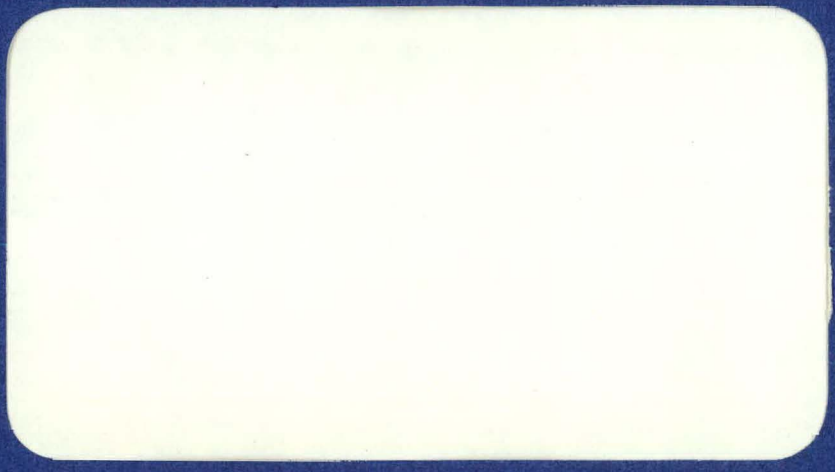

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SAXTON PLUTONIUM PROGRAM

SEMIANNUAL PROGRESS REPORT

FOR THE PERIOD

ENDING DECEMBER 31, 1967

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Prepared for the New York Operations Office
U.S. Atomic Energy Commission
Under AEC Contract AT(30-1)-3385

Printed in USA. Available from Clearinghouse for Federal Scientific and Technical Information National Bureau of Standards, U.S. Department of Commerce, Springfield, Va. 22151
Price: Printed Copy $3.00; Microfiche $0.65.

March 1968

WESTINGHOUSE ELECTRIC CORPORATION
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TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

LIST OF TABLES

SECTION 1

SUMMARY

1. GENERAL

This semi-annual report replaces the thirteenth and fourteenth quarterly reports previously planned and covers the work done during the second half of calendar year 1967.

2. OBJECTIVES

The purpose of the Saxton Plutonium Project is to develop information concerning the utilization of plutonium enriched fuel in pressurized water reactor systems through design, fabrication and operation of a partial core of PuO_2-UO_2 in the Saxton reactor. In-pile performance of this fuel will be evaluated and post-irradiation examination of fuel samples will be made. Performance will be analyzed and compared with analytical predictions.

3. PROGRESS DURING PERIOD

The reactor operated for 850 equivalent full power hours during this six month period. At the end of the period the average burnup in the plutonium-fueled region was calculated to be 13,820 MWD/MTM; peak burnup in the plutonium fuel was calculated to be 24,075 MWD/MTM.

The analytical data from the post-irradiation evaluation of the four low burnup rods have been analyzed. Predicted fuel burnup and isotopic compositions were compared to those determined using the chemical and spectrometric data. Good agreement has been obtained. The operations follow and depletion analysis effort has continued with good results. The non-uniform axial depletion correction has been re-evaluated; the use of a more exact representation for the buildup of fission product poisons has been reviewed. Work was begun on a revised depletion analysis for the core using a LEOPARD-PDQ7-HARMONY sequence.

The fission gas release has been calculated for the low burnup rods. These calculated values are very close to those observed experimentally.

SECTION 2

SAX-100, PROJECT ADMINISTRATION

(R. S. Miller and J. B. Roll)

The remaining program objectives are: 1) to continue operating the plutonium fuel for a minimum of one equivalent full-power year in the Saxton reactor, following operation periodically and analyzing the nuclear performance of the partial plutonium core; 2) to conduct post-irradiation examination of selected fuel samples; 3) to compare nuclear and materials performance with predicted performance; and 4) to analyze any discrepancies found.

During the report period the reactor operated for an additional 850 equivalent full power hours. At the time of the December shutdown, Saxton Core II had achieved an average burnup of 8430 MWD/MTM; with an average burnup of 13,820 MWD/MTM in the plutonium-fueled region, 17,860 MWD/MTM in the peak plutonium rod, and 24,075 MWD/MTM in the peak plutonium pellet. Completion of Core II power operation is now scheduled for the second quarter of 1968. Approximately 6 weeks of operation at 35 MWt are scheduled immediately prior to completion of power operation. Following the end of power operation, end-of-life zero power physics evaluations will be conducted.

The Semi-Annual Progress Report for the period ending June 30, 1966, WCAP-3385-12/EURAEC-1877, was prepared and distributed. The Work Program for fiscal year 1968 was approved by the AEC.

A paper titled "Operating Experience with the Saxton Reactor Partial Plutonium Core II" was presented October 4 at the AIME Symposium on Plutonium Metallurgy in Phoenix, Arizona. This paper discussed the design philosophy, operating experience, examination of the low burnup fuel, and plans for the end-of-life examination for the Saxton Plutonium fuel. In addition, the results of the examination of the low burnup fuel were presented at the recent US-UK Libby-Cockroft Exchange meeting in San Francisco and the 25th High Temperature Fuels Committee Meeting in San Diego.

It has been pointed out that the previously reported gamma-scan curves (Figures 660.4 through 660.7 of Reference 1; Figures 660.1 and 660.2 of Reference 2) were erroneously labeled. The identification of TOP and BOTTOM was inadvertently reversed on all figures.

SECTION 3

SAX-510, NUCLEAR ANALYSES OF OPERATING PERFORMANCE

(F. L. Langford, Jr., W. L. Orr, R. L. Thompson
R. J. Nodvik and D. Dabby)

1. EVALUATION OF FUEL CHEMICAL ANALYSES

1.1 Introduction

The post-irradiation evaluation of four low burnup rods includes a sequence of chemical analyses of selected fuel samples for fission product and heavy element isotope burnup indicators and the evaluation of these analytical data with respect to predicted fuel performance. The analytical data have been summarized previously.[2] The evaluation of these data is discussed below.

