion with exposure, developed at the beginning of driver cycle 2, was used throughout cycle 2.

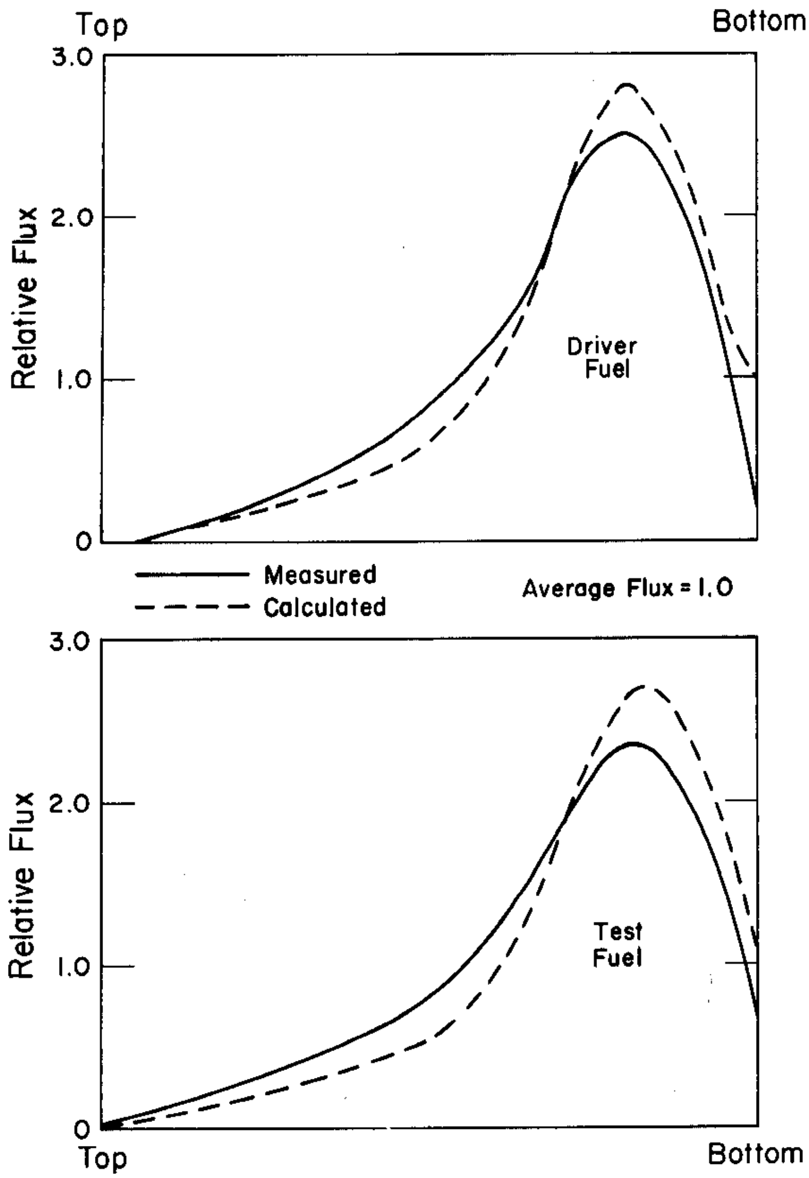

FIG. 7 AXIAL FLUX DISTRIBUTIONS
End of Driver Cycle 1 (old control rods)

Driver cycle 2 was terminated after a ^{235}U burnup of 28% in the driver fuel. At the end of the cycle, axial flux distributions were measured in selected fuel elements by a wire irradiation test. Some of the flux profiles are shown in Figure 8. The measured ratios of the maximum-to-average axial flux were 5 to 8% higher than the calculated ratios. Thus, the empirical method used to calculate control rod burnup overestimated slightly the degree of rod burnup.

The control rod insertion during the wire irradiation test was again larger than the control rod insertion at power, so that the differences between the calculated and measured flux distributions were no larger at power than the 5 to 8% differences just given. Averaged over the full second cycle, the difference between the maximum-to-average ratios of the calculated and measured flux distribution was approximately ±5%.

