APR 25 1966

GEAP-4819
AEC RESEARCH AND
DEVELOPMENT REPORT
SEPTEMBER 1965

MASTER

EXPERIMENTAL STUDIES OF
TRANSIENT EFFECTS IN
FAST REACTOR FUELS
SERIES III
PRE-IRRADIATED MIXED OXIDE
(PuO_2 - UO_2) IRRADIATIONS
INTERIM REPORT
STEADY-STATE IRRADIATIONS

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

J. E. HANSON
S. A. RABIN

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 10

ADVANCED PRODUCTS OPERATION

GENERAL ELECTRIC

SAN JOSE, CALIFORNIA

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

GEAP-4819
AEC Research and
Development Report
September 1965

EXPERIMENTAL STUDIES OF TRANSIENT EFFECTS IN FAST REACTOR FUELS

CFSTI PRICES

SERIES III

H.C. \$ 4.00; MN 1.00

PRE-IRRADIATED MIXED OXIDE (PuO_2 - UO_2) IRRADIATIONS

INTERIM REPORT STEADY-STATE IRRADIATIONS

J. E. Hanson
S. A. Rabin

APPROVED:

J. H. Field, Manager
Fuel Element and Component Testing

F. J. Leitz, Project Engineer
Fast Ceramic Reactor Development Program

U. S. Atomic Energy Commission
Contract AT(04-3)-189
Project Agreement 10

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

*Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia
Price: \$4.00 per copy*

2631-APO-97
165-AG-3/66

ADVANCED PRODUCTS OPERATION
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or*
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.*

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

TABLE OF CONTENTS

	<u>Title</u>	<u>Page</u>
	ABSTRACT	xi
SECTION I	SUMMARY AND CONCLUSIONS	1-1
SECTION II	INTRODUCTION	2-1
SECTION III	RESULTS	3-1
	3.1 Irradiation History	3-1
	3.2 RML Examinations	3-1
SECTION IV	DISCUSSION OF RESULTS	4-1
	4.1 Fuel Burnup Determinations	4-1
	4.2 Rod Power Determinations	4-5
	4.3 $\int k d\theta$ Determinations	4-8
	4.4 Correlation of Microstructure with Rod Power	4-9
	4.5 Consequence of Axial Redistribution	4-11
APPENDIX A	FUEL SPECIMEN AND CAPSULE DETAILS	A-1
	A.1 Series 3 Capsule Design	A-1
	A.2 Series 3 Fuel Specimen Design and Fabrication	A-5
APPENDIX B	SAMPLE HEAT TRANSFER CALCULATIONS	B-1
	B.1 Basis	B-1
	B.2 Capsule Dimensions	B-1
	B.3 Capsule Materials	B-1
	B.4 Film Coefficient	B-1
	B.5 Gamma Heating of Components	B-2
	B.6 Linear Heat Rates, Including Gamma Heating	B-3
	B.7 Capsule Surface Heat Flux	B-3
	B.8 Capsule Surface Temperature	B-3
	B.9 Internal Temperatures	B-4
	B.10 Fuel Surface Temperature for 29.3 kW/ft Rod Power	B-6
APPENDIX C	CYCLE INTEGRATED RELATIVE FLUX CURVES	C-1
APPENDIX D	POWER GENERATION VERSUS TIME CURVES	D-1

TABLE OF CONTENTS (Continued)

	<u>Title</u>	<u>Page</u>
APPENDIX E	MULTICHANNEL PULSE HEIGHT ANALYZER RESULTS ON CAPSULE C3D	E-1
	ACKNOWLEDGMENTS	-1-
	REFERENCES	-3-

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3-1	Series 3 Capsule Cross Section	3-2
3-2	Gross Gamma Scan of Capsule C3A	3-6
3-3	External Appearance of Fuel Pin C3A Following Steady-State Irradiation	3-7
3-4	Gross Gamma Scan of C3A Fuel Pin	3-8
3-5	Specimen C3A Following Steady-State Irradiation (C3A Transverse and Metallography Sections)	3-11
3-6	Transverse Cross Section - Specimen C3A	3-13
3-7	Longitudinal Sections (as-polished), and Alpha and Beta-Gamma Autoradiographs of Lower End of Specimen C3A	3-15
3-8	Composite Photomicrograph of Lower Fuel Region of Specimen C3A	3-17
3-9	Longitudinal Sections of Specimen C3A at Upper End of Fueled Region	3-19
3-10	Composite Photomicrograph of Upper Fuel Region of Specimen C3A	3-21
3-11	Photomicrograph of Fuel - Insulator Area Showing Grey Phase	3-25
3-12	Longitudinal Section of Specimen C3A Near Center of Fueled Region	3-27
3-13	Gross Gamma Scan of Capsule C3D	3-29
3-14	Close-up of C3D Fuel Pin at Failed End	3-30
3-15	End-on View of C3D Fuel Pin at Failed End	3-31
3-16	Transverse and Longitudinal Sections of Specimen C3D	3-33
3-17a	Specimen C3D - Microstructure of Cladding at Rupture	3-37
3-17b	Specimen C3D - Microstructure of Cladding above Failure Region	3-37
3-18	Bottom of Fuel Pin C3D at Varying Axial Positions Obtained by Consecutive Grinding	3-39
3-19	Panoramic View of Transverse Section from C3D	3-43
3-20	Bottom Fuel Section	3-45
3-21	Metallic Particle in Top of Fuel Specimen C3D	3-47
3-22	Gamma Scan of Metallographic Specimen with Metallic Particle	3-49
3-23	GETR Pool Position Axial Flux Shapes	3-50
3-24	Gross Gamma Scan of Capsule C3B	3-54
3-25	Gross Gamma Scan of Capsule C3C	3-55
3-26	Pinhole Camera Autoradiograph of Bottom of C3C Fuel Pin	3-56
3-27	Gross Gamma Scan of Capsule C3E	3-58
3-28	Pinhole Camera Autoradiograph of Bottom of C3E Fuel Pin	3-59
4-1	Capsule Aluminum Heat Sink ΔT versus Linear Power Generation	4-3
4-2	$\int_0^{T_C} k d' \quad$ versus Linear Power for Specimens C3A and C3B	4-7

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
4-3	$\int_0^{T_c} k d\theta$ versus Linear Power for Specimens C3C, C3D, and C3E. with Indicated Flux Depression	4-10
C-1	Integrated Cycle Average Relative Flux - Cycle 42	C-2
C-2	Integrated Cycle Average Relative Flux - Cycle 44	C-3
C-3	Integrated Cycle Average Relative Flux - Cycle 45	C-4
C-4	Integrated Cycle Average Relative Flux - Cycle 46	C-5
C-5	Integrated Cycle Average Relative Flux - Cycle 47	C-6
C-6	Integrated Cycle Average Relative Flux - Cycle 48	C-7
C-7	Integrated Cycle Average Relative Flux - Cycle 49	C-8
C-8	Integrated Cycle Average Relative Flux - Cycle 50	C-9
C-9	Integrated Cycle Average Relative Flux - Cycle 51	C-10
C-10	Integrated Cycle Average Relative Flux - Cycle 52	C-11
C-11	Integrated Cycle Average Relative Flux - Cycle 53	C-12
D-1	Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 42	D-3
D-2	Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 44	D-5
D-3	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 45	D-7
D-4	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 46	D-9
D-5	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 47	D-11
D-6	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 48	D-13
D-7	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 49	D-15
D-8	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 50	D-17
D-9	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 51	D-19
D-10	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 52	D-21
D-11	Series III - Irradiations in GETR - Linear Power Generations versus Time - Cycle 53	D-23