Under the Yankee Core Evaluation Program, a significant effort was expended to develop and refine methods for reducing radiochemical and mass spectrometric analytical data to a value of fissions/cc for selected fuel samples. However, this effort was oriented to uranium fuel in general and Yankee fuel in particular. The primary tool derived for this purpose is the computer code REBUF,[3] which was modified to make use of this prior work in the Saxton Plutonium Program. These modifications were necessary because the fuel in the Saxton Plutonium Program had an initial plutonium content, whereas the Yankee fuel has no plutonium initially.

1.2 Test of Data Reduction Methods

Prior to use of the modified REBUF code for reduction of the measured low burnup plutonium fuel data, the internal consistency of the code was tested, using synthetic data sets obtained from LEOPARD[4] runs. Using LEOPARD, the operation of the Saxton reactor was simulated to approximately the burnup anticipated for these fuel samples. The simulated sample isotopic compositions were used as input to REBUF. The resulting inferred fuel characteristics

are compared to the corresponding fuel characteristics obtained from the LEOPARD calculations in Table 3-1. These results indicate that the modified REBUF code is internally consistent.

1.3 Burnup Evaluations

For each of four fuel samples, (two from a pellet rod, two from vipac rod) values of fuel burnup were inferred by several different methods:

1. Mass spectrometric analysis for the stable fission product Nd-148;
2. Radiochemical analysis for Cs-137 activity;
3. Radiochemical analysis for Sr-90 activity; and
4. Mass balance of pre-irradiation and post-irradiation uranium and plutonium isotopic concentrations, referred to as the heavy element (HE) method.

These inferred burnups are shown in Table 3-2 along with the burnup calculated based upon the reactor operating log and the average radial power factors for the particular fuel subassembly containing these rods. Considering the sample average of the burnup values inferred from the fission product data, an average absolute difference of ~10 percent is noted. This discrepancy is due in part to the use of average subassembly radial power factors. The power skewing effects, although known to exist across the subassembly, were not considered in obtaining the calculated burnup. The algebraic average percent difference of only 2.4 percent (calculated > inferred) is consistent with the reported[5] estimated error in measurement of burnup by the neodymium-148 fission product method of ±2 percent at 1σ (not including the uncertainty in the MEV/fission energy conversion factor).

Whereas the values of burnup inferred from the fission product indicators show a reasonable degree of consistency, the values inferred from the HE data are relatively low. The data reduction scheme for the latter depends upon a precise knowledge of the initial fuel composition and upon a knowledge of the neutron spectrum "seen" by the fuel during its irradiation history. In the case of these four samples, pre-irradiation isotopic composition was not

TABLE 3-1

COMPARISON OF CALCULATED FUEL CHARACTERISTICS AND FUEL CHARACTERISTICS INFERRED FROM CALCULATED U AND PU ISOTOPIC DATA

	Type of Fuel			
	Pellet (Case 1)	Pellet (Case 2)	Vipac (Case 3)	Vipac (Case 4)
Burnup (GWD/MTM)				
Direct Calculation	2.530	5.056	2.670	5.336
Inferred with REBUF Code	2.559	5.076	2.661	5.366
U-235 Fractional Depletion				
Direct Calculation	0.02808	0.05574	0.02938	0.05835
Inferred with REBUF Code	0.02807	0.05573	0.02935	0.05834
Alpha-25				
Direct Calculation	0.3216	0.3206	0.3185	0.3176
Inferred with REBUF Code	0.3215	0.3207	0.3190	0.3176
Pu/U Mass Ratio				
Direct Calculation	0.6936	0.6804	0.6927	0.6786
Inferred with REBUF Code	0.6936	0.6804	0.6927	0.6786
Final-to-Initial U-238 Atom Ratio				
Direct Calculation	0.99879	0.99758	0.99874	0.99747
Inferred with REBUF Code	0.99879	0.99759	0.99878	0.99748

TABLE 3-2

FUEL BURNUP

SAMPLE	D-BU-1	D-BU-2	X-BU-1	X-BU-2
Inferred Burnup:				
HE	5240	5200	5130	4330
Cs-137	6146	5520	5207	4881
Sr-90	6445	6143	5277	4819
Nd-148	6223	5514	5246	4717
Avg. (of fission product indicators)	6305	5725	5243	4805
Calculated Burnup	6015	5210	6240	5240
$\frac{\text{Calc.-Avg.}}{\text{Calc.}} \times 100$	-4.8	-9.9	16.0	8.3

redetermined at the time of the post-irradiation analyses, and the neutron spectrum was quite variant since the subassembly was relocated part way through its life.

1.4 U and Pu Isotopic Composition

Using the fuel burnup inferred from the Nd-148 data as a reference value, the uranium and plutonium isotopic abundance and net isotopic production or destruction are plotted in Figures 3-1 through 3-4. Further, the LEOPARD calculations for a pellet fuel asymptotic unit cell are shown for reference. It is emphasized that any conclusions drawn from these evaluations are tentative since the fuel samples were exposed to mixed neutron spectra during irradiation; that is, a portion of the exposure time was in Core I in a uranium fuel environment on the edge of the core, and the remainder of the exposure time was in Core II in a plutonium fuel environment in the center of the core.

The measured U-235 and U-236 abundance is shown on Figure 3-1. The measured data are consistent and, in the case of the U-236 data, are in excellent agreement with the calculation. In the case of the U-235 abundance, the calculation underpredicts by about three percent relative; this discrepancy may be due to a bias in the measurements or to an inaccurate knowledge of the initial U isotopic composition. It is noted, however, that the initial fuel in content consisted of natural uranium. During the final fuel examination, it is planned to submit archive samples for mass spectrometric analysis to further investigate this noted discrepancy. Figure 3-2 shows the net destruction of U-235 and U-238 and the net production of U-236. The U-235 and U-236 data are consistent with the measured atom percent data shown on Figure 3-1. Only three of the four results obtained for U-238 net destruction are shown on Figure 3-2; the value obtained for sample D-BU-1 is inconsistent and is not shown.