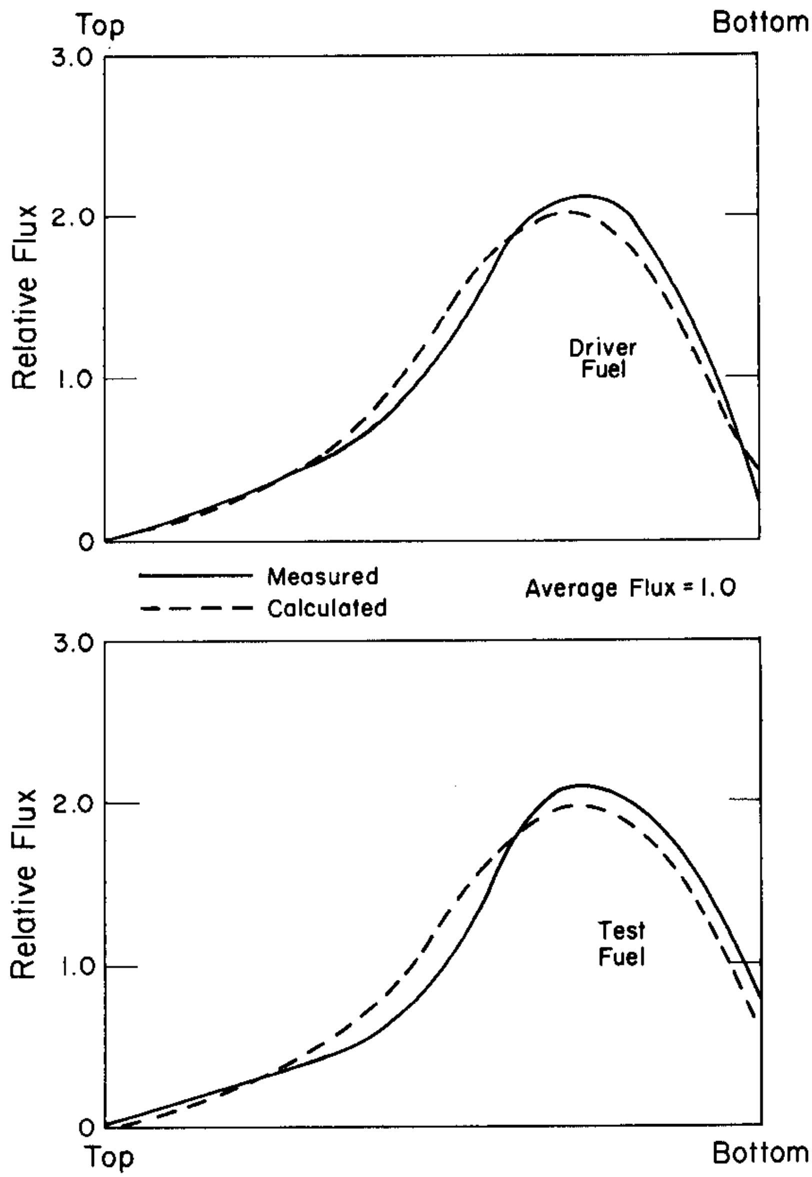

FIG. 8 AXIAL FLUX DISTRIBUTION
End of Driver Cycle 2

Axial Flux Flattening with Exposure

Only a small reduction in the maximum-to-average ratio of the neutron flux resulted from driver fuel depletion. The largest reduction in the calculated ratio was 10%, experienced by test fuel in the inner lattice positions. Figure 9 shows the relationship between the control rod position and the maximum-to-average ratio in the driver fuel. Figure 10 shows the relationship for the inner test fuel.

