**Cover Sheet for a Hanford
Historical Document
Released for Public Availability**

Released 1994

**Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830**

**Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute**

DECLASSIFIED

HW-83193

SERIES AND COPY NO.

GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

DATE _____

June 30, 1964

X

THIS DOCUMENT CONTAINS RESTRICTED DATA AS
DEFINED IN THE ATOMIC ENERGY ACT OF 1954
IT IS TRANSFERRED OR DISCLOSED OF
CONTENTS TO ANY MANNER TO AN UNAUTHORIZED
PERSON IS PROHIBITED

TITLE

PHYSICS CALCULATIONS FOR MIXED

PuO₂-UO₂ NPR LOADINGS

7

OTHER OFFICIAL CLASSIFIED INFORMATION

THIS MATERIAL CONTAINS INFORMATION AFFECTING THE NATIONAL DEFENSE OF THE UNITED STATES WITHIN THE MEANING OF THE ESPIONAGE LAWS, TITLE 18, U. S. C., SECS. 793 AND 794, THE TRANSMISSION OR REVELATION OF WHICH IN ANY MANNER TO AN UNAUTHORIZED PERSON IS PROHIBITED BY LAW.

AUTHOR

C. L. Bennett

ISSUING FILE

THIS DOCUMENT MUST NOT BE LOANED, REPRODUCED OR HAVE AN UNAUTHORIZED PERSON HAVE ACCESS TO IT. WHEN THE DOCUMENT MUST BE STORED IN AN UNCLASSIFIED AREA, THE INFORMATION MUST BE CLASSIFIED BY THE PERSON RESPONSIBLE FOR KEEPING AND CONTROLLING THE INFORMATION. THIS PROCESS IS FROM AN AUTHORIZED PERSON'S PERMISSION, AND STORAGE OF THE INFORMATION IS FROM THE RELATIONSHIP OF THE INFORMATION. AND PERSONS HAVING ACCESS TO THE INFORMATION ARE RELEVANT TO THE SPACE PROVIDED BELOW.

ROUTE TO:

PAYROLL NO.

LOCATION

FILE	ROUTE	DATE
100-443888-100	100-443888-100	100-443888-100

SIGNATURE AND DATE

REFERENCE COPY
RECEIVED SOC AREA

~~JUL 26 1964~~

RETURN	20
--------	----

TECHNICAL INFORMATION FILES

53011

203

APR 18 1958

2-11-65

MASTER

54-3000-340 (3-57)

AEC-GE RICHLAND, WASH

(CLASSIFICATION)

DECLASSIFIED

DECLASSIFIED

HW-83193

This document classified by:

HA Snyder

Classification Cancelled and Changed To

DECLASSIFIED

By Authority of DS Lewis

CG-PR-2, 4-8-94

By DK Hanson, 4-19-94

Verified By Jerry Mully, 4-19-94

PHYSICS CALCULATIONS FOR MIXED

PuO₂-UO₂ NPR LOADINGS

by

C. L. Bennett

June 30, 1964

[REDACTED]

DISTRIBUTION

Copy

1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18

C. L. Bennett
R. A. Bennett
G. J. Busselman
F. G. Dawson
P. L. Hofmann
M. C. Leverett
E. E. Mills
W. S. Nechodom
R. V. Poe
E. G. Pierick
J. W. Riches
R. J. Shields
300 File
Record Copy

DECLASSIFIED

PHYSICS CALCULATIONS FOR MIXED PuO_2 - UO_2 NPR LOADINGS

I. INTRODUCTION

At the request of NRD (NPR Physics Subsection) a study was initiated to determine the physics characteristics of various plutonium-uranium composites as fuel for the NPR. From this study, the PuO_2 - UO_2 fuel system was selected to receive major attention. The effect of adding a burnable poison, B-10, in intimate contact with the mixed oxide fuel was also to be considered. The present report summarizes the results of these investigations.

The studies were confined to a water cooled, graphite moderated NPR lattice employing a tube-in-tube fuel element geometry with zirconium cladding. The plutonium isotopic composition, used in the study, was 84 a/o Pu-239 and 16 a/o Pu-240, while that of the uranium (depleted) was 0.22 w/o U-235 and 99.78 w/o U-238.