The measured Pu isotopic composition is shown on Figure 3-3. The measured data are consistent and show very little scatter. Although agreement between calculation and measurement is excellent for Pu-239 and Pu-240 abundance, the calculation overpredicts the Pu-241 abundance by about 10 percent relative,

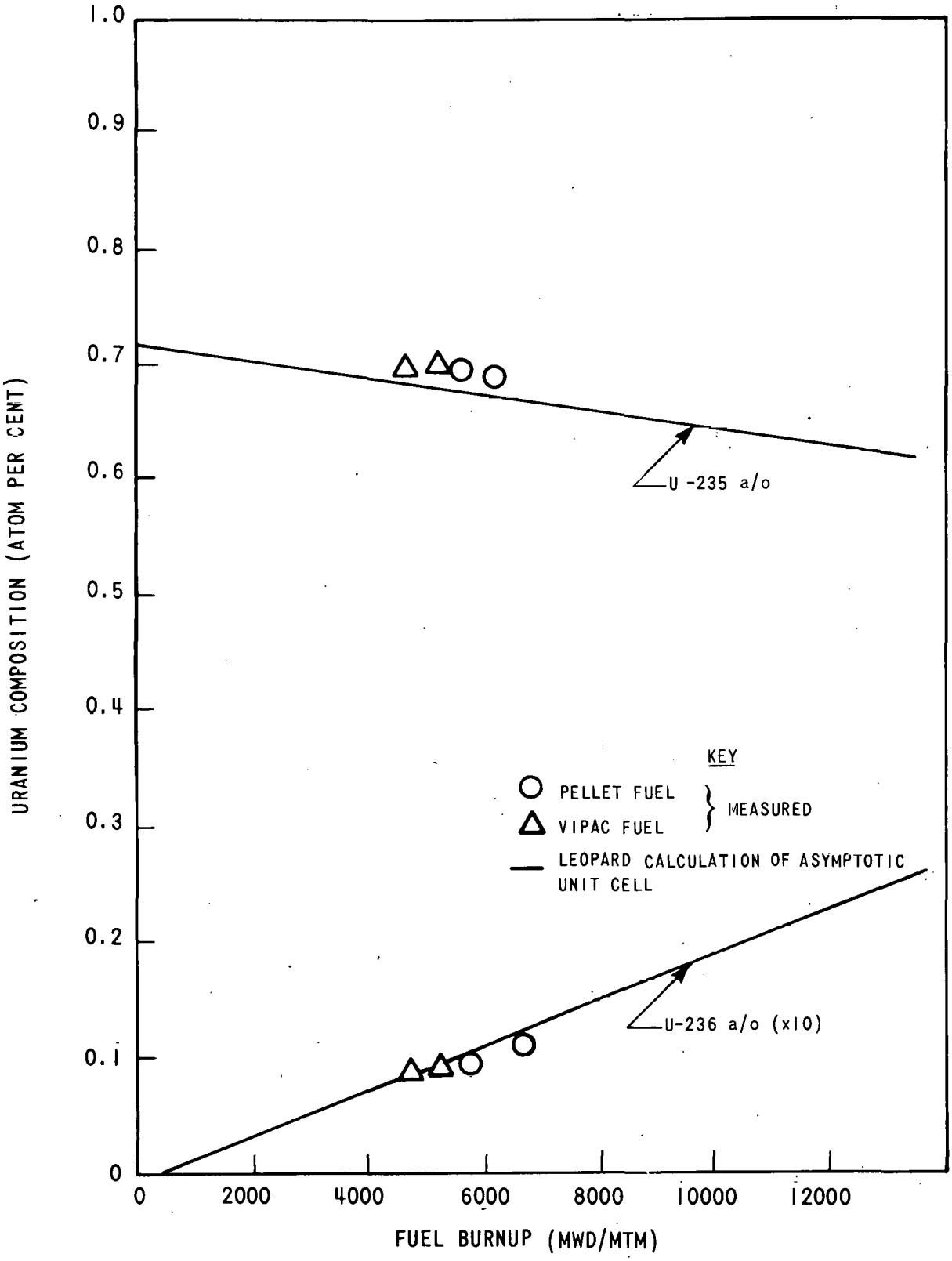

Figure 3-1. Variation of Uranium Isotopic Abundance with Fuel Burnup

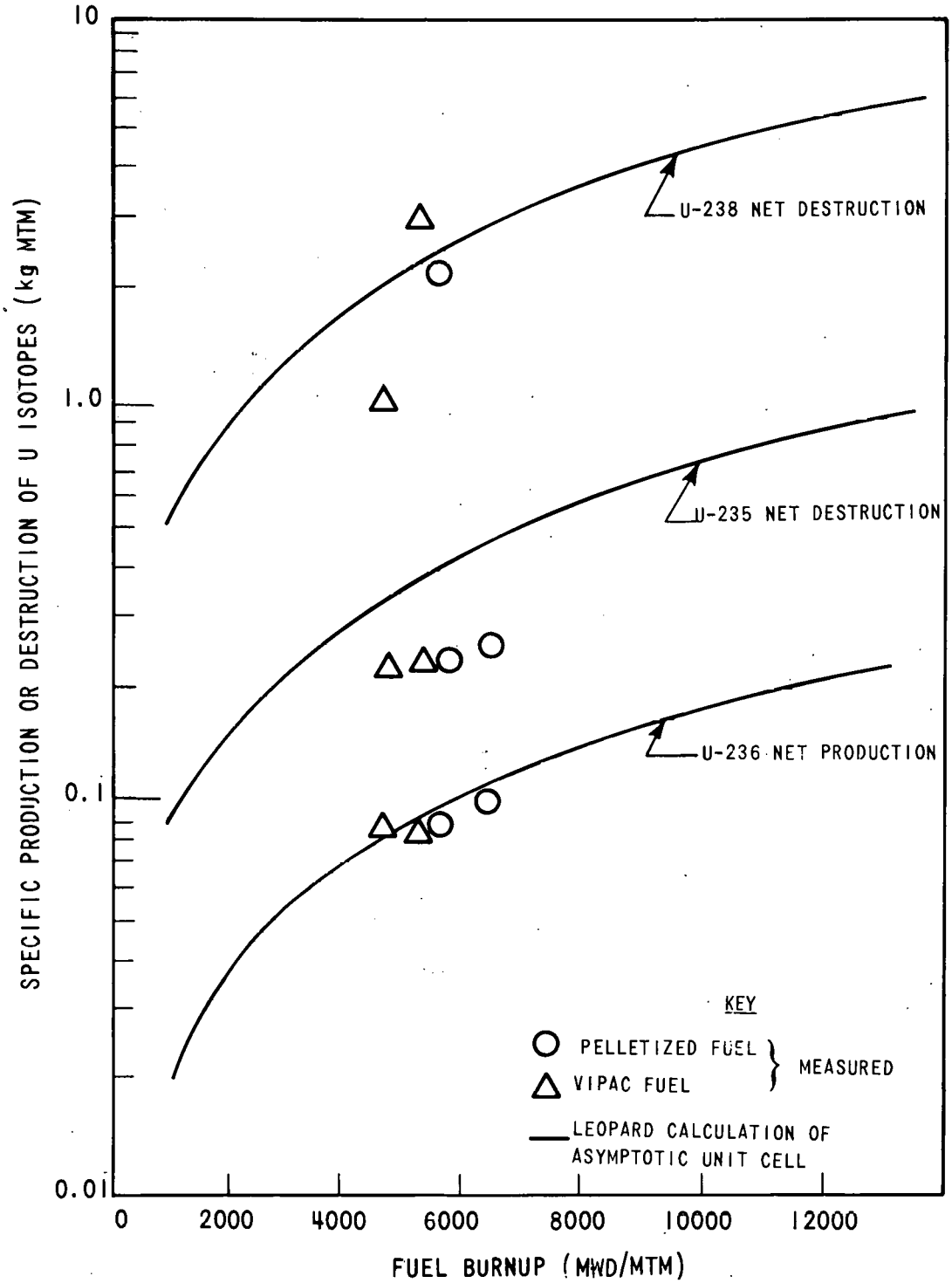

Figure 3-2. Specific Production and Destruction of Uranium Isotopes Versus Fuel Burnup