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
E-1	Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3D	E-2
E-2	Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3B	E-4
E-3	Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3C	E-5
E-4	Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3E	E-6

4

W. J.

d

W. J.

d

d

W. J.

d

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
3-1	Steady-State Irradiation History of Series III Capsules Based on Thermocouple Data	3-3
3-2	Steady-State Irradiation History of Series III Capsules Based on Physics Calculations	3-4
3-3	Dimensional Measurements - Specimen C3A	3-9
3-4	Gas Composition in Fuel Specimen C3A	3-23
3-5	Dimensional Measurements Specimen C3D	3-35
3-6	Chromatographic Analysis of Gases from Specimen C3D	3-51
3-7	Mass Spectrometric Analysis of Xenon and Krypton from Specimen C3D	3-53
4-1	Comparison of Fuel Specimen Burnup Determinations	4-1
4-2	Adjusted Fuel Specimen Burnup Values	4-4
4-3	Cycle Average and Peak Relative Flux Values for 11 to 17 Inches Above Core Bottom	4-6
A-1	Data for Series 3 Fuel Pins	A-8
A-2	Isotopic Analysis of Plutonium Feed Solution for Series 3 Fuel Pellets	A-9
A-3	Pressing and Sintering Data for Series 3 Pellets	A-10
E-1	Gamma Spectrum Analysis Capsule C3D	E-3
E-2	Gamma Spectrum Analysis Capsule C3B	E-7
E-3	Gamma Spectrum Analysis Capsule C3C	E-8
E-4	Gamma Spectrum Analysis Capsule C3E	E-10

2

4

7

8

9

4

ABSTRACT

The irradiation performance of five 0.250-inch-diameter mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel specimens is presented and discussed.

The burnup attained by two specimens was about 8000 MWd/T_e . The three remaining specimens attained goal exposures of about 65,000 MWd/T_e .

One low burnup and one high burnup specimen were subjected to destructive examinations. The remaining specimens were examined by means of nondestructive techniques prior to re-encapsulation for transient irradiation. The transient behavior will be the subject of a future report.

x

g

4

4

4

SECTION I

SUMMARY AND CONCLUSIONS

Five 0.250-inch-diameter, 28 wt percent PuO_2 - 72 wt percent UO_2 mixed oxide fuel specimens were irradiated in the General Electric Test Reactor at high linear rod powers to various burnups. Two specimens, one of low burnup (8,700 MWd/ T_e) and one of high burnup (65,000 MWd/ T_e), were destructively examined. The remaining three specimens (7,900, 64,000, and 63,000 MWd/ T_e , respectively) were nondestructively examined by means of gamma scanning and pinhole camera gamma radiography.

On the basis of these examinations and analysis of the data generated during the irradiations it was concluded that:

1. Central fuel melting occurred in all of the specimens.
2. No gross plutonium migration occurred during the irradiations.
3. Some axial fuel relocation was observed at the bottom end of all of the fuel columns. Relocation of the fuel into the hollow insulator pellet below the fuel is believed to have occurred by melting of the fuel. However, the grain structure of the redistributed fuel was not definitive enough that it could be stated unequivocally whether relocation of the fuel occurred by melting or vapor deposition.
4. Failure of the high burnup specimen C3D was attributed to locally excessive fuel expansion caused by a combination of fuel transport and excessively high local rod powers.
5. An ingot of metallic fission products found at the top of the fuel column in specimen C3D is believed to have been formed by a "zone-refining" process which occurred during the irradiation.
6. Substantial evidence of the mobility of fission products sometimes considered to be relatively immobile (zirconium, niobium, lanthanum, barium) was found in all the high burnup specimens.
7. The temperatures at which equiaxed and columnar grain growth occurred in the oxide fuel was estimated to be about 1600 and 1900°C, respectively.
8. On the basis of the nondestructive examinations, it was concluded that specimens C3B, C3C, and C3E were suitable for transient irradiation experiments.

SECTION II

INTRODUCTION

In the development of fuel for the Fast Ceramic Reactor⁽¹⁾ it is necessary to evaluate fuel cladding performance under nonstandard operating conditions. This requirement stems from the fact that the excursion-limiting Doppler effect is dependent on maintaining fuel cladding integrity during an accidental power excursion. Thus, the definition of fuel cladding performance and failure limits is vital to the establishment of operating and design criteria for the reactor.

Several failure mechanisms for fuel and cladding subjected to a power excursion have been postulated.⁽²⁾ The more important of these mechanisms, with respect to performance limits, are the large increase in volume attendant with melting fuel,⁽³⁾ the pressure stresses arising from steady-state fission gas pressure, and the effect on the fuel of retained fission gases that are suddenly released as a result of the large and rapid increase in fuel temperature attendant with a power excursion.

An experimental program to evaluate the transient performance of mixed oxide fuel is in progress. The initial work, Series I irradiations,⁽⁴⁾ was concerned with the transient performance of large (~1-inch diameter) non-preirradiated fuel rods of UO_2 . This work was directed more specifically toward fuel development for the Experimental Fast Oxide Reactor⁽⁵⁾ which is now known as SEFOR*. Series II irradiations were concerned with the investigation of the fuel cladding transient performance of 0.250-inch-diameter mixed oxide fuel with no burnup. This work has been reported in GEAP-4804.⁽⁶⁾

The Series III irradiations that are the subject of this report consist of two phases:

- a. Steady-state irradiations, and
- b. Transient irradiations.

It is the purpose of this report to present the data obtained during the steady-state irradiation, and destructive and nondestructive examinations, of five 0.250-inch-diameter mixed oxide fuel specimens which were irradiated in pool positions in the General Electric Test Reactor (GETR).

The results from the transient irradiation of the pre-irradiated specimens will be presented in a subsequent report.

*Southwest Experimental Fast Oxide Reactor Development Program.

.

.

.

.

.

.

SECTION III

RESULTS

Five 0.250-inch-diameter mixed oxide fuel specimens (designated C3A, C3B, C3C, C3D, and C3E) were irradiated in pool positions in the GETR to goal exposures of about 8,000 MWd/T_e (specimens C3A and C3B), and about 65,000 MWd/T_e (specimens C3C, C3D, and C3E). Two of these capsules, C3A and C3D, were destructively examined in the Radioactive Materials Laboratory (RML) of the Vallecitos Atomic Laboratory. The three remaining capsules are to be re-encapsulated for transient irradiation in the TREAT facility⁽⁷⁾ as part of the program to evaluate fuel cladding transient performance of pre-irradiated fuel.

3.1 IRRADIATION HISTORY

As seen in Figure 3-1, which is a schematic cross section of the capsule*, instrumentation for the steady-state irradiations consisted of four thermocouples located at the axial midplane of the fuel specimen. The temperature data obtained from these thermocouples were used to calculate the linear heat generation rates presented in Table 3-1. Integration of the instantaneous linear heat generation rates over the GETR operating cycle provided an estimate for the fuel burnup. These data are also given in Table 3-1.