To assess the accuracy of the methods employed, a series of calculations was carried out for the present NPR fuel geometry with 0.947 w/o U-235 enriched uranium. Various geometry idealizations were examined and comparisons to the accepted value of 1.056 for cold, clean k_∞ were made. (1)

Section II contains a description of the methods employed. Section III contains the results of the calculations for the standard fueled NPR and a description of the various calculational models considered. Section IV lists the results of the PuO_2 - UO_2 fuel calculations and contains a short discussion of the results.

DECLASSIFIED

II. METHODS

A. k_{∞} Calculations

The infinite multiplication constant of the cell is determined by means of a one-dimensional, four group, diffusion calculation using the HFN code.⁽²⁾ The following group structure is used:

TABLE I - HFN GROUP STRUCTURE

<u>Group</u>	<u>Lethargy Limits</u>
1	0 - 6.75
2	6.75 - 15.25
3	15.25 - 16.5
4	16.5 - ∞

The small third group is chosen to accomodate the 1.06 ev Pu-240 absorption resonance. The thermal group constants are obtained from the thermalization code TEMPEST.⁽³⁾ For most of the calculations the Wilkins heavy moderator equation is used to generate the thermal spectrum. The epithermal and fast group constants are obtained from the slowing-down code GAM.⁽⁴⁾ The Bell correction⁽⁵⁾ is employed to correct for fuel rod shadowing.

B. Burnup Calculations

The lifetime physics statics characteristics of the cell are obtained from the burnup code ZODIAC.⁽⁶⁾ ZODIAC combines GAM, TEMPEST, and HFN with a one-group isotopic transmutation code NORMAL MODES⁽⁷⁾ and a cross section averaging code CLERK.⁽⁸⁾ This chain of codes allows multiregion burnup calculations to be carried out automatically. Spectrum dependent group constants are recalculated at discrete time steps.

DECLASSIFIED

Low cross-section fission product poisoning is treated by means of a pseudo fission product representation based on reference (9).

C. Reactor Model

Since the two fuel regions of the tube-in-tube geometry are depleted at different rates, the fuel cross-sections should be averaged over spectra representative of the individual fuel volumes and associated moderator regions. On this basis - for the purpose of cross-section averaging - the heterogeneous cell was divided into three parts: an inner fuel region with surrounding clad and coolant; an outer fuel region with surrounding clad, coolant and process tube; and a graphite moderator region. The regions within each part were then homogenized.

Separate GAM-TEMPEST calculations are carried out for each part to obtain 4-group microscopic group constants. These cross-sections are then used in the HFN, 1-dimensional, 4-group diffusion calculations with full cell geometry details.

While this choice of model is fairly arbitrary, it has several advantages:

- 1) Separate U-238 resonance integrals for the inner and outer fuel regions can be obtained.
- 2) Microscopic cross sections, averaged over different spectra characteristic of the separate burnable fuel regions, can be obtained as a function of fuel composition, and hence reactor life.
- 3) The separate contributions of graphite and water to the moderator temperature coefficient can be estimated more easily. This is an important consideration since the

DECLASSIFIED

graphite and water temperature are grossly different at full power operation.

It is important to point out that the accuracy of the model remains to be tested. This should be done via critical experiments and by means of more refined calculations.

III. NPR CALCULATIONS FOR STANDARD FUEL

The standard fueled NPR affords an opportunity to do at least some testing of the methods and calculation models employed in this study. This test is not really adequate since the present NPR does not contain Pu, while $\text{PuO}_2\text{-UO}_2$ fueling of NPR is assumed in the study. Nevertheless, at least a partial check is obtained.

The NPR lattice dimensions, temperatures and densities are given in Table II.