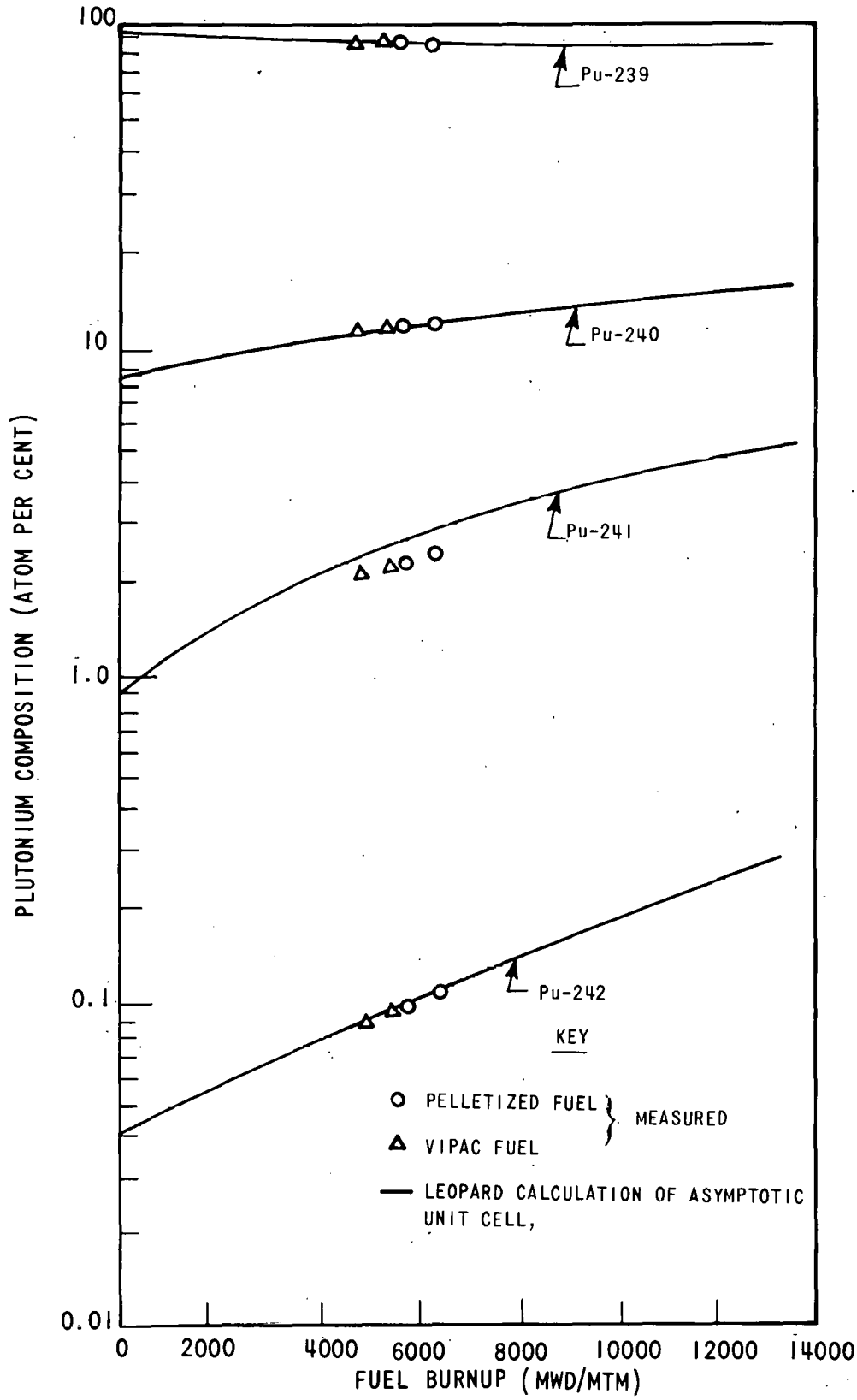

Figure 3-3. Variation of Plutonium Isotopic Abundance with Fuel Burnup

and underpredicts the Pu-242 abundance by about two percent, relative. Figure 3-4 shows the net destruction of Pu-239 and the net production of Pu-240, Pu-241, and Pu-242. The results are consistent with the Pu abundance data plotted on Figure 3-3.

Agreement between experiment and calculation may be fortuitous since the samples were exposed to mixed neutron spectra, whereas the calculation considers an asymptotic neutron spectrum.