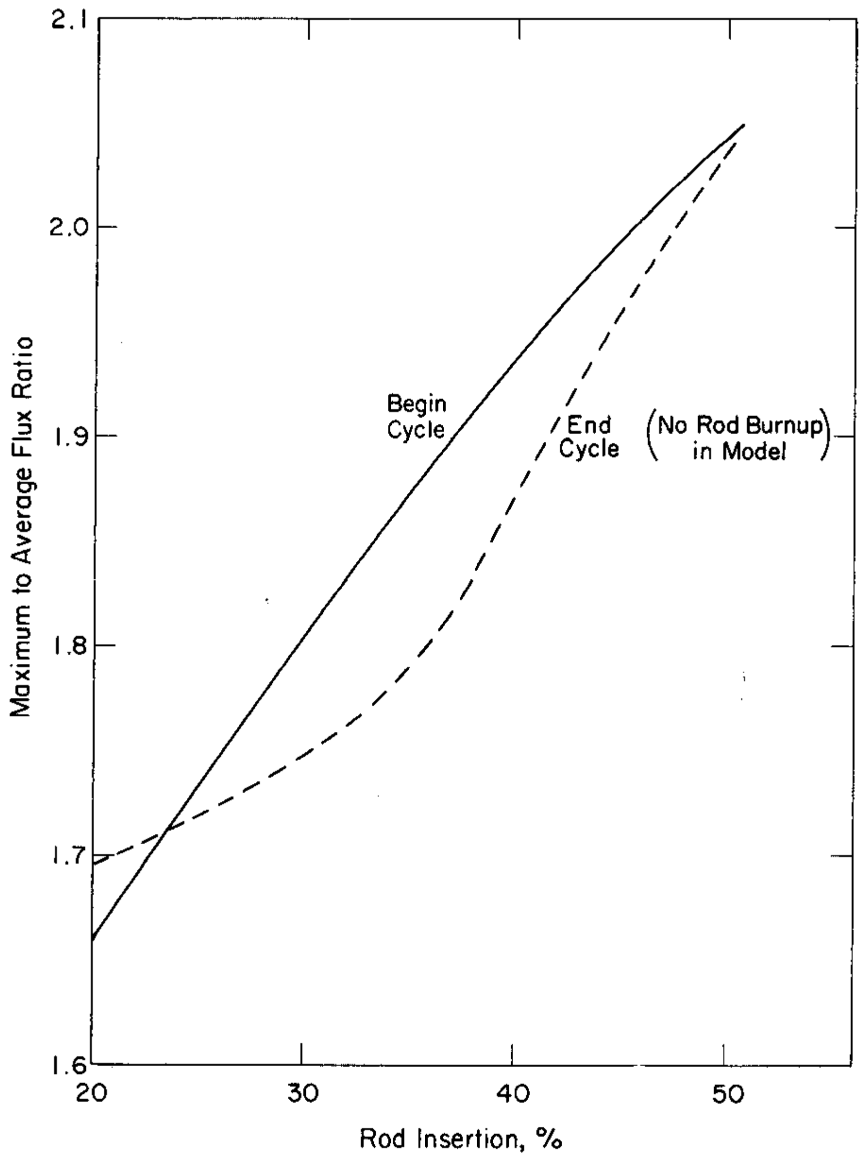

FIG. 9 MAXIMUM TO AVERAGE FLUX RATIO VERSUS CONTROL ROD POSITION
Driver Fuel

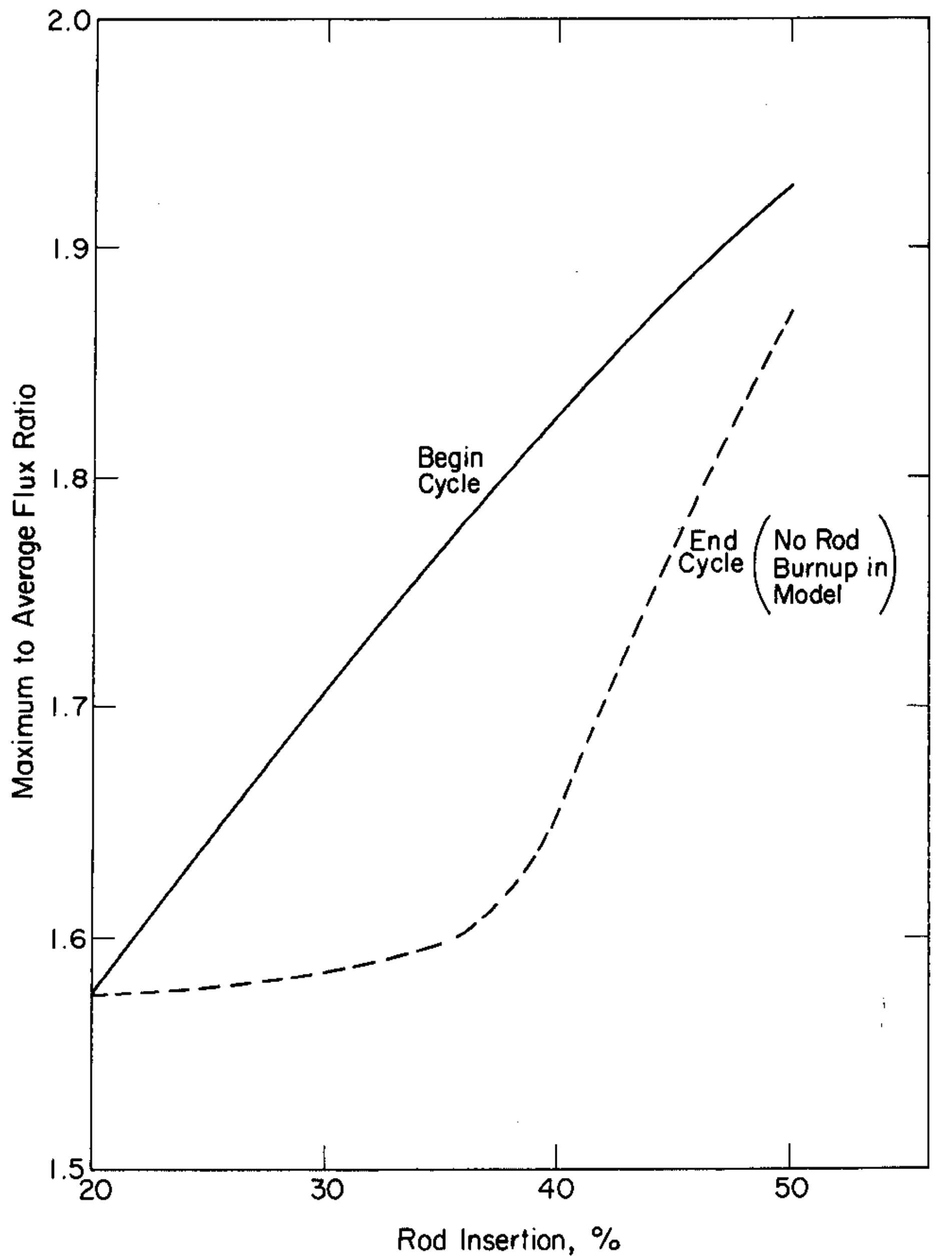

FIG. 10 MAXIMUM TO AVERAGE FLUX RATIO VERSUS CONTROL ROD POSITION
Inner Test Fuel

The axial position at which the maximum flux occurred moved upward with fuel depletion, for a fixed control rod position and strength. At the end of each driver cycle, the calculated maximum flux occurred above the core midplane when control rods were inserted less than 30%.

The changes in the axial power distributions with driver exposure were more marked than the changes in flux distributions. At the end of the driver cycles, the maximum-to-average ratio of power was always less than the flux ratio, and was displaced toward the top of the core. Figure 11 compares a power and a flux distribution near the end of a driver cycle.