In addition to the continuous thermocouple data, physics calculations made at the end of each irradiation cycle also provided a source of fuel specimen performance and burnup data. The results of these calculations are given in Table 3-2. Although the burnup data based on physics calculations were originally given in terms of MWd/T_e (mixed oxide) based on 180 MeV per fission, they have been converted to MWd/T_e (U + Pu) based on 200 MeV per fission in order to be consistent with Table 3-1. The reasoning for selection of these particular quantities is discussed in subsection 4.1 of this report.

3.2 RML EXAMINATIONS

Capsules C3A and C3D were subjected to destructive post-irradiation examinations. The remaining three capsules, C3B (low burnup), C3C, and C3E (high burnup), were examined by nondestructive techniques to evaluate the condition of the fuel and cladding prior to transient irradiation.

*Details of fuel specimens and capsule design are given in Appendix A.

Figure 3-1. Series 3 Capsule Cross Section

TABLE 3-1

STEADY-STATE IRRADIATION HISTORY OF SERIES III
CAPSULES BASED ON THERMOCOUPLE DATA

GETR Cycle Number	Peak Rod Power (kW/ft)					Average Rod Power (kW/ft)					Burnup (MWd/T _e)*				
	Capsule					Capsule					Capsule				
	C3A	C3B	C3C	C3D	C3E	C3A	C3B	C3C	C3D	C3E	C3A	C3B	C3C	C3D	C3E
42	29.3	25.1	--	--	--	20.4	18.1	--	--	--	9,200	8,100	--	--	--
44	--	--	27.9	28.8	30.8	--	--	19.7	19.7	20.5	--	--	7,100	7,100	7,300
45	--	--	27.7	28.6	24.0	--	--	21.9	22.4	17.6	--	--	8,100	8,400	6,500
46	--	--	26.7	24.3	25.8	--	--	19.2	19.2	18.2	--	--	7,200	7,200	6,800
47	--	--	27.3	29.2	23.9	--	--	20.1	19.9	17.8	--	--	6,700	6,600	5,900
48	--	--	27.5	26.0	26.0	--	--	20.1	20.9	20.1	--	--	7,600	7,900	7,600
49	--	--	27.9	31.7	23.1	--	--	19.6	22.1	16.9	--	--	9,000	10,200	7,800
50	--	--	26.9	27.5	22.3	--	--	19.9	21.9	18.1	--	--	8,600	9,500	7,800
51	--	--	25.1	24.9	22.2	--	--	18.6	19.3	17.7	--	--	6,100	6,300	5,700
52	--	--	23.1	22.2	22.2	--	--	18.6	17.8	18.7	--	--	6,400	6,200	6,500
53	--	--	25.3	25.0	20.0	--	--	17.0	17.6	20.2	--	--	6,500	6,800	7,800
												73,000	76,000	70,000	

* MWd/T_e = megawatt days per metric ton of Pu + U at 200 MeV/fission.

Peak values. Average over 6-inch fuel length is ~ 1.5 percent less.

TABLE 3-2

STEADY-STATE IRRADIATION HISTORY OF SERIES III
CAPSULES BASED ON PHYSICS CALCULATIONS

GETR Cycle Number	Peak Rod Power (kW/ft)					Cycle Average Rod Power (kW/ft)					Burnup (MWd/T _e)*				
	Capsule					Capsule					Capsule				
	A	B	C	D	E	A	B	C	D	E	A	B	C	D	E
42	23	22	--	--	--	18.2	17.0	--	--	--	7,900	7,600	--	--	--
44	--	--	17.4	20.2	21.3	--	--	14.0	16.2	17.1	--	--	4,200	4,800	5,100
45	--	--	20.9	23.2	19.9	--	--	16.8	18.6	16.0	--	--	5,100	5,600	4,800
46	--	--	21.5	22.2	22.1	--	--	17.2	17.7	17.7	--	--	5,100	5,200	5,200
47	--	--	21.9	23.1	22.7	--	--	18.3	19.3	18.9	--	--	5,100	5,400	5,300
48	--	--	19.9	22.1	22.1	--	--	16.1	18.0	18.0	--	--	5,200	5,800	5,800
49	--	--	20.5	21.9	20.5	--	--	17.1	18.3	17.0	--	--	7,100	7,600	7,100
50	--	--	20.1	22.1	20.0	--	--	16.8	18.4	16.7	--	--	6,700	7,300	6,600
51	--	--	18.8	19.8	19.2	--	--	15.7	16.5	16.1	--	--	5,000	5,100	5,100
52	--	--	18.8	18.3	20.0	--	--	15.5	15.1	16.5	--	--	4,800	4,700	5,200
53	--	--	17.1	16.4	19.5	--	--	14.2	13.7	16.3	--	--	5,400	5,200	6,200
												54,000	57,000	56,000	

* MWd/T_e = megawatt days per metric ton of Pu + U at 200 MeV/fission.

3.2.1 Capsule C3A (Low Burnup Specimen) Destructive Examination

As seen in Table 3-1, capsule C3A was irradiated for one GETR cycle, and attained a burnup of about 9,200 MWd/T_e. Post-irradiation examination of the capsule was initiated by performing a gross gamma scan to determine the extent of fuel movement prior to sectioning the fuel pin. As seen in Figure 3-2, the gamma scan indicated that no gross fuel movement had occurred. Disassembly of the capsule revealed that the external appearance of the fuel pin was virtually unchanged, as evidenced by Figure 3-3. Gamma scanning of the bare fuel pin (Figure 3-4) further indicated that no gross fuel movement had occurred. Pre- and post-irradiation diameter and profile measurements as given in Table 3-3 revealed that no change in cladding diameter and only a small amount of bowing had occurred during the irradiation of the radially-unrestrained fuel pin. Sectioning of the fuel pin and selection of the metallographic samples were as shown in Figure 3-5.

As seen in Figure 3-5, the fuel structure is generally typical of irradiated mixed oxide. Shown in Figure 3-6 is an etched microstructure of a transverse cross section, illustrating equiaxed and columnar grain growth, and lenticular pore migration.

Figure 3-7 illustrates an as-polished longitudinal section at the lower end of the fuel column, together with autoradiographs of the same section. The lower insulator pellets in these experiments were cored to provide communication between the fueled zone and the pressure transducer installation diaphragm in the end plug. As evidenced by the fuel in the upper portion of the insulator pellet, some small amount of fuel redistribution occurred. This area is shown in more detail in an enlarged view, Figure 3-8. The appearance of the grain structure is not definitive enough to determine unequivocally if relocation of the fuel in this area occurred by melting and/or vapor deposition. However, a porous mass of oxide containing a cellular structure is generally associated with once-molten material. In addition, the similarity in the alpha activity between this plug and the fuel in the center above it suggests that it relocated en masse.

The photomicrograph depicting the fuel insulator interface (Figure 3-8) reveals that considerable diffusion proceeded across the boundary where heat generation was sufficiently high, as evidenced by the columnar grain growth which partially obliterated the original interface. The original interface is delineated by the dark etching band. The differently shaded grains in the UO₂ may be associated with the formation of facets during etching. The beta-gamma autoradiograph shows negligible activity in the interfacial zone, from which it is deduced that no gross mass transport of plutonium and fission products occurred. The alpha autoradiograph confirmed that little plutonium migration had occurred across the interface.