1.5 Pu/U Mass Ratio

The Pu/U mass ratio obtained from mass spectrometric analysis is compared with corresponding values obtained by X-ray fluorescence analysis in Table 3-3. Two of the four X-ray fluorescence analyses appear to be inconsistent. The basis for this conclusion is the plot of Pu/U mass ratio versus burnup shown on Figure 3-5 where the two X-ray data points are out of line with the other data. With these exceptions, the measured and calculated Pu/U mass ratio data are in general agreement. The scatter in the data, however, is greater than would be expected from quoted measurement uncertainties. As above, a valid comparison between measurement and calculation cannot be made due to spectral differences.

2. OPERATIONS ANALYSIS

2.1 Operations Follow

The reactivity follow of the core (critical boron concentration versus burnup) is being conducted using the FOLLOW code. This code, developed with Westinghouse funds, provides an automated procedure whereby boron concentration measurements made with variations in moderator temperature, pressure, reactor power, and control rod positions are adjusted to a corresponding critical boron requirement for nominal operating conditions. Thus, data are determined for a consistent set of operating conditions as a function of burnup.

For Saxton, nominal operating conditions are 23.5 MWt, 530°F, 2000 psi and all control rods withdrawn. Burnup is continuously updated from the measured

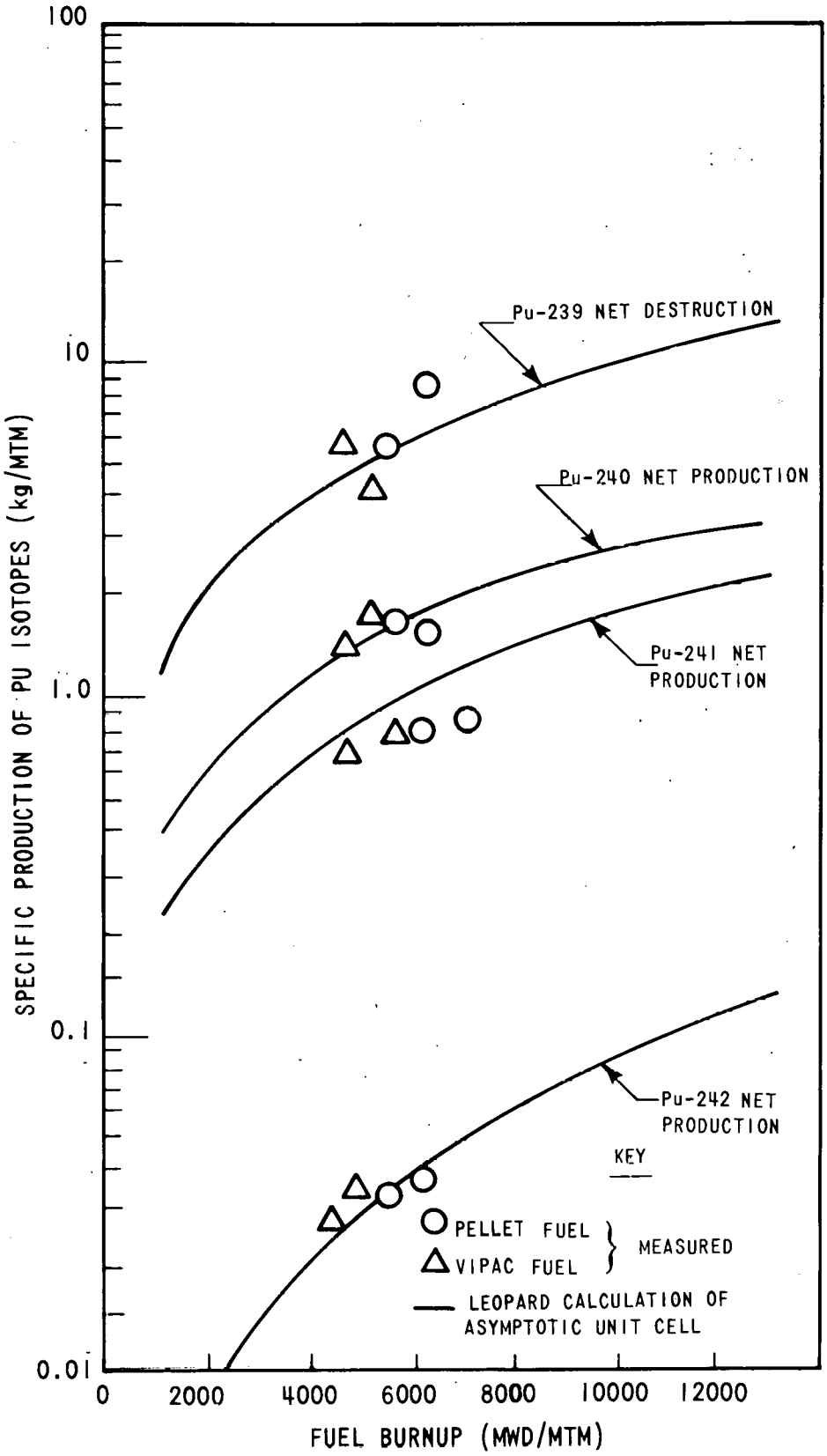

Figure 3-4. Specific Production and Destruction of Plutonium Isotopes Versus Fuel Burnup

TABLE 3-3

COMPARISON OF PU/U MASS RATIO DATA OBTAINED FROM MASS SPECTROMETRIC AND X-RAY FLUORESCENCE ANALYSES

Sample	Type of Fuel	Pu/U Mass Ratio		$\frac{\text{X-Ray - Mass Spec.}}{\text{Mass Spec.}} \times 100$
		Mass Spec.	X-ray	
X-BU-1	Vipac	0.0688	0.0502	-27.0
X-BU-2	Vipac	0.0672	0.0498	-25.9
D-BU-1	Pelletized	0.0650	0.0685	+ 5.4
D-BU-2	Pelletized	0.0677	0.0638	- 5.8