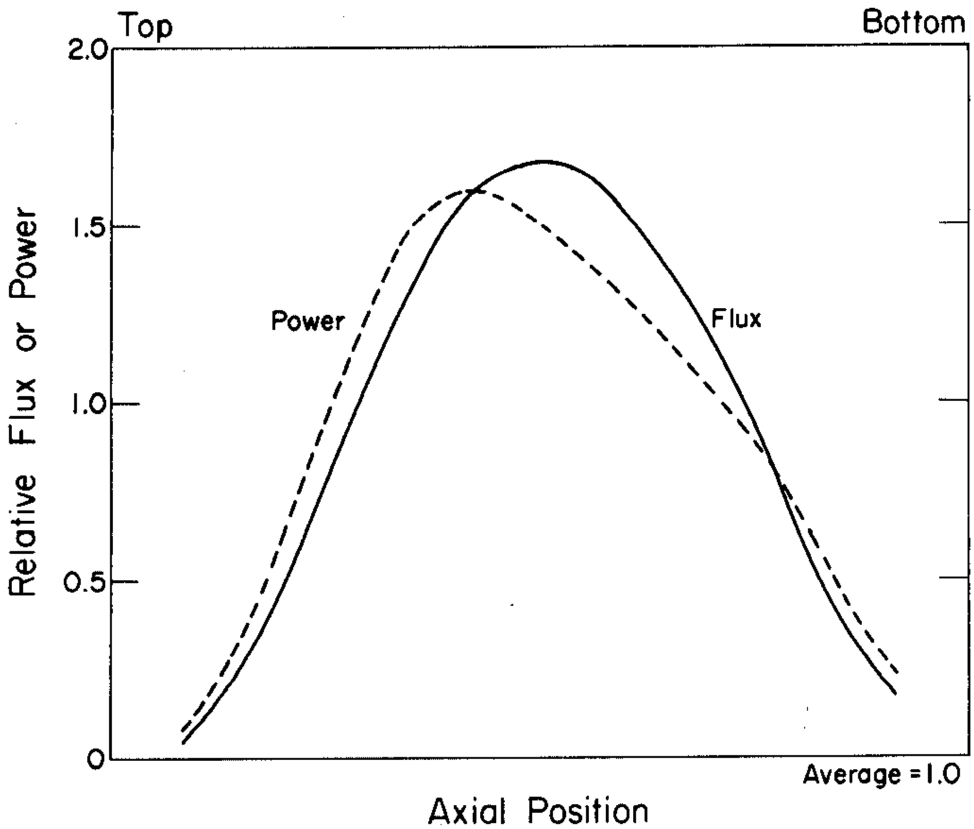

FIG. 11 FLUX AND POWER DISTRIBUTIONS IN DRIVER FUEL AT END OF LIFE

Gamma Activity Scans of Irradiated Fuel

The integrated exposure distribution of the depleted driver fuel was estimated from gamma activity scans made along the longitudinal axis. This estimate is only approximate because of the complexities introduced by the accumulation of the exposure in many short increments over a period of a year. Because the cooling or decay interval between shutdown and time of measurement was short compared to the total operating period, exposure intervals that occurred late in the driver cycle contributed a disproportionate amount of the activity, compared to exposure intervals at the beginning of the cycle. Figure 12 shows the measured activity scans and the calculated activity distributions. The calculated activity distributions were obtained by adding power distributions from each short exposure increment, weighted by a factor for the decay of an average fission product energy. The measured scans in Figure 12 are almost identical for each cycle. The calculated distribution for driver cycle 2 is displaced toward the top of the core more than in cycle 1, showing the effect of control rod burnup on the calculated flux distributions.

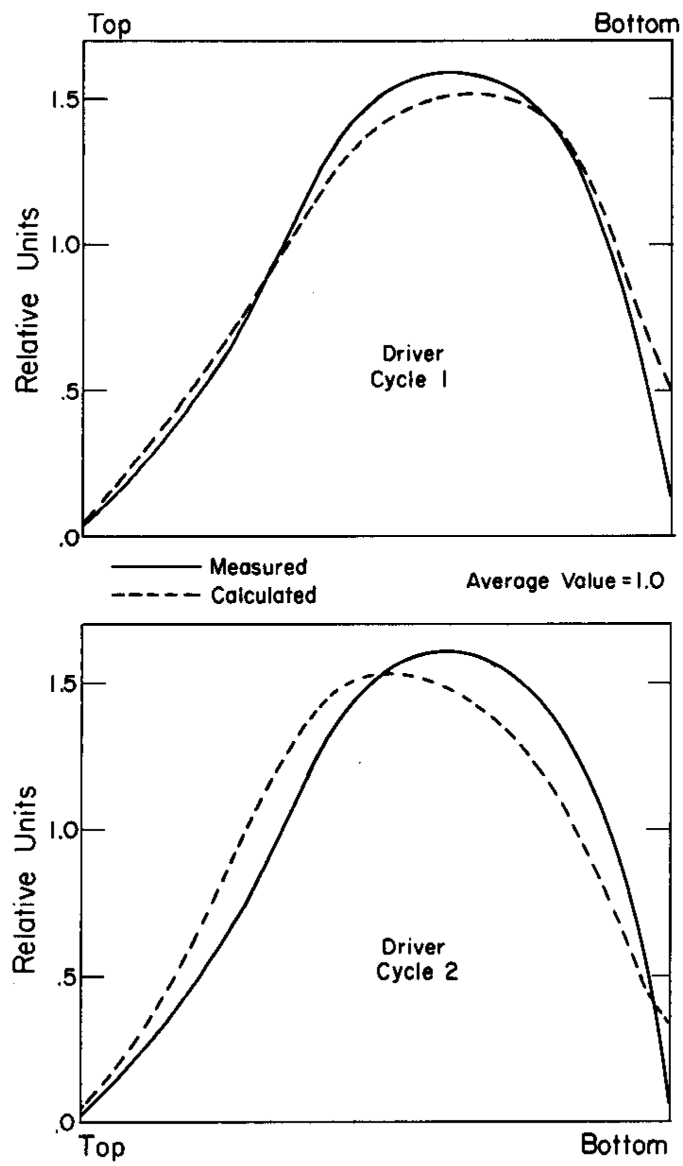

FIG. 12 GAMMA ACTIVITY DISTRIBUTIONS OF DRIVER FUEL

Application of Axial Flux Data

A knowledge of the axial flux distributions was required at all times during reactor operation to permit evaluation of the operating heat fluxes and temperatures in the fuel cores. The flux distributions used for these evaluations gave heat fluxes and temperatures that were higher than actual at the end of the first driver cycle. The true, maximum specific power achieved in the driver fuel, which normally was reactor-power limiting, was 5 to 10% less than calculated. During the second driver cycle, the true maximum-to-average value of the maximum flux was within ±5% of the calculated value. This uncertainty was included as a "hot spot factor" in making heat flux calculations, and did not represent a major penalty to the limiting heat flux or reactor total power.

A second important use of the calculated axial flux distributions was in the evaluation of test fuel performance. The express purpose of operating the HWCTR was to observe the behavior of fuel elements of several designs under power reactor conditions. A knowledge of the specific power generation and specific exposure of an element was necessary to assess its performance quantitatively. Calculated and observed exposure data for several test fuel elements are compared in the next section of this report.

EXPOSURES OF INDIVIDUAL ELEMENTS

Description of RITE Code

The treatment of the driver fuel as a single region in the computations was a valid approximation, as described earlier. However, the test lattice was comprised of a variety of fuel elements, each having a different irradiation history and exposure. Because of the heterogeneity of the test region, it was necessary to provide a detailed accounting of the operating and exposure parameters of each element. A program called RITE (Record of Integrated Test Exposure) was written for the IBM-704 for this purpose. The RITE program calculated the element exposure, specific exposure, and specific content of several uranium and plutonium isotopes at each of 21 axial positions of each HWCTR fuel element. This information was an essential part of the fuel irradiation program to evaluate the performance of each test fuel design as a function of power, temperature, and exposure.