Similar structures to those just described were noted at the upper end of the fuel column, as seen in Figures 3-9 and 3-10. Here again the gamma activity distribution seen in the autoradiograph indicates no mass transport across the fuel insulator interface. Some upward migration of fission products is evident in the fuel cladding interface and in the two large fractures in the insulator pellet. The enlarged central void at each end of the fueled section is believed to be the result of end flux peaking during irradiation.

Figure 3-2. Gross Gamma Scan of Capsule C3A

Figure 3-3. External Appearance of Fuel Pin C3A Following Steady-State Irradiation

Figure 3-4. Gross Gamma Scan of C3A Fuel Pin

TABLE 3-3

DIMENSIONAL MEASUREMENTS - SPECIMEN C3A

Location (Inches from the Bottom)	Diameter (Inches)				Profile (Inches)			
	0°		90°		0°		90°	
	Pre-	Post-	Pre-	Post-	Pre-	Post-	Pre-	Post-
1	0.253	0.253	0.253	0.253	0.1985	0.204	0.1945	0.211
2	0.253	0.253	0.253	0.253	0.1950	0.204	0.1950	0.217
3	0.253	0.253	0.253	0.254	0.1920	0.206	0.1950	0.223
4	0.253	0.253	0.253	0.253	0.1900	0.208	0.1945	0.228
5	0.259	0.253	0.253	0.253	0.1885	0.210	0.1940	0.233
6	0.253	0.253	0.253	0.253	0.1880	0.211	0.1940	0.236
7	0.253	0.253	0.253	0.253	0.1885	0.212	0.1940	0.237
8	0.253	0.253	0.253	0.254	0.1910	0.213	0.1950	0.238
9	0.253	0.253	0.253	0.253	0.1930	0.214	0.1960	0.237
10	0.253	0.253	0.253	0.253	0.1935	0.213	0.1955	0.230
11	0.253	0.253	0.253	0.253	0.1920	0.211	0.1960	0.220
12	0.253	0.252	0.253	0.253	0.1920	0.210	0.1940	0.215
13	0.253	0.253	0.253	0.253	0.1930	0.209	0.1915	0.212
14	0.253	0.252	0.253	0.253	0.1925	0.205	0.1935	0.209
15	0.253	0.253	0.253	0.253	0.1920	0.203	0.1950	0.207

Figure 3-5 Specimen C3A Following Steady-State Irradiation (C3A Transverse and Metallography Sections)

Figure 3-6. Transverse Cross Section -
Specimen C3A (reduced to 70
percent for reproduction purposes)

(a.) PHOTOMACROGRAPH

(b.) ALPA AUTORADIOGRAPH

(c.) BETA-GAMMA AUTORADIOGRAPH

Figure 3-7. Longitudinal Sections (as-polished), and Alpha and Beta-Gamma Autoradiographs of Lower End of Specimen C3A

Figure 3-8. Composite Photomicrograph of Lower Fuel Region of Specimen C3A

Figure 3-9. Longitudinal Sections of Specimen C3A at Upper End of Fueled Region

ETCHED WITH 30% (10% HNO_3 - 20% H_2O_2)
70% H_2O FOR 60 SECONDS

1442-1

Figure 3-10. Composite Photomicrograph of
Upper Fuel Region of Specimen
C3A

The presence of a distinct, unidentified second phase was also noted in this specimen (Figures 3-10 and 3-11). This phase was found predominantly in the cooler portions of the fuel and was often located near a pellet surface. It is postulated that the second phase may be a condensed fission product or a $(U, Pu) O_x$ phase where x is other than two (see Table 3-4 for fission gas analysis). As seen in Figures 3-10 and 3-11, the phase lacks any definite morphology.

Localized cladding attack at the inner surface of the stainless steel was also found in this vicinity (see Figure 3-11). The reaction is thought to be due to oxidation by volatile fission products or impurities, and may be related to the grey phase observed at the fuel insulator interface.

Figure 3-12 is a photomacrograph of the as-polished longitudinal specimen taken near the center of the axial fuel length. This section illustrates the continuous and relatively uniform nature of the central void in the absence of any significant axial fuel movement. Dimensional checks from this figure and the transverse enlargement indicated an average central void diameter of 0.035 to 0.040 inch, corresponding to ~ 3 percent of the volume within the cladding. The diameter of columnar grain growth is approximately 0.170 inch, and the central void constitutes ~ 5 percent of the volume within this diameter.

Table 3-4 comprises the composition of the gas present in the fuel pin plenum after irradiation as determined by chromatographic analyses. It is seen that the primary constituent appeared to be air rather than helium, which may mean that the second phase seen in Figures 3-10 and 3-11 could be a complex (U, Pu, O, N) and/or fission product compound.

TABLE 3-4

GAS COMPOSITION IN FUEL SPECIMEN C3A

<u>Gas</u>	<u>Volume Percent</u>
Helium	None detected
Hydrogen	1.5
Oxygen	15.9
Nitrogen	56.2
Krypton	1.5
Xenon	24.9

The total gas volume in the pin (at STP) was determined to be 7.8 cc.

.

.

.

.

.

Figure 3-11. Photomicrograph of Fuel - Insulator Area Showing Grey Phase

Figure 3-11. Photomicrograph of Fuel - Insulator Area Showing Grey Phase

Figure 3-12. Longitudinal Section of Specimen C3A Near Center of Fueled Region

Gamma counting indicated a krypton-85 concentration of 1.95×10^{16} atoms/cc. Estimation of the atoms of krypton-85 formed on the basis of the burnup, as determined by radiochemical analyses (8900 MWd/T_e), results in a calculated fission gas release value of 15.6 percent.

3.2.2 Capsule C3D (High Burnup Specimen) Destructive Examination

Since capsule C3D operated at the highest rod power of the three high-burnup capsules during the ten-cycle irradiation (see Table 3-1), it was selected as the high burnup specimen to be subjected to destructive examination.

Preliminary post-irradiation examination of the capsule by means of gross gamma scanning and examination of the peak activities by a multichannel pulse height analysis (see Appendix E) indicated that the fuel cladding integrity had been breached. As seen in Figure 3-13, the large activity peaks at the bottom of the capsule, below the original position of the fuel, strongly indicated the presence of fuel in this location. Subsequent disassembly of the capsule revealed that the fuel pin cladding had failed. The nature of the failure was as shown in Figures 3-14 and 3-15. The location of the failure as determined by length measurements corresponds to the interface between the fuel column and the UO₂ insulator pellet, which also corresponds to the interface between the aluminum heat sink and the axial insulator can. This is shown in Figure 3-16, which also illustrates the condition of the fuel at various transverse and longitudinal sections. Table 3-5 comprises the dimensional measurements data taken before and after the irradiation. It should be noted that except at the failed end, there was no change in the pin diameter. At the failed end, the measured circumferential strain of about 19 percent showed that a significant amount of ductility was available prior to failure. This is indicative that cladding melting occurred after most of the deformation had taken place, since liquefaction would have caused a brittle failure. In Figure 3-17a, melting of the stainless steel is evidenced by the white grain boundary network and cored structure near the edges of the cladding. Melting was localized to the grain boundaries because they are segregated in compounds which possess a lower melting point than the austenitic matrix.⁽¹⁰⁾ The relatively fine grain size suggests that melting occurred rather suddenly. It is noteworthy that the cladding above the failed area exhibited only minor damage (Figure 3-17b). In other areas along the pin, a cladding attack similar to that manifested in Figure 3-11 was found.