TABLE II: NPR LATTICE FOR STANDARD FUEL

Region Number	Region Material	Hot Lattice			Cold Lattice (20°C)	
		Outside Radius (cm)	Effective Density (Gm/cc)	Temperature (°C)	Outside Radius (cm)	Effective Density (gm/cc)
1	H ₂ O	0.5601	0.839	225	0.5563	1.000
2	Zr	0.6287	7.621	243	0.6248	7.621
3	U	1.4948	18.260	366	1.4808	18.624
4	Zr	1.6002	6.706	236	1.5862	6.706
5	H ₂ O	2.2530	0.846	220	2.2403	1.000
6	Zr	2.3228	7.021	239	2.3099	7.021
7	U	3.0112	18.258	337	2.9896	18.624
8	Zr	3.0772	6.895	235	3.0556	6.895
9	H ₂ O	3.4455	0.846	220	3.4392	1.000
10	Zr	4.1440	6.400	250	4.1377	6.400
11	Void	4.2418	0.0	567	4.2418	0.0
12	Graphite	12.1603	1.312	541	12.1603	1.312

NOTE: Data in the above table were obtained from N Reactor Physics. The data contain corrections for cross coolant tubes in the graphite, thermal expansions of the lattice, slug spacings, and slug end caps.

DECLASSIFIED

A series of k_{∞} calculations for the present NPR cell were carried out with a succession of models used for generating cross-section averages. These models are described below.

Method 1-A - Single TEMPEST, Single HRG* - 0th Iterations

The complete NPR cell is treated as one single homogeneous region and the multigroup diffusion constants are obtained from a single TEMPEST and a single HRG* run. The moderator temperature used in the TEMPEST calculation is obtained by weighting the water and graphite temperatures by their slowing down power ($\xi\Sigma_s$).

In the calculation of resonance parameters for HRG, the mean chord length in the absorber lump $\bar{l}$, is defined as follows:

$$\bar{l} = \frac{4V}{S_{\text{eff}}}$$

where V is the volume of the absorber lump and S_{eff} is the effective surface of the absorber lump. For the tube-in-tube geometry, S_{eff} is determined from a formula by Richey. (10)

$$S_{\text{eff}} = S_0 + 4 \left[\Sigma_1 V_1 (1 - P'_{11}) - \Sigma_2 V_2 (1 - P'_{22}) \right]$$

where subscript 1 refers to the innermost coolant region, and subscript 2 refers to the coolant channel between the concentric fuel tubes. S_0 is the surface area of the outer fuel tube and P'_{11} is the collision probability in region 1 for a neutron uniformly generated in the same region without previously crossing any region interior to region 1. Implicit in the above definition of $\bar{l}$ is the assumption that the fuel is black to resonance neutrons. P'_{11} was obtained using a cylindrical Monte Carlo code. (11)

* Hanford Revised GAM (Unpublished)

DECLASSIFIED

Method 1-B - Single TEMPEST, Single HRG - 1st Iterations

This method is an iteration on Method 1-A. The diffusion theory results using Method 1-A are utilized to generate thermal, spatial, shielding factors which are used in turn to modify the effective nuclear densities in the thermal cross-section calculation. The new thermal cross-sections are then used in a second diffusion theory run to obtain a new k_{∞} .

Method II - Double TEMPEST - Single HRG

This method consisted of dividing the cell into two homogeneous regions for the TEMPEST runs and retaining one homogeneous region (same as in Method 1-A and 1-B) for the HRG run. This method has the advantage, compared to methods 1-A and 1-B, of eliminating the necessity of using a single weighted moderator temperature in the TEMPEST calculation.

Method III - Triple TEMPEST - Triple HRG

This is the method used for the $\text{PuO}_2\text{-UO}_2$ study. It is described in Section II.

Method IV - Triple TEMPEST - Triple HRG

This method only differs slightly from Method III. The spectra calculated in TEMPEST for the two homogeneous regions containing fuel and H_2O coolant are obtained via the Wigner-Wilkins light moderator equation. In Methods I - III the Wilkins equation is applied throughout.

Since some of the individual temperature coefficients are known with some confidence from experimental measurements⁽¹²⁾⁽¹³⁾⁽¹⁴⁾ various temperature dependent diffusion calculations were run for each of the above described models. The results are given in Table III below. In these calculations, effective temperature interaction between the water and graphite cell regions was ignored.