The RITE program contained explicit solutions to the differential equations for the formation and burnup of several uranium and plutonium isotopes. One-group cross sections were used, containing both a thermal and an epithermal component. Input data consisted of observed element powers, axial flux distributions, and operating time increments.

Data from Two Oxide Assemblies

Chemical analyses were made of the isotopic content of two oxide assemblies at 12 axial positions. The results are presented in Figures 13 through 17. The assemblies are designated as SOT, for Segmented Oxide Tube.

The agreement is good between the measured and calculated distributions. The exposure profiles, which are of most interest in evaluating fuel irradiation performance, agree within the experimental error of the analyses. The measured exposure values were obtained from observed ^{137}Cs concentrations.

The observed ^{235}U depletion was somewhat less than calculated, but the ^{239}Pu present was higher than calculated; the ratio of resonance to thermal absorptions in ^{238}U may have been higher than calculated.

Because of the good agreement between the observed and calculated exposure distributions, no additional chemical analyses were made until after the HWCTR operation was terminated.

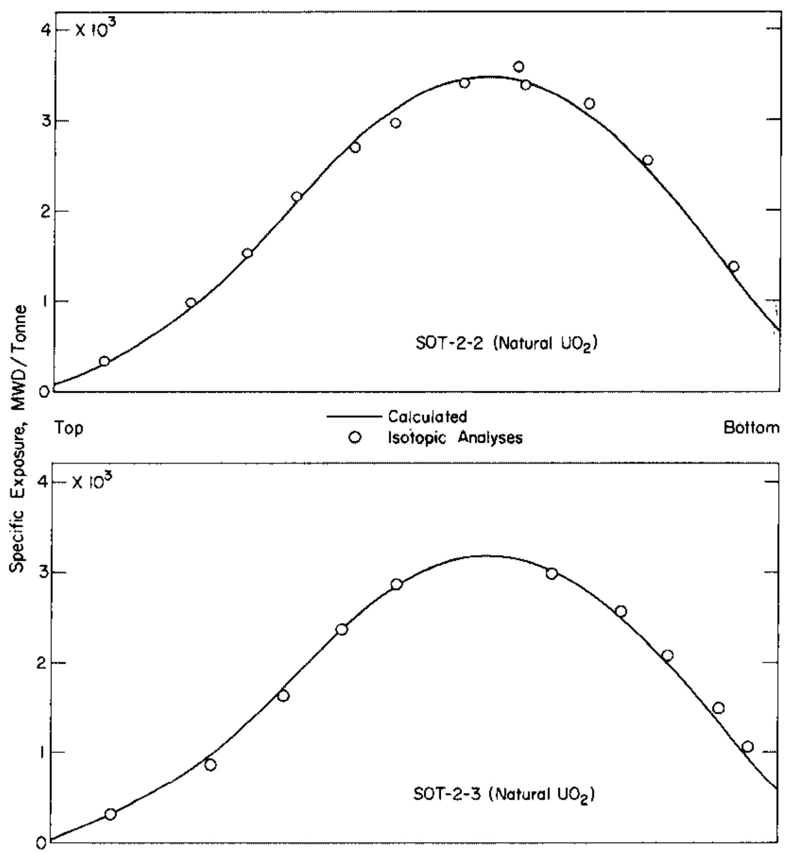

FIG. 13 AXIAL EXPOSURE DISTRIBUTIONS

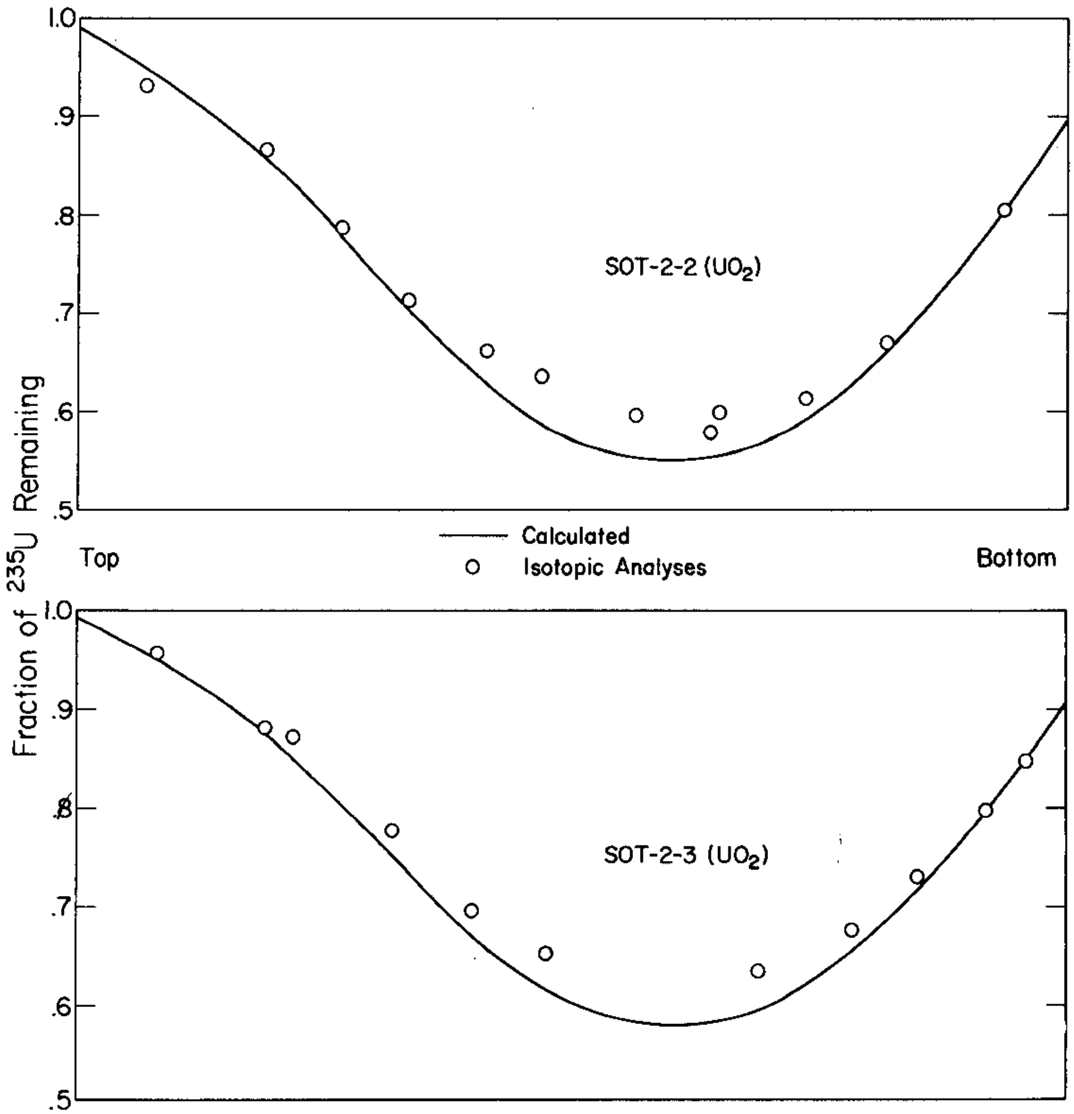

FIG. 14 AXIAL DISTRIBUTIONS OF ^{235}U

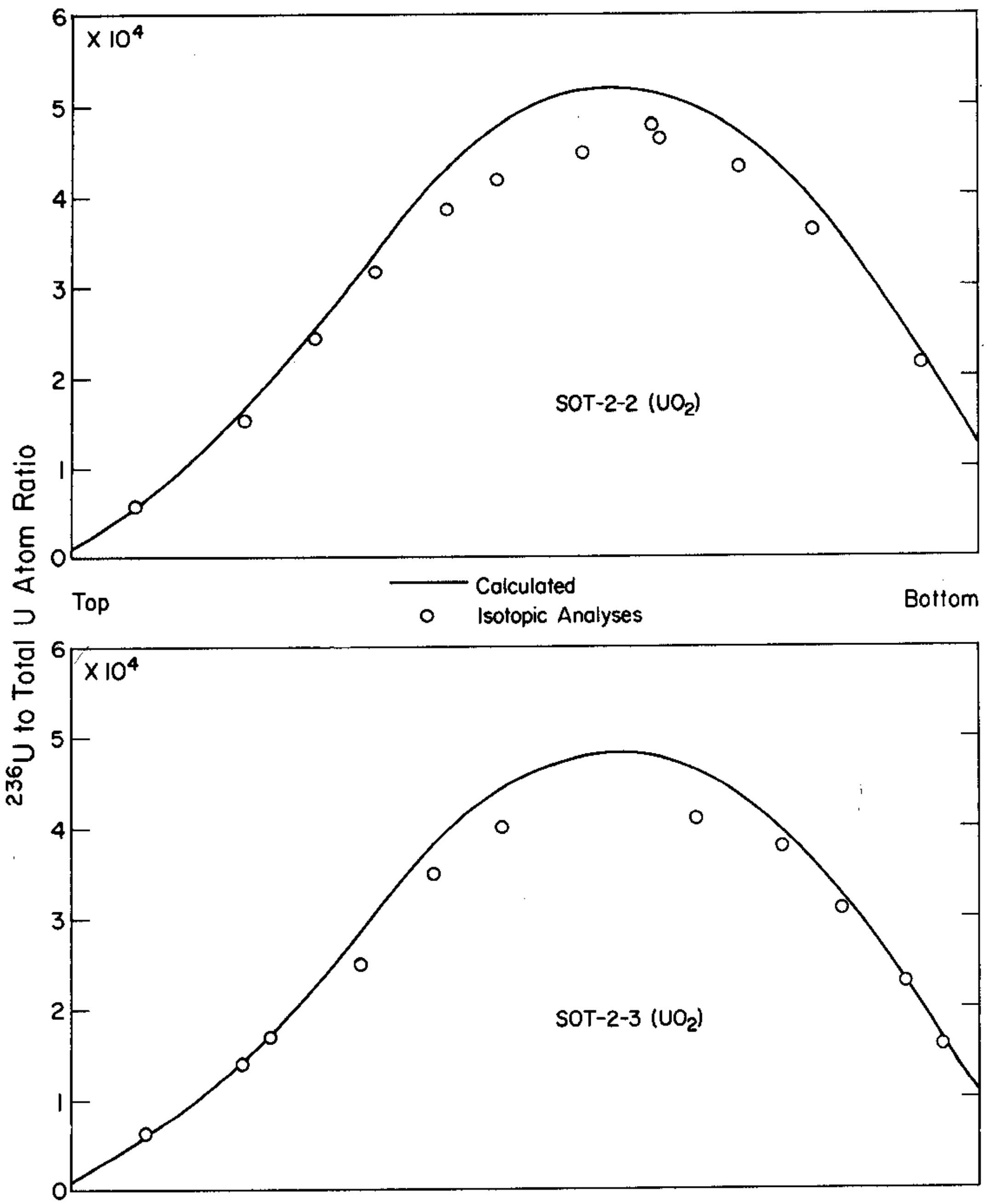