The portion of the fuel pin below the failed area (Figure 3-18) appears to have been the most seriously affected region. Consecutive grinding and examination revealed that portions of the fuel pin cladding had melted and disappeared (Figure 3-18, a, b, c, d, e). It is apparent that molten fuel flowed into the hollow insulator pellet and that the insulator pellet lost much of its structural integrity. As seen in Figure 3-18 (f, g, h), the cladding and insulator pellet appear to have been forced against the surface of the inner wall of the thermoflex insulator can, and the wall itself was ruptured. The pressure required to exert such a force can, of course, be postulated as arising from the combination of fuel expansion, fuel vapor pressure, and fission gas pressure in conjunction with the loss of strength of the components due to high temperatures.

Figure 3-13. Gross Gamma Scan of Capsule C3D

Figure 3-14. Close-up of C3D Fuel Pin at Failed End

Figure 3-15. End-on View of C3D Fuel Pin at Failed End

METALLOGRAPHY SAMPLES

SECTIONS

EXTERIOR

SCHEMATIC

Figure 3-16. Transverse and Longitudinal Sections of Specimen C3D

TABLE 3-5

DIMENSIONAL MEASUREMENTS SPECIMEN C3D

Location (Inches from the top)	Diameter (Inches)				Profile			
	0°		90°		0°		90°	
	Pre-	Post-	Pre-	Post-	Pre-	Post-	Pre-	Post-
1/4	0.2538	0.254	0.2528	0.253	0.1900	0.070	0.1910	0.076
1-1/4	0.2528	0.252	0.2529	0.252	0.1910	0.074	0.1925	0.071
2-1/4	0.2527	0.252	0.2528	0.252	0.1930	0.079	0.1930	0.070
3-1/4	0.2528	0.252	0.2528	0.252	0.1940	0.085	0.1930	0.068
4-1/4	0.2528	0.252	0.2528	0.252	0.1950	0.088	0.1920	0.065
5-1/2	0.2528	0.252	0.2529	0.252	0.1960	0.090	0.1920	0.065
6-1/4	0.2529	0.252	0.2527	0.252	0.1970	0.093	0.1920	0.064
7-1/4	0.2527	0.252	0.2526	0.252	0.1980	0.095	0.1920	0.062
8-1/4	0.2527	0.252	0.2526	0.252	0.2000	0.096	0.1920	0.061
9-1/4	0.2528	0.252	0.2526	0.252	0.2010	0.094	0.1925	0.065
10-1/4	0.2527	0.252	0.2527	0.252	0.2020	0.087	0.1930	0.069
11-1/4	0.2527	0.252	0.2526	0.252	0.2020	0.078	0.1940	0.075
12-1/4	0.2527	0.252	0.2527	0.252	0.2020	0.067	0.1950	0.085
13-1/4	0.2527	0.253	0.2527	0.253	0.1990	0.063	0.1950	0.090
14-1/4	0.2530		0.2524		0.1935		0.1945	
At the failure		0.268		0.308

Figure 3-17a. Specimen C3D - Microstructure of Cladding at Rupture

Figure 3-17b. Specimen C3D - Microstructure of Cladding above Failure Region

Figure 3-18. Bottom of Fuel Pin C3D at Varying Axial Positions Obtained by Consecutive Grinding

The fuel above the failure was characterized by a large, relatively uniform, central void which extended over the fuel length. This is illustrated in Figure 3-16 (b through i, and m and n). It can be seen that NaK has permeated virtually all of the fuel surfaces. It is shown in Figure 3-19 that surrounding the central void is a region of columnar grains extending to within 13 mils of the cladding, a band of equiaxed grains, a thin layer containing silver-colored precipitates, and a 3-1/2-mil-thick grey reaction product* at the periphery of the fuel. White precipitates, some rather massive, were scattered throughout the fuel. Since the molten fuel flowed toward the bottom of the pin, the microstructural features that define the molten boundary were erased.

The peripheral fuel - NaK reaction product exhibited significant alpha and beta-gamma activities. This is shown in the longitudinal section incorporating the bottom of the fuel (Figure 3-16), which is depicted again as Figure 3-20 and includes the autoradiographs. It is characteristic of this reaction product that it is obliterated by acidic etchants. From these observations it is probable that the reaction layer contained fuel, NaK, and fission products.

The top longitudinal section in Figure 3-16 (o), which is repeated in Figure 3-21 along with corresponding alpha and beta-gamma autoradiographs, revealed the cause for the large activity spikes seen in this area in the original gamma scan (Figure 3-13). Apparently, the metallic particle is primarily composed of fission products as indicated by the extreme intensity of the beta-gamma autoradiograph and the lesser intensity of the particle with relation to the fuel in the alpha autoradiograph. Analysis of the activity of the particle by multichannel pulse height analysis further indicates that the particle apparently consisted of ruthenium, zirconium, niobium, cesium, barium, and lanthanum, as evidenced by Figure 3-22 a, b, c, and d.

An attempt to prepare the particle for chemical analysis revealed that it was only barely soluble in aqua regia, which would indicate that the particle did not contain stainless steel from the cladding. The genesis of the particle is not known. One mechanism that can be postulated is that it formed as a result of fuel centermelting which, at the start of each cycle of irradiation, began at the bottom of the fuel and slowly proceeded upward as the flux peak in the reactor shifted upward during the cycle. This, then, would in essence be tantamount to zone-refining of the fuel and would result in the accumulation of insoluble fission products at the top of the fuel column.

The postulation that fuel melting started at the bottom of the pin and progressed upward during the cycle is also the basis for explanation of the cause of the failure**. Figure 3-23 illustrates the flux peak shift during a cycle in the GETR. It is believed that as the flux progressed upward, the fuel at the bottom of the pin cooled and contracted, permitting molten fuel from above

* This reaction product is characteristic of mixed oxide fuel which has been intentionally or inadvertently exposed to sodium or NaK for extended periods of time. (8) (9)

** Evaluation of the $\int k_d dt$ indicates melting should have occurred at rod powers obtained.

2

3

4

Figure 3-19. Panoramic View of Transverse Section from C3D

(a.) PHOTOMACROGRAPH

(b.) ALPHA AUTORADIOGRAPH

(c.) BETA-GAMMA AUTORADIOGRAPH

Figure 3-20. Bottom Fuel Section

BETA-GAMMA AUTORADIOGRAPH

ALPHA AUTORADIOGRAPH

Figure 3-21. Metallic Particle Top of Fuel
Specimen C3D

(a)

(b)

(c)

(d)

Figure 3-22. Gamma Scan of Metallographic Specimen with Metallic Particle

Figure 3-23. GETR Pool Position Axial Flux Shapes

to flow into any void. During subsequent startup, when the start-of-cycle-peak was again concentrated at the bottom of the specimen, a very localized melting occurred such that solid fuel above the molten area prevented vertical expansion, thus requiring that the volume expansion on melting proceed radially. Repetition of this process would cause cladding failure. This subject will be treated further in Section IV.

Although the ruptured cladding permitted the fission gases to escape into the surrounding capsule, it was still possible to get a sample for chromatographic analysis. Table 3-6 presents these data.