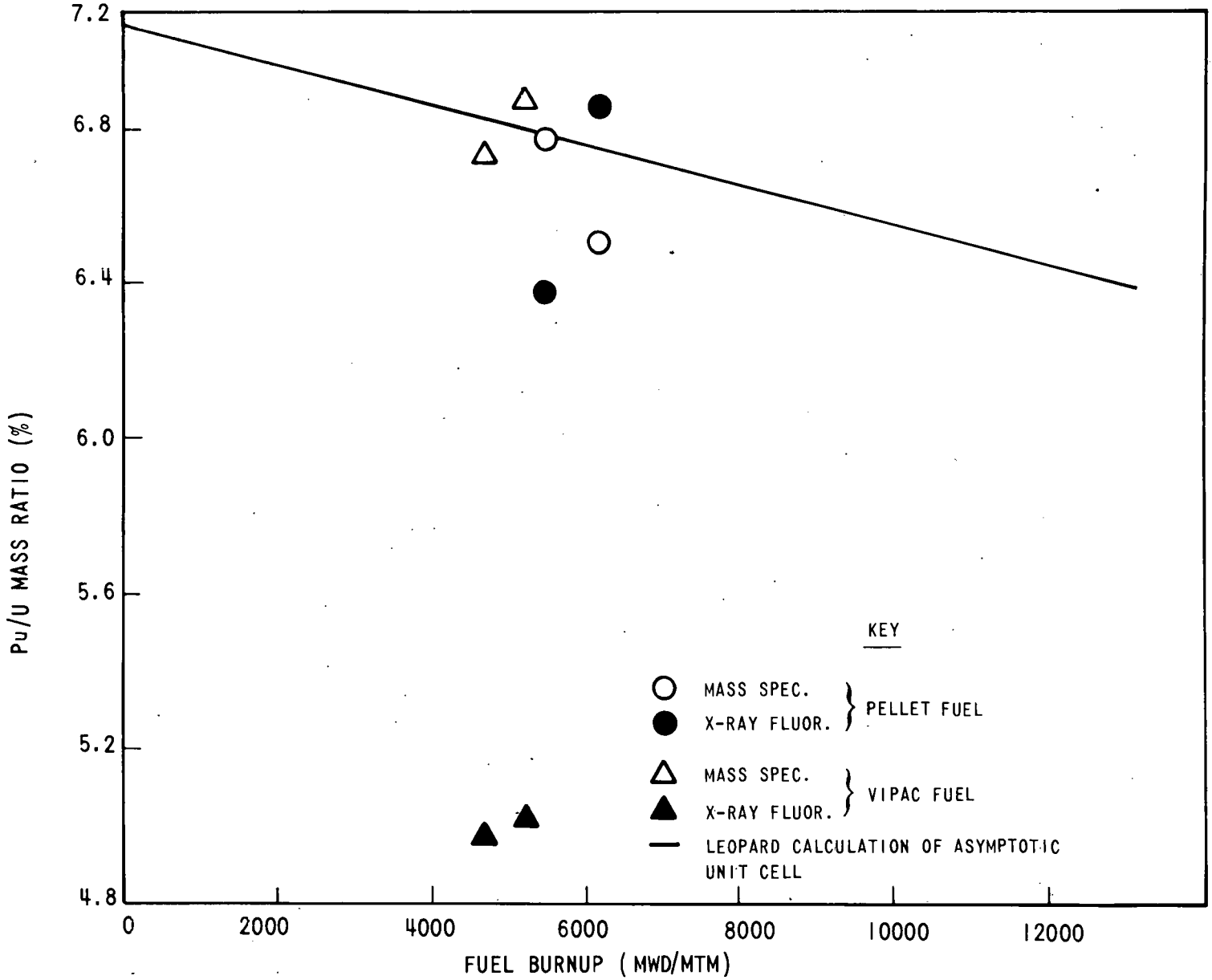

Figure 3-5. Variation of Pu/U Mass Ratio with Fuel Burnup

total energy release of the core. This procedure provides a substantial increase in the number of data points that can be processed. Figure 3-6 shows the critical boron concentration as a function of core average burnup in MWD/MTM through shutdown at a core average burnup of 7390 MWD/MTM.

2.2 Depletion Analysis

During the period reported upon, an extensive re-evaluation was made of the depletion and burnup history of Saxton Core II. Calculations of the remaining core life and core power distributions were made after experimental results from power operation were available. The just-critical boron concentrations from these calculations are shown in Figure 3-6 together with the experimental depletion data. In these calculations, the irradiation history of each fuel assembly was known from power distribution measurements, and was used to determine group constants for use in the PDQ-3 calculations.

The depletion analysis contains fission product cross sections from LEOPARD which are based on the correlation of depletion data from a uranium reactor. Therefore, the effect of differences in fission product yields from plutonium fissions and uranium fissions are neglected. While this latter effect is small early in life, its importance increases with burnup. The CINDER[6] code determines an effective fission product cross section by following the buildup and depletion of individual isotopes and equates them to a single multi-group fission product cross section set expressed in barns per fission. Therefore, a basic purpose of this re-evaluation was to determine whether the more rigorous method of calculating fission product cross sections, using CINDER, results in an improvement in the correlation of the experimental depletion data.

Since the effects of non-uniform axial depletion are not considered in the two-dimensional PDQ-3 calculations, a correction for the reactivity difference between uniform and non-uniform axial depletion of the core is applied. The axial correction that has been applied was reported previously[2] as an effective axial buckling which, when input to PDQ-3, would reproduce the calculated reactivity difference between a uniform and non-uniform one-dimensional axial calculation. Because the reactivity effect of non-uniform