DECLASSIFIED

TABLE III: TEMPERATURE DEPENDENT INFINITE MULTIPLICATIONS CONSTANTS
STANDARD NPR FUEL

Water Constants Graphite Constants Fuel Constants	Method <u>IA</u>	Method <u>IB</u>	Method <u>II</u>	Method <u>III</u>	Method <u>IV</u>
C C C	1.0494000	1.0505316	1.0361922	1.0710353	1.0752243
C C H	1.0422828		1.0291696	1.0645350	1.0687054
C H C			1.0470993	1.0824302	1.0863442
H C C			1.0149286		
H H H	1.0208628	1.0217477	1.0213496	1.0590481	1.0633635
H H C	1.0282652		1.0265313	1.0659299	1.0702755
H C H	1.0305119		1.0098107	1.0469311	1.0515753
C H H	1.0327162		1.0399985		

NOTE: The above numbers are reported to seven places to facilitate temperature coefficient calculations. They do not imply absolute accuracy to this number of significant figures.

C = Cold temperature; H = hot temperature.

DECLASSIFIED

As stated in Section I the best value, to date, of the cold, clean, k_{∞} is 1.056 as reported by P. F. Nichols and Staff in Reference (1). This compares to an experimental number measured in the PCTR of 1.075 ± 0.003 as reported in Reference (12) which was later corrected to 1.065 ± 0.003 .⁽¹³⁾ The calculated value using Method III obtained by the author was 1.071. Using the same method a fast fission factor of 1.041 was calculated as compared to 1.047 ± 0.001 also measured in the PCTR.⁽¹²⁾ The moderator (graphite) temperature coefficient obtained with Method III was $+ 0.023 \text{ mk}/^{\circ}\text{C}$ which is identical to the value quoted in reference (15) (hot "green" case) which is obtained from measured values.⁽¹⁴⁾ The Doppler coefficient is calculated to be $-2.1 \times 10^{-5}/^{\circ}\text{C}$ ($20^{\circ} \rightarrow 350^{\circ}\text{C}$ range). This compares with $-2.3 \times 10^{-5}/^{\circ}\text{C}$ given in reference (15).

The Δk_{∞} from cold to hot obtained by the author was - 12 mk compared to an expected value⁽¹⁵⁾ of about - 8 mk. Other temperature coefficients given in reference (15) cannot be easily compared with the Table III results.

IV. RESULTS: $\text{PuO}_2\text{-UO}_2$ Fuel System

The NPR lattice characteristics for the $\text{PuO}_2\text{-UO}_2$ fuel system is shown in Table IV.

DECLASSIFIED

TABLE IV: NPR LATTICE FOR MIXED OXIDE FUEL

<u>Region Number</u>	<u>Region Material</u>	<u>Outside Radius (cm)</u>	<u>Effective Densities</u>	
1	H ₂ O	0.5563	Graphite	1.30 g/cm ³
2	Zr	0.6223	H ₂ O (hot)	0.82 "
3	Oxide	1.4821	Zr	6.50 "
4	Zr	1.5862	PuO ₂ (95% TD)	10.90 "
5	H ₂ O	2.2403	UO ₂ (95% TD)	10.42 "
6	Zr	2.3063		
7	Oxide	2.9896	<u>Temperatures (Hot)</u>	
8	Zr	3.0556	Oxide	800°C
9	H ₂ O	3.4417	Coolant	232°C
10	Zr	4.1402	Graphite	550°C
11	Graphite	12.1603		

Required PuO₂ enrichments for the depleted UO₂ were determined with and without burnable poison (B-10) mixed in intimate contact with the fuel. In addition to the "no B-10" case, two concentrations of B-10 were chosen for investigation: 0.01 w/o B-10 and 0.02 w/o B-10. Loading curves (see Figure 1) were constructed to obtain PuO₂ enrichments which would give a hot clean infinite multiplication of ~ 1.06 . The three cores thus obtained were fueled with: 0.5 w/o PuO₂, 99.5 w/o UO₂; 2.0 w/o PuO₂, 98 w/o UO₂, 0.01 w/o B-10; and 3.6 w/o PuO₂, 96.4 w/o UO₂, 0.02 w/o B-10.

The results of the burnup calculation are summarized in Figures 2-8. Figure 2 shows the hot, k_{∞} of the cell plotted against Full Power Hours (FPH). A full power hour is defined as one hour of exposure at the

DECLASSIFIED

flux required to maintain a constant full power operation. Since just one cell is being considered, its specific power is assumed to be that of the average tube. For NPR this is 113.8 KW/ft.⁽¹⁵⁾ The burnup data for the three cores considered is tabulated in the Appendix. The relative atom densities of the plutonium isotopes versus FPH are shown in Figures 3, 4, and 5.