TABLE 3-6

CHROMATOGRAPHIC ANALYSIS OF GASES FROM SPECIMEN C3D

<u>Constituent</u>	<u>Volume Percent</u>
O ₂ + Ar	0.2
N ₂	0.1
Xe	8.5
He	90.0
Kr	0.5
H ₂	None detected
CO ₂	None detected
Total percent detected	99.2
Xe/Kr ratio	17.0
Total gas	496 cc

The high helium content is, of course, due to its presence as a cover gas for the NaK. Gamma counting of the gas sample indicated a release of 3.84×10^{18} atoms of krypton-85. Estimation of the atoms of krypton-85 formed on the basis of the burnup, as determined by radiochemical analysis (63,000 MWd/T_e), results in a calculated fission gas release value of ~54 percent.

Mass spectrometric analysis of the xenon and krypton showed an isotopic distribution as given in Table 3-7.

3.2.3 Nondestructive Examination of Capsules C3B, C3C, and C3E

Discovery of the failure of the high burnup specimen C3D prompted extensive nondestructive examinations of the capsules to be transient irradiated: C3B, C3C, and C3E. The techniques employed for the nondestructive examinations were:

- a. Gross gamma scanning, and
- b. Pinhole camera autoradiographs.

3.2.3.1 Nondestructive Examination of Capsule C3B

Shown in Figure 3-24 is the gross gamma scan obtained from capsule C3B. Comparison of this scan with that obtained from capsule C3A (Figure 3-2) indicates that no significant fuel redistribution has occurred. This is substantiated by the post-irradiation examination of specimen C3A, which did not reveal any cladding deformation or fuel redistribution. Since specimen C3B operated at a lower rod power than specimen C3A, no fuel redistribution was to be expected.

Due to the low induced radioactivity level of the capsule and fuel specimen, the resultant pinhole camera autoradiograph was not suitable for reproduction. However, careful study of the negative did not reveal any signs of cladding deformation.

3.2.3.2 Nondestructive Examination of Capsule C3C

Illustrated in Figure 3-25 is the gross gamma scan obtained from capsule C3C. By comparison with the gamma scans from C3A and C3B (Figures 3-2 and 3-24), it appears that some internal fuel redistribution has occurred, but it does not appear that the cladding has been breached. This is confirmed by Figure 3-26, which is a pinhole camera autoradiograph of the bottom end of the fuel column. As can be seen, the hollow insulator pellet contains a significant amount of fuel, but the cladding does not seem to be ruptured since the fuel outline is sharply delineated.

TABLE 3-7

MASS SPECTROMETRIC ANALYSIS OF XENON AND
KRYPTON FROM SPECIMEN C3D

<u>Isotope</u>	<u>Atom Percent</u>
Kr-83	12.98 $\pm$ 1.2
Kr-84	29.16 $\pm$ 0.6
Kr-85	6.80 $\pm$ 1.0
Kr-86	51.06 $\pm$ 1.2
Xe-131	11.26 $\pm$ 0.2
Xe-132	18.90 $\pm$ 0.2
Xe-134	25.85 $\pm$ 0.2
Xe-136	44.01 $\pm$ 0.5

Figure 3-24. Gross Gamma Scan of Capsule C3B

Figure 3-25. Gross Gamma Scan of Capsule C3C

Figure 3-26. Pinhole Camera Autoradiograph of Bottom of C3C Fuel Pin

3.2.3.3 Nondestructive Examination of Capsule C3E

In the same manner as for capsule C3C, the gamma scan and pinhole camera autoradiographs (Figures 3-27 and 3-28) were used to evaluate the condition of the fuel and cladding. As can be seen, the extent of fuel redistribution appears to be somewhat less in this specimen than in C3C. This was to be expected since C3E operated at a lower rod power than C3C.

Figure 3-27. Gross Gamma Scan of Capsule C3E

Figure 3-28. Pinhole Camera Autoradiograph of Bottom of C3E Fuel Pin

1

2

3

4

SECTION IV

DISCUSSION OF RESULTS

4.1 FUEL BURNUP DETERMINATIONS

In addition to the fuel burnup determinations by thermocouple data and physics calculations, radiochemical and mass spectrographic determinations were made on fuel from the destructively examined specimens C3A and C3D. The burnup determined from each method is summarized in Table 4-1.

TABLE 4-1

COMPARISON OF FUEL SPECIMEN BURNUP* DETERMINATIONS

<u>Specimen</u>	<u>Thermo- couple Data ***</u>	<u>Physics Calcu- lations</u>	<u>Pu-239 Depletion</u>	<u>Cs-137 Analysis</u>	<u>Physics/ TC Data</u>	<u>Physics/ RCA</u>	<u>RCA/ TC Data</u>
C3A	9,200	7,900	8,900	8,700	0.86	0.89	0.97
C3B	8,100	7,600	**	**	0.94	-	-
C3C	73,000	54,000	**	**	0.74	-	-
C3D	76,000	57,000	63,000	†	0.75	0.90	0.83
C3E	70,000	56,000	**	**	0.80	-	-

* Megawatt days per metric ton of metal (U + Pu) and 200 MeV per fission.

** Not yet analyzed.

*** Peak values. Average over 6 inches fuel length is ~1.5 percent less.

† U-238 contamination of sample obscured measurement.

4.1.1 Thermocouple Data

Direct and continuous temperature measurements of the inner and outer NaK annuli (see Figure 3-1) permitted a calculation of the capsule linear power generation. The accuracy of this calculation is dependent on fairly precise knowledge of the physical properties and dimensions of the capsule materials, as well as the exact location of the thermocouples. As illustrated in Figure 4-1, the estimated accuracy of the linear heat generation is +7 and -5 percent. Sample calculations for heat transfer within the capsule are presented in Appendix B. The thermocouples, being located at the axial center of the 6-inch-long specimens, measure approximately the cycle average power according to the GETR flux shape versus control rod position curves presented in Figure 3-23. The burnup accumulated at the center of the fuel specimen is estimated to be, on the basis of Figure 3-23, approximately 1.5 percent higher than the average over the 6-inch length. This estimation agrees well with the observed deviation (from the average) based on gamma scans of C3A, and is further substantiated by the cycle integrated relative flux curves given in Appendix C. Integrating the curves of measured power generation versus time (given in Appendix D) yields the fuel specimen burnup based on thermocouple data.

4.1.2 Physics Calculations

Knowledge of the flux distribution during each cycle throughout the GETR core and pool, combined with test capsule geometry, permits a calculation of the power generation in the experiment. Cycle average values are computed (see Appendix C) and used to determine total fuel burnup. The accuracy of this method depends on how well the flux distribution in the GETR pool and test fuel can be determined, and on the reliability of the value selected for alpha 239 (capture cross section: fission cross section). The over-all accuracy in these calculations is believed to be about ± 10 percent.

4.1.3 Mass Spectrometry

This burnup value is obtained by measuring the Pu-240/Pu-239 ratio before and after irradiation, and from the result, calculating the atom percent burnup of Pu-239. Corrections for U-235 burnup are approximated and included in the total. The greatest uncertainty in this calculation evolves from the choice of the alpha 239 value. The error in the final burnup value by this method is about ± 15 percent.