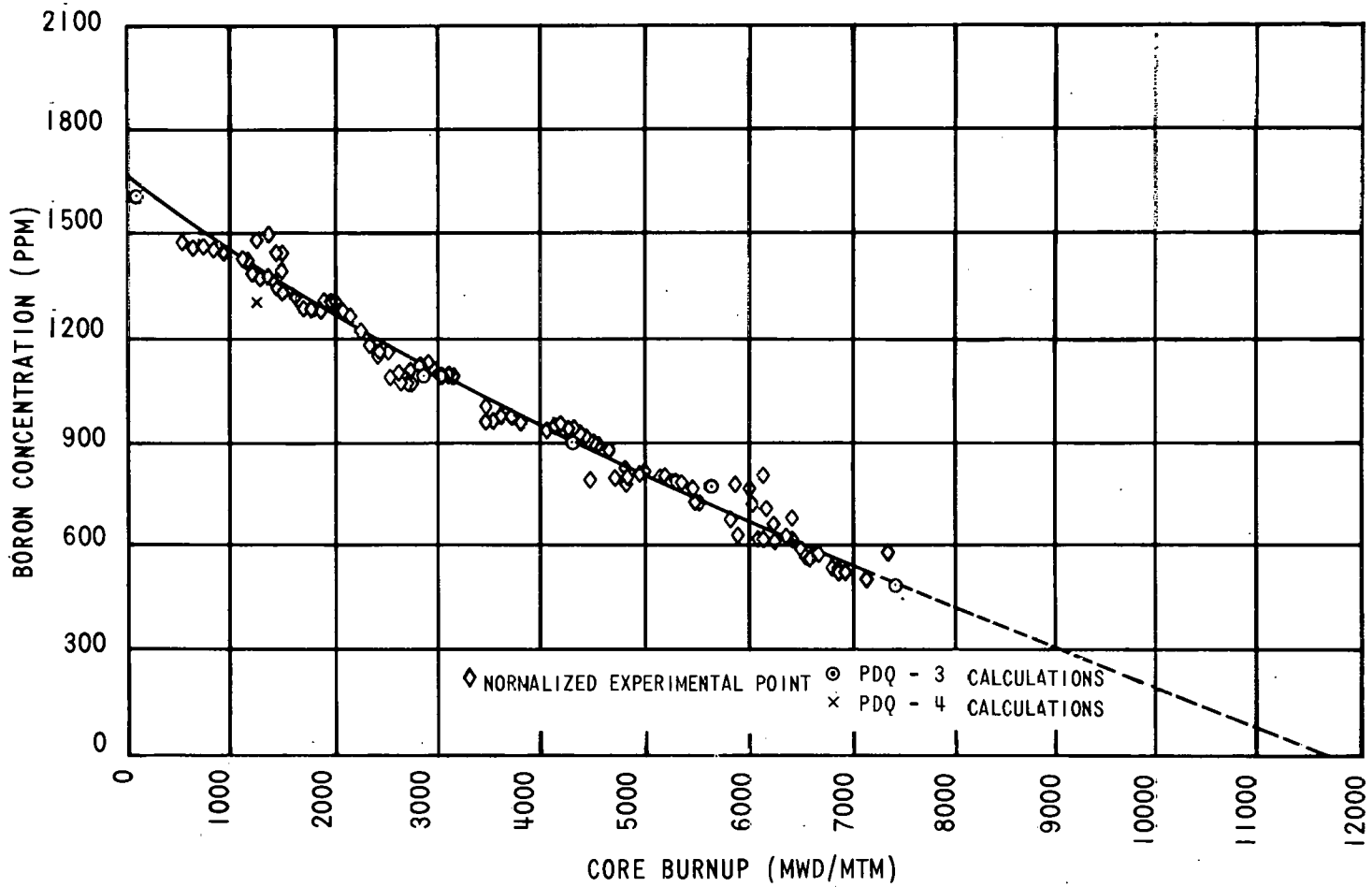

Figure 3-6. Boron Concentration (PPM) as a Function of Core Burnup (MWD/MTM)

axial depletion is larger than the reactivity effect of differences in the fission product treatment, a more rigorous calculation of the axial correction was performed, using the PANDA code, prior to a more exact evaluation of the fission product treatment. This code permits a large number of mesh points, pointwise water density effects, and pointwise xenon effects. These options were not available in the prior calculation method. The reactivity difference determined in this calculation for the two fuels is shown in Figure 3-7. These results are in good agreement with those determined previously.

Further, the calculations show that at a Saxton Core II burnup of 7300 MWD/MTM, the reactivity difference between LEOPARD and CINDER fission products is equivalent to approximately 35 ppm in the critical boron concentration. This difference is too small to establish definitively whether the more exact fission product treatment contained in CINDER or the use of the simple empirical equations contained in LEOPARD provides better agreement with the experimental data. As shown in Figure 3-6, the analysis using PDQ-3 with LEOPARD fission product cross sections is in good agreement with that observed experimentally. From these initial comparisons, it appears that the CINDER treatment may overestimate fission product poisoning effects.

2.3 Revised Depletion Analysis Method

During the report period work was also begun on a revised diffusion theory depletion analysis for Saxton Core II that will follow local burnup effects more accurately than the current LEOPARD - BUBBLE - PDQ-3 sequence.

The revised analysis is being carried out with a LEOPARD-PDQ-7-HARMONY[7] sequence. In this sequence, the diffusion theory group constants were generated using improved plutonium yields for iodine, xenon, samarium and promethium. The required LEOPARD calculations were made using the actual operating conditions of the core which includes a slightly lower moderator temperature than that specified in the original design operating conditions. New fuel temperatures for the PuO_2-UO_2 were generated with the latest thermal calculation models. A new homogenization method which more accurately represents the control rod followers was adopted.