FIG. 15 AXIAL DISTRIBUTIONS OF ^{236}U

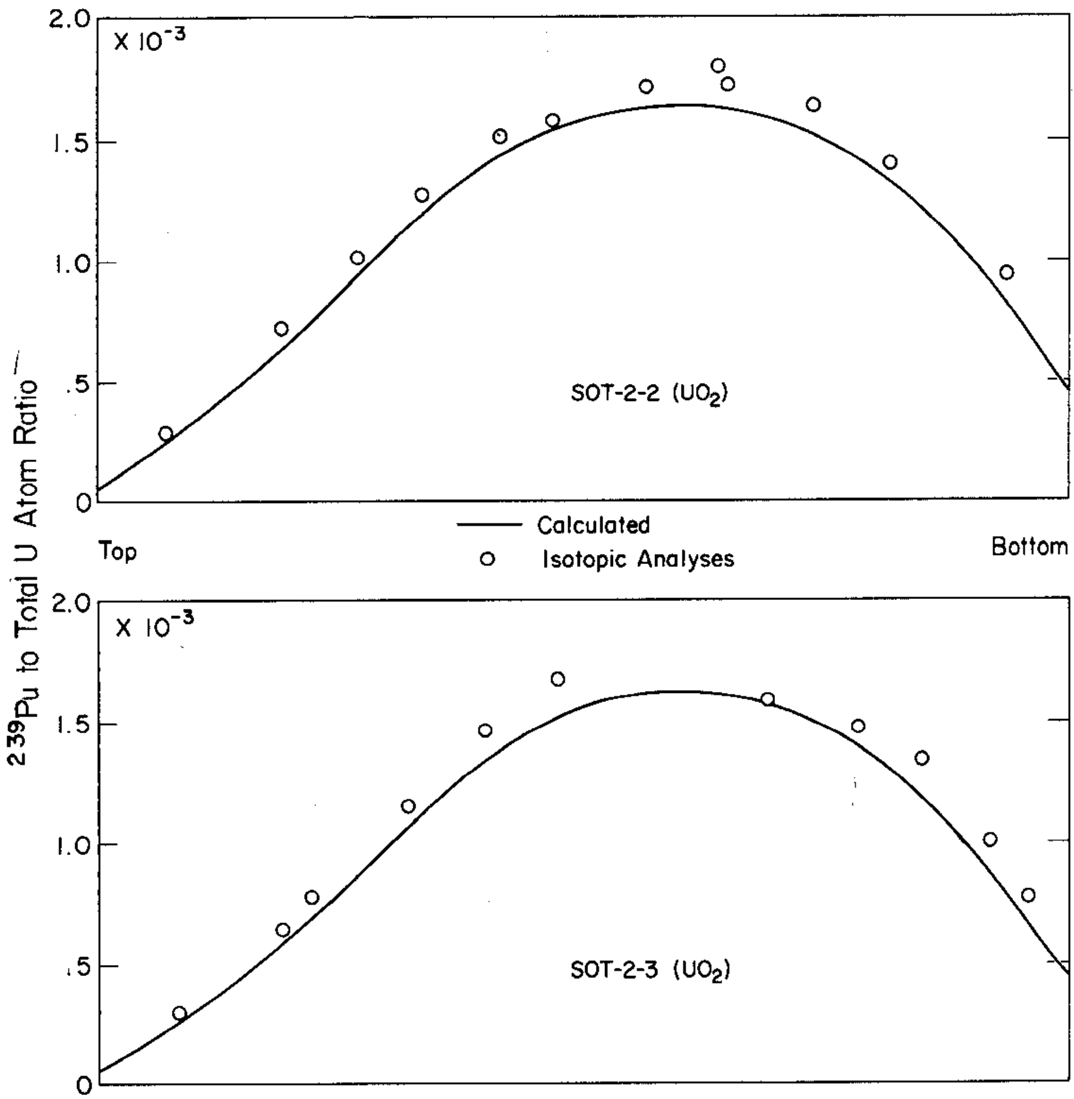

FIG. 16 AXIAL DISTRIBUTIONS OF ^{239}Pu

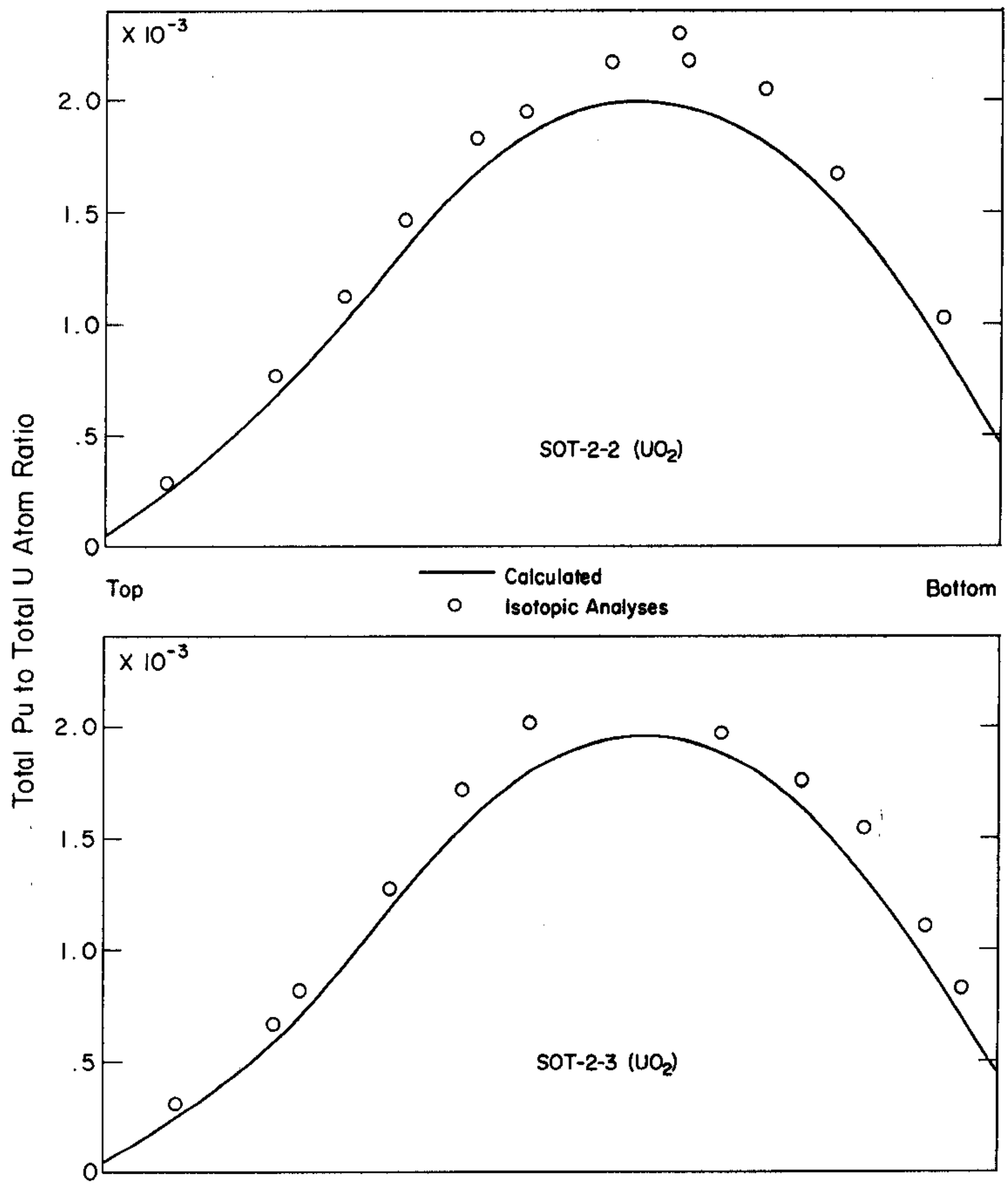

FIG. 17 AXIAL DISTRIBUTIONS OF TOTAL Pu

Data from CANDU Assembly

One fuel assembly tested in the HWCTR throughout most of the operating period of the reactor was the CANDU 19-rod bundle of oxide fuel.* This assembly was a prototype of the fuel rod bundles used in the Canadian power program. Following the discharge of the assembly from the reactor, samples were taken from fuel rods at different radial positions in the assembly. The results of the ^{235}U, ^{236}U, and Pu analyses are not consistent with each other and appear to contain errors.

The primary purpose in making chemical analyses of the CANDU assembly was to show that some fuel rods had achieved specific exposures in excess of 6000 MWD/Tonne, as the calculations showed. Because the data indicated that the 6000 MWD/Tonne exposure was achieved, and because HWCTR operation was being terminated, no new samples were taken to determine the reason for the discrepancies in the experimental data.

Data from SOT-1-2 Assembly

Samples were taken from the SOT-1-2 assembly for chemical analysis. This fuel element contained seven annular fuel pieces containing UO_2, originally enriched to 1.5 wt % ^{235}U. It achieved the highest specific exposure of any assembly irradiated in the HWCTR, and was present in the reactor longer than any other fuel assembly. The calculated and experimentally determined parameters are compared in Table I.