4.1.4 Radiochemical Analysis

The approach in this technique involves the measurement of a fission product to heavy element (H.E.) ratio in the irradiated fuel. Basically, the number of fissions occurring in the fuel is determined by measuring the number of atoms of a fission product isotope present and dividing by the fission yield of that isotope. Quoted accuracies for this method are dependent principally on the fission yields and half-life values. Cesium-137 and cerium-144 were both analyzed in sample C3A, and their ratios to the heavy elements were determined. Three separate

Figure 4-1. Capsule Aluminum Heat Sink ΔT versus Linear Power Generation

measurements were performed on the sample, and in all cases, the Cs/H. E. ratio was reproduced with less than 10 percent variation. The Ce/H. E. ratio varied between 9 and 30 percent higher than Cs/H. E. values, and produced up to a 25-percent difference between final burnup values. This difference is attributed to the uncertainty in the value used for cerium fission yield (3.8 percent). The cerium-144 analysis has been discounted in the final burnup determination. The Cs-137 values are considered to be accurate to ± 10 percent. In the case of specimen C3D, apparent contamination of the sample with U-238 distorted the results so they were discounted for this report.

4.1.5 Comparison of Burnup Data

It is apparent from Table 4-1 that the most consistent comparison seems to be that between the physics calculations and the radiochemical burnup analysis, and on this basis the physics determined burnup is about 10 percent low. By adjusting the physics-determined burnup upward by 10 percent, better agreement is obtained with the thermocouple data; the deviation between adjusted physics values and thermocouple data values then falls within about 15 percent for the high burnup specimens, which is probably well within the range of uncertainties for these experiments. The same range of deviation is obtained by averaging the thermocouple data values, the physics calculation values, and the Pu-239 depletion values. Since this approach seems to be a generally accepted one, ⁽¹¹⁾ the burnup for each specimen shall be considered to be as given in Table 4-2.

TABLE 4-2

ADJUSTED FUEL SPECIMEN BURNUP VALUES

<u>Specimen</u>	<u>Burnup MWd/T_e (U + Pu)</u>
C3A	8,700
C3B	7,900
C3C	64,000
C3D	65,000
C3E	63,000

In this analysis it has been assumed that the effective energy to be attained from fuel in a fast reactor would amount to about 200 MeV of the approximately 205 MeV associated with fissioning of Pu-239. This is based on total fission energy plus energy from $(n - \gamma)$ reactions, minus neutrino energy, minus 5 MeV/fission to allow for factors such as non-useful gamma energy heating outside the core. ⁽¹²⁾

4.2 ROD POWER DETERMINATIONS

4.2.1 By ΔT Measurements

Rod powers determined from the temperature differences indicated by the internal thermocouples (see Figure 3-1) reflect instantaneous power generation values. The average rod power for a cycle of irradiation is determined by integration of the instantaneous power-time curves (Appendix D) corrected by the ratio of the integrated relative flux at the thermocouple position (14 inches above the bottom of the core) to the average relative flux over the length of the fuel specimen (11 to 17 inches above the bottom of the core). From the integrated relative flux curves given in Appendix C it is seen that this correction amounts to about 1.5 percent.

An estimate of the maximum local rod power attained in each cycle can be made by calculating the rod power based on thermocouple data and correcting it for the start-of-cycle peaking factor obtained from the GETR flux curves (Figure 3-23). As can be seen from Figure 3-23, the highest relative flux value occurs at the bottom of the fuel specimen (11 inches above the bottom of the core) at the start of the cycle. The rod power at this position then is estimated by multiplying the rod power, as determined by the thermocouple data, by the ratio of the start-of-cycle relative flux at the 11-inch position to the relative flux at the 14-inch position (the location of the thermocouples). Table 3-1 summarized these data for each cycle.

4.2.2 By Physics Calculations

The determination of rod power by physics calculations is based on the pool-position cycle-average integrated flux and a value of 180 MeV per fission.

From the integrated flux curves given in Appendix C it was determined that the cycle-average and peak relative flux values over the length of the fuel (11 to 17 inches above the bottom of the core) were as given in Table 4-3. Table 3-2 presented the cycle-average rod power as computed for each cycle of the irradiation and the start-of-cycle peak rod power. The start-of-cycle peak rod power was obtained by multiplying the cycle-average rod power by the ratio of the start-of-cycle peak relative flux at 11 inches above the core bottom ($\phi_{rel} = 1.82$, Figure 4-2) to the cycle-average-relative flux given in Table 4-3.

A comparison of the data given in Tables 3-1 and 3-2 shows that there is only marginal agreement between the thermocouple data and the physics calculations. It is felt that the major part of these differences arises from the fact that the physics calculations cannot account for the many local, time-dependent variables that affect rod power, whereas the thermocouple data will reflect these perturbations.

TABLE 4-3
CYCLE AVERAGE AND PEAK RELATIVE FLUX VALUES
FOR
11 TO 17 INCHES ABOVE CORE BOTTOM

<u>Cycle Number</u>	<u>Average</u>	<u>Peak</u>	<u>Average/Peak</u>
42	1.42	1.45	0.98
44	1.46	1.51	0.967
45	1.46	1.51	0.967
46	1.45	1.49	0.973
47	1.52	1.55	0.980
48	1.48	1.51	0.980
49	1.51	1.54	0.980
50	1.52	1.55	0.980
51	1.52	1.55	0.980
52	1.50	1.53	0.980
53	1.52	1.55	0.980

Figure 4-2. $\int_0^{T_c} k d\theta$ versus Linear Power for Specimens C3A and C3B

4.3 $\int k d\theta$ DETERMINATIONS

4.3.1 Specimens C3A and C3B

It will be recalled that the photomicrograph of specimen C3A did not reveal any sign of gross centermelting. However, it was stated that the fuel redistribution into the hollow insulator pellet is thought to have possibly occurred by melting of the fuel above the insulator pellet. Thus, one could assume that the fuel at the bottom of the specimen experienced melting at the beginning of the irradiation, and the molten boundary disappeared as a consequence of subsequent irradiation at a lower rod power. From Table 3-1, the peak rod power at the bottom of the specimen was calculated to be 29.3 kW/ft. If it is assumed that the integral to melting is ~ 85 watts/cm,⁽¹³⁾ and the melting point of the 28 percent PuO_2 - 72 percent UO_2 mixed oxide is $\sim 2590^\circ\text{C}$,⁽¹⁴⁾ then it is possible to calculate the total integral and compare it to the integral for melting. From the heat transfer calculations given in Appendix B it can be determined that the temperature at the cladding inside diameter at a rod power of 29.3 kW/ft was $\sim 1180^\circ\text{F}$, and the fuel surface heat flux was 1.74×10^6 Btu/h-ft². Assuming a fuel cladding gap coefficient of 1500 Btu/h-ft²°F,

the fuel surface temperature is estimated to have been 2320°F (1270°C) or $\int_0^{T_S} k d\theta$ equaled

34.0 watts/cm. From physics calculations it was determined that the radial flux depression,

ϕ_C/ϕ_S , was 0.24. Assuming no center void in the fuel results in an $\int_{T_S}^{T_C} k d\theta$ of ~ 51 watts/cm

and a total integral of ~ 85 watts/cm, which is just at melting. From the post-irradiation photomicrographs it appeared that the center void in the fuel was about 0.035-inch diameter.

Correcting the $\int_{T_S}^{T_C} k d\theta$ calculations for the presence of a void⁽¹⁵⁾ results in an estimated

total integral of ~ 81 watts/cm, which is slightly less than that required for melting.