In Figures 6, 7, and 8 the net Pu-239 transformation is separated into its two components: the loss of the original Pu-239 and the gain of Pu-239 from U-238 burnup; losses of U-238 atoms from fission being neglected.

The temperature coefficients can be obtained from Table V.

TABLE V: TEMPERATURE DEPENDENT k_{∞} 's, PuO₂-UO₂ CORE

<u>Water Temp. and Density</u>	<u>Graphite Temp.</u>	<u>Fuel Temp.</u>	<u>0.5 w/o PuO₂- No Boron</u>	<u>2.0 w/o PuO₂- 0.01 w/o B</u>	<u>3.6 w/o PuO₂- 0.02 w/o B</u>
Cold	Cold	Cold	1.0191682	1.0462734	1.0663444
Cold	Cold	Hot	1.0093222		
Cold	Hot	Cold	1.0332084		
Hot	Cold	Cold	1.0519439		
Hot	Hot	Hot	1.0562315	1.0574394	1.0601632
Hot	Hot	Cold	1.0669493		
Hot	Cold	Hot	1.0413664		
Cold	Hot	Hot	1.0232383		

NOTE: The above numbers are reported to seven places to facilitate temperature coefficient calculations. They do not imply absolute accuracy to this number of significant figures.

DECLASSIFIED

The results show that the use of B-10 as a burnable poison yields very favorable reactivity - time characteristics. On the other hand, the net temperature coefficients for some cores are calculated to be positive.

The net temperature coefficients of the 0.5 w/o PuO_2 - "No Boron" case is substantially positive, but as the Pu and B loading is increased the positive net temperature effect decreases in magnitude and, in fact, becomes negative for the 3.6 w/o PuO_2 - 0.02 w/o B core.

This temperature coefficient behavior is due to interactions of the non $1/v$ Pu cross sections, the spectrum dependent alpha of Pu and the spectral hardening of the Pu-B¹⁰ system. The exact mechanisms coupling these effects remain to be fully analyzed. The temperature coefficient calculations do not include the Pu Doppler effects. Furthermore, no exposure dependent temperature coefficient calculations have been carried out.

While the calculational methods employed appear to give reasonable results for the standard fueled NPR, no such direct experimental check of the PuO_2 - UO_2 fuel system is possible at present.

V. ACKNOWLEDGEMENT

The author is indebted to Dr. P. L. Hofmann for his technical advice, and to R. H. Holeman and Mrs. D. Adrian for having carried out some of the numerical work for this report.

DECLASSIFIED

VI. REFERENCES

- (1) Nichols, P.F. and Reactor Physics Staff, "Monthly Technical Report, February, 1964, Reactor Physics," HW-82335, 2-28-64.
- (2) Lilley, J.R., "Computer Code HFN - Multi-group, Multi-region Neutron Diffusion Theory in One Space Dimension," HW-71545, 10/30/61.
- (3) Shudde, R.H. and J. Dyer, NAA Program Description, TEMPEST - A Neutron Thermalization Code," September 1960.
- (4) Joanou, G.D. and J.S. Dudek, "GAM-I: A Consistent P-1 Multi-group Code for the Calculation of Fast Neutron Spectra Multi-group Constants," GA-1850 (1961)
- (5) Bell, G.E., "A Simple Treatment for Effective Resonance Absorption Cross Sections in Dense Lattices," Letter to the Editor, N.S.E. 5, 75-77 (1959).
- (6) Matsumoto, D.D. and G.J. Busselman, "ZODIAC User's Manual 03-30-64," HW-81302, 3/30/64.
- (7) Duane, B.H., "Variational Optimum Kinetic Normal Modes," Physics Research Quarterly Report, October, November, December, 1962, HW-76128, January 1963.
- (8) Busselman, G.J., "CLERK - A Cross Section Averaging Routine," HW-75521, 11/12/63.
- (9) Nephew, E.A., "Thermal and Resonance Absorption Cross Sections of U-233, U-235, and Pu-239 Fission Products," ORNL-2869.
- (10) Richey, C. R. and L.L. Carter, "Lattice Parameter Correlation for a Concentric Tube-in-Tube Fuel Element," Physics Research Quarterly Report, April, May, June, 1963. HW-77871, July 15, 1963.
- (11) Toffer, Hans, Private Communication.
- (12) Johnson, D.L., G.W.R. Endres, K.R. Birney, D.E. Wood, and E.Z. Block, "Lattice Parameter Measurements for N-Reactor," HW-72696, November 1963. (Confidential)
- (13) Davenport, L.C., "Measured Differences in k_{∞} Between the N-Lattice and the PCTR Mockup," HW-80731, February 10, 1964. (Confidential)
- (14) Smith, R.I., "The Variation of k_{∞} With Fuel Temperature for the NPR," HW-69343. April 21, 1961. (Confidential)
- (15) Research and Engineering Section of N-Reactor Department, "100-N Technical Manual, Vol. 1," HW-69000, March 5, 1961. (Confidential)