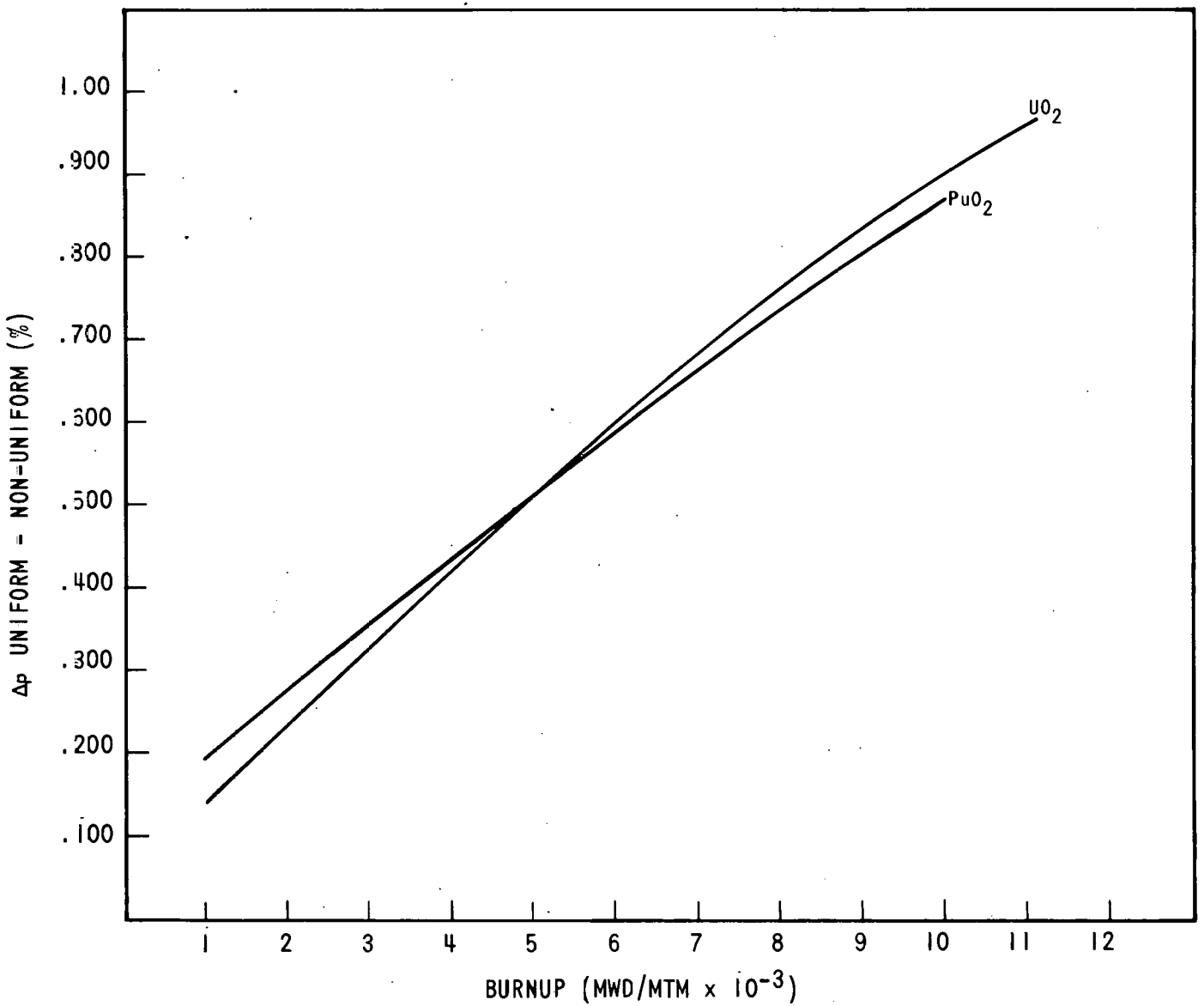

Figure 3-7. Reactivity Difference Between Uniform and Non-Uniform Depletion - 6.6 w/o PuO_2 and 5.7 w/o UO_2 Fuel

In the PDQ-7-HARMONY sequence, burnup-dependent macroscopic diffusion theory group constants are input in the form of a burnup matrix. Power and accumulated burnup is computed at each point. The burnup at each point is then used to find new group constants from the burnup matrix. The boron concentration is changed separately at each time step. This work is still in progress, but will be completed before the end of the next report period.

SECTION 4

SAX-520, THERMAL AND HYDRAULIC ANALYSIS

(R. A. Dean and W. Bezella)

An analysis of the fission gases related from two typical SAXTON PuO_2-UO_2 fuel rods was completed. Two rods were evaluated, one containing 94 percent dense pellet fuel and the other containing 87 percent dense vipac fuel material. The experimentally-determined fission gas release results were 2.1 percent and 10.8 percent for the pelletized and vipac fuel, respectively. The fission gas release predictions, using the FIGHT code,[8] yielded percentage releases of 1.5 percent for the pellet fuel rod and 9.0 percent for the vipac fuel rod. The results of this analysis are summarized in Table 4-1.

The data tabulated in Table 4-1 represent the final estimates of the gas release. These predicted gas release values are lower than the values reported previously.[9] They reflect the effect of an improved more realistic thermal analysis which resulted in reductions in fuel temperatures and hence, release of fission gases. They also reflect changes in the logic of the computer code which result in a more realistic representation of the fission gas release. In the case of the pellet fuel, an improved clad creep model reduced the peak fuel temperature about 60°F. For the vipac fuel analysis, a constant fuel-to-cladding gap conductance of 1000 Btu/hr-ft^2-°F was employed[10] to provide a more realistic estimate of the fuel temperature. This constant gap conductance along with the revised clad creep model resulted in a peak fuel temperature reduction of almost 600°F. The strong dependence of the modified diffusion type fission gas release model with fuel temperature, as well as the logic changes, explain the resulting reduction in predicted gas release.

TABLE 4-1

RESULTS OF SAXTON PuO_2-UO_2 TEST RODS FISSION GAS EVALUATION

	Pelletized Fuel Rod	Vipac Fuel Rod
Fuel Rod Designation	D & E	X-1 & X-5
Average Rod Burnup, MWD/MTM	4530	4670
Total Irradiation Time, Days	134	134
Time Average Fuel Maximum Temp., °F	1975	2065
Time Average Volume Average Fuel Temp., °F	1243	1219
Predicted Percentage Fission Gas Release, %*	1.5	9.0
Observed Percentage Fission Gas Release, %	2.1	10.8

* Based Upon: $\frac{30 \text{ Atoms (Kr \& Xe)}}{100 \text{ Fissions}}$

Modified diffusion release model

Activation Energy = 87.2 kcal/mole

Threshold Temperature = 800°C

Diffusion Constant, D' (1400°C)$(\text{sec})^{-1}$ $= 7.5 \times 10^{-9}$ for pellet

$= 1.5 \times 10^{-7}$ for vipac

SECTION 5

SAX-630, POST-IRRADIATION EXAMINATION

(D. T. Galm, E. S. Schwartz, J. A. Corbett)

The result of the post-irradiation examination of Saxton Core rods has been reported in the previous Semi-Annual Progress Reports.[1,2]

SECTION 6

SAX-660, MATERIALS EVALUATION

(K. R. Jordan and C. J. Kubit)

The materials evaluation of the four low-burnup rods, as defined in the current work program, is complete and was reported in the previous Semi-Annual Report.[2] Thus, no further work is scheduled until the final post-irradiation examination of the fuel.

SECTION 7

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