TABLE I

SOT-1-2 Assembly Irradiation Data

	Fuel Piece 1		Fuel Piece 2	
	Chem Anal	Calc	Chem Anal	Calc
^{235}U depleted, % of original	88.3	85.3	87.3	85.8
^{236}U present, atom %	0.22	0.21	0.22	0.21
Total Pu/U, %	0.005	0.0055	0.0043	0.0055
Specific exposure, MWD/Tonne	17,200	16,800	17,400	17,100

* The 19 rods were oriented radially in the standard, close-packed arrangement consisting of 1 rod in the center, 6 rods in a ring, and 12 rods in a second ring.

The agreement is excellent, except for the Pu/U ratio of fuel pieces. The conservatism in the axial flux calculations for driver cycle 1 resulted in calculated maximum neutron flux values slightly higher than the true maximum flux. Uncertainties in other parameters, such as ^{239}Pu production from resonance capture, compensated for this known error and resulted in the good agreement in exposure values for the full irradiation period.

The close agreement in the exposure values is better than would be expected, if all uncertainties are considered. Parameters measured or calculated to obtain the calculated exposure included flux distribution, coolant flow, coolant temperature rise, fuel element gamma escape energy, and the ratio of resonance to thermal absorption in ^{238}U. If sampling had been more detailed among the test fuel assemblies that were irradiated, the difference between calculated exposures and those inferred from ^{137}Cs analyses would have been 7 to 10%. The difference would have been 10 to 15% for sections of the test fuel assemblies near the top or bottom of the reactor fuel core; agreement between calculated and measured flux distributions was poorer in those regions than at the axial position of maximum flux.

CALCULATIONS IN FUTURE HWCTR LATTICES

If the reactor is operated again, improvements can be made in the treatment of fuel elements in the lattice. If the test lattice contains a variety of element designs as it did previously, lattice calculations can be improved by a three-dimensional lattice code. The French have recently developed the TRIHET code,[4] which probably could be adapted for use in representing the HWCTR lattice pattern. Treatment of individual elements in a 3-dimensional model would permit a better evaluation of interactions between test elements, which was impossible in the 2-dimensional TURBO code.

Also, methods have been improved for calculating resonance integrals and resonance absorptions since the relationships used in the RITE code were developed. Revised input data to the RITE code or its equivalent should include the improved values for the resonance-to-thermal capture ratios. Significant differences in the quantities of Pu determined by chemical analyses and by calculation were probably the result of errors in the calculation of resonance production.

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