DECLASSIFIED

VII. APPENDIX

Tabulation of Burnup Data

[REDACTED]

DECLASSIFIED

TABLE A I : BURNUP DATA; 0.5 w/o PuO₂ FUEL

Full Power Hours	k_{∞}	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242
0	1.0569	5.1588-5 5.1588-5		2.3082-2 2.3082-2	9.7229-5 9.7229-5	1.8520-5 1.8520-5		
699	1.0246	4.8739-5 4.8056-5	4.8449-7 5.8650-7	2.3071-2 2.3069-2	8.9922-5 8.9061-5	2.2588-5 2.3157-5	1.9843-6 2.3185-6	3.9260-8 5.6300-8
1401	0.9969	4.5855-5 4.4556-5	9.6245-7 1.1531-6	2.3059-2 2.3055-2	8.3444-5 8.2120-5	2.6171-5 2.7077-5	3.9669-6 4.6178-6	1.6576-7 2.2907-7
2104	0.9731	4.2985-5 4.1136-5	1.4349-6 1.7032-6	2.3047-2 2.3040-2	7.7806-5 7.6319-5	2.9296-5 3.0348-5	5.8779-6 6.7857-6	3.7924-7 5.2802-7
2809	0.9528	4.0156-5 3.7831-5	1.8922-6 2.2276-6	2.3034-2 2.3025-2	7.2975-5 7.1554-5	3.1991-5 3.3051-5	7.6679-6 8.7679-6	6.8510-7 9.5634-7
3517		3.7396-5 3.4676-5	2.3350-6 2.7178-6	2.3021-2 2.3009-2	6.8908-5 6.7691-5	3.4284-5 3.5263-5	9.3123-6 1.0526-5	1.0893-6 1.5077-6

NOTE: Nuclear concentrations in nuclei/barn-cm.; top line refers to inner fuel region and bottom line refers to outer fuel region.

DECLASSIFIED

HM-83193

-14-

TABLE A II: BURNUP DATA; 2.0 w/o PuO₂, 0.01 w/o B FUEL

Full Power Hours	k_{∞}	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	B-10
0	1.0574	5.0839-5 5.0839-5		2.2747-2 2.2747-2	3.9082-4 3.9082-4	7.4442-5 7.4442-5			6.2762-5 6.2762-5
2062	1.0539	4.9220-5 4.8244-5	3.8980-7 5.4785-7	2.2728-2 2.2724-2	3.5949-4 3.3159-4	8.3068-5 9.0607-5	9.0968-6 1.2716-5	8.8948-8 2.2911-7	5.2924-5 4.6426-5
4118	1.0524	4.7485-5 4.5452-5	7.8574-7 1.0979-6	2.2709-2 2.2700-2	3.2677-4 2.7834-4	9.1530-5 1.0308-4	1.7929-5 2.4535-5	3.8157-7 9.8667-7	4.3625-5 3.2845-5
6170	1.0513	4.5608-5 4.2467-5	1.1840-6 1.6662-6	2.2689-2 2.2675-2	2.9303-4 2.3160-4	9.9635-5 1.1195-4	2.6318-5 3.4821-5	9.4621-7 2.3597-6	3.4966-5 2.2054-5
8217	1.0486	4.3578-5 3.9295-5	1.5950-6 2.2313-6	2.2669-2 2.2649-2	2.5893-4 1.9161-4	1.0713-4 1.1738-4	3.4038-5 4.3107-5	1.8317-6 4.3855-6	2.7084-5 1.3947-5
10261	1.0429	4.1357-5 3.5972-5	2.0182-6 2.8006-6	2.2648-2 2.2622-2	2.2533-4 1.5837-4	1.1367-4 1.1966-4	4.0837-5 4.9124-5	3.1113-6 7.0694-6	2.0112-5 8.2445-6
12303		3.8939-5 3.2545-5	2.4529-6 3.3526-6	2.2626-2 2.2594-2	1.9325-4 1.3154-4	1.1888-4 1.1915-4	4.6496-5 5.2833-5	4.8425-6 1.0354-5	1.4199-5 4.5263-6

NOTE: Nuclear concentrations in nuclei/barn-cm.; top line refers to inner fuel region and bottom line refers to outer fuel region.

DECLASSIFIED

TABLE A III : BURNUP DATA; 3.6 w/o PuO₂, 0.02 w/o B FUEL

Full Power Hours	k _∞	U-235	U-236	U-238	Pu-239	Pu-240	Pu-241	Pu-242	B-10
0	1.0602	5.0048-5 5.0048-5		2.2393-2 2.2393-2	7.0099-4 7.0099-4	1.3352-4 1.3352-4			1.2561-4 1.2561-4
2101	1.0589	4.9080-5 4.8515-5	2.7781-7 3.7310-7	2.2376-2 2.2373-2	6.7340-4 6.3021-4	1.3991-4 1.5175-4	9.4527-6 1.4747-5	3.9600-8 1.3674-7	1.1523-4 1.0632-4
4199	1.0582	4.8093-5 4.6922-5	5.5334-7 7.4050-7	2.2359-2 2.2353-2	6.4462-4 5.6275-4	1.4640-4 1.6758-4	1.8747-5 2.9107-5	1.6513-7 5.7745-7	1.0519-4 8.8806-5
6294	1.0579	4.7083-5 4.5272-5	8.1938-7 1.1086-6	2.2342-2 2.2332-2	6.1453-4 4.9902-4	1.5300-4 1.8096-4	2.7869-5 4.2784-5	3.8396-7 1.3674-6	9.5489-5 7.3036-5
8385	1.0580	4.6049-5 4.3556-5	1.0761-6 1.4674-6	2.2325-2 2.2311-2	5.8306-4 4.3936-4	1.5969-4 1.9184-4	3.6791-5 5.5479-5	7.1052-7 2.5513-6	8.6115-5 5.9000-5
10473	1.0582	4.4979-5 4.1763-5	1.3335-6 1.8267-6	2.2308-2 2.2290-2	5.5018-4 3.8416-4	1.6646-4 2.0017-4	4.5472-5 6.6917-5	1.1663-6 4.1733-6	7.7060-5 4.6682-5
12557		4.3875-5 3.9895-5	1.5916-6 2.1960-6	2.2291-2 2.2268-2	5.1590-4 3.3373-4	1.7326-4 2.0594-4	5.3865-5 7.6834-5	1.7622-6 6.2552-6	6.8333-5 3.6078-5

NOTE: Nuclear concentrations in nuclei/barn-cm.; top line refers to inner fuel region and bottom line refers to outer fuel region.

DECLASSIFIED

DECLASSIFIED

Figure 1

DECLASSIFIED

Figure 2

Normalized Atomic Density

DECLASSIFIED

Figure 5

Figure 6

Normalized Atom Density

DECLASSIFIED

Pu-239 LOSS AND GAIN DURING FUEL LIFE
2.0 w/o PuO_2 , 0.01 w/o B-10

LEGEND

Inner Fuel Region
Outer Fuel Region

- (1) Net Pu-239 Burnout
- (2) Loss of Original Pu-239
- (3) Gain of Pu-239 from U-238

Figure 7

FPH ($\times 10^{-3}$)

Normalized Atom Density

DECLASSIFIED