Thus, it appears that melting probably occurred early in the cycle and was quite local in extent since the high linear heat generation rate, 29.3 kW/ft, occurred only at the lower end of the fuel rod. In contrast, the linear heat generation rate at the upper end of the rod at the start of the cycle was estimated to be ~ 20 kW/ft - much below that required for melting.

Illustrated in Figure 4-2 is the total $\int k d\theta$ for specimens C3A and C3B as a function of rod

power. As noted on the figure, the calculations for the plot are based on an 0.035-inch-diameter central void and a flux depression, ϕ_C/ϕ_S , of 0.24.

4.3.2 Specimens C3C, C3D, and C3E

The discussion regarding $\int k d\theta$ for specimens C3A and C3B is also applicable to the high burnup specimens C3C, C3D, and C3E, except that as burnup was accumulated the flux depression, ϕ_c/ϕ_s , decreased. Physics calculations indicated that ϕ_c/ϕ_s changed from 0.24 at the start of the irradiation, to 0.41 at the end of the irradiation. The effect on $\int k d\theta$ of the changing flux depression is illustrated in Figure 4-3, which serves again to illustrate that centermelting of the specimens would be expected only at the bottom of the fuel column since this is the only place where the linear heat generation attained values high enough to cause melting (see Table 3-1). There is an added complication in interpreting the fuel temperatures reached in C3D owing to the penetration of NaK through the fuel and subsequent formation of the fuel - NaK reaction product.

4.4 CORRELATION OF MICROSTRUCTURE WITH ROD POWER

Inasmuch as data on the temperatures at which the boundaries for equiaxed and columnar grain growth occur are not yet available for mixed oxide fuel, it is instructive to attempt to deduce these values from the measured temperatures in the C3A experiment. As can be seen in Figure D-1 of Appendix D, the maximum rod power as determined from the thermocouple data was 25 kW/ft. It is assumed that the grain structure which developed at this rod power would be retained during the remaining portion of the irradiation wherein the rod power constantly decreased. From the calculations given in Appendix B, it can be determined that the fuel surface temperature was 1040°C (based on an estimated fuel cladding gap conductance of 1500 Btu/h-ft²-F°). The photomicrograph of a C3A transverse cross section (Figure 3-6) shows that the boundary for equiaxed grain growth occurred at a radius of about 0.097 inch, and the columnar grain growth boundary occurred at about 0.086 inch. By employing the expressions

$$\int_0^{T_r} k d\theta = \int_0^{T_s} k d\theta + \int_{T_s}^{T_r} k d\theta \quad *$$

and

$$\int_{T_s}^{T_r} k d\theta = \frac{Q' \text{ PSF}(r)}{4 \pi} .$$

it is possible to estimate T_r .

* T_r = Temperature at grain growth radius of interest.

T_s = Fuel surface temperature - °C.

Q' = Linear heat generation - watts/cm.

PSF(r) = Power shape factor at radius of interest (flux depression correction factor)⁽¹³⁾.

T_c = Fuel center temperature - °C.

Figure 4-3. $\int_0^{T_c} k d \theta$ versus Linear Power for Specimens C3C, C3D, and C3E, with Indicated Flux Depression.

Performing the arithmetic and using an $\int_0^{T_c} k dT$ estimate for mixed oxide, ⁽¹³⁾ yields temperatures of about 1600 and 1900°C for the equiaxed grain growth boundary and the columnar grain growth boundary, respectively. Uncertainties in the heat transfer calculations, the thermal conductivity of the mixed oxide, and the fuel cladding gap conductance forces an uncertainty of at least ± 20 percent on these values.

4.5 CONSEQUENCE OF AXIAL REDISTRIBUTION

In subsection 3.2.2 of Section III it was postulated that the cladding failure in capsule C3D was due to the repeated densification and melting of fuel in the lower portion of the fuel pin, which induced cladding expansions beyond the ductility limit of the stainless steel. While there is an apparent contradiction here to the molten fuel studies on UO_2 , ⁽¹⁷⁾ the discrepancy may not be real.

The work with UO_2 under boiling water reactor conditions demonstrated that molten fuel expansion can be accommodated axially if the fuel is brought to power sufficiently slowly so that the hot fuel can flow within the cladding and the outer, non-plastic annulus of fuel. This slow-startup procedure was not followed during all of the cycles in which C3D was in-pile. Moreover, the diametral restraint to expansion is considerably greater in the BWR case. It may be concluded, therefore, that overpower conditions which rapidly take the oxide into the molten regime without access to a void will cause swelling of the cladding.

APPENDIX A

FUEL SPECIMEN AND CAPSULE DETAILS

A.1 SERIES 3 CAPSULE DESIGN

The program for transient irradiation of 0.250-inch-diameter mixed oxide fuel specimens in TREAT involves both non-preirradiated (Series 2) and steady-state preirradiated (Series 3) fuel specimens. In order to minimize differences in nuclear characteristics, effect economies in design and fabrication, and permit direct comparison of results between non-preirradiated and preirradiated specimens, all the capsules were built to the same basic design.

A.1.1 Design Criteria

- a. An external capsule configuration that would permit installation in the TREAT reactor core by substitution for a TREAT fuel element.
- b. Sufficient heat capacity to absorb all the heat generated by the fuel specimen during transient irradiation in TREAT.
- c. Calorimeter-type construction so that heat generated by the fuel specimen during transient irradiation may be accurately measured by temperature rise.
- d. Heat transfer characteristics such that steady-state irradiations at desired heat generation rates would not result in excessive fuel and cladding temperatures.
- e. An external capsule configuration that would permit steady-state irradiation in a General Electric Test Reactor pool position.
- f. Sufficient strength and integrity to sustain internal pressures and provide containment for fuel and other capsule materials.

The manner in which these criteria were satisfied for the steady-state irradiations in the GETR are depicted by the capsule illustrated in Figure A-1.

1

2

3

4

A. 2 SERIES 3 FUEL SPECIMEN DESIGN AND FABRICATION

A. 2. 1 Design

Fuel specimen mechanical design was predicated on the simulation of the then-current FCR reference fuel pin design⁽¹⁶⁾. In this regard, fuel pellet density and diameter, cladding diameter and thickness, and fuel volume-to-fission-gas plenum volume ratio duplicate the reference design. In addition, means for installing a pressure transducer on the fuel pin after completion of the steady-state irradiation also had to be provided. Shown in Figure A-2 are the details of the fuel pin construction. The diaphragm at the bottom of the pin permitted the installation of a pressure transducer which was previously equipped with a hypodermic needle to puncture the diaphragm.

Table A-1 contains the salient as-fabricated data for each fuel specimen.

A. 2. 2 Fabrication

A. 2. 2. 1 Fuel Fabrication

A 270-gram batch (B-4) of mixed oxide powder was prepared for the Series 3 specimens. The feed solution contained 28 g/l Pu, 72 g/l natural U, and ~1 M nitric acid (isotopic analysis of the feed solution is given in Table A-2). Precipitation was accomplished by addition of ammonium hydroxide to a final pH of 7 to 8. The coprecipitate was stirred for 15 minutes at ambient temperature, and then filtered. The cake was washed three times with 0.1 M NH_4OH , and dried with air and acetone. The reduction cycle consisted of heating to 750°C, holding at 750 to 780°C for 3 hours, and cooling to room temperature in flowing 6 percent H_2 - 94 percent He. The reduced powder was ball-milled for 7-1/2 hours to provide a -325 mesh powder for pressing.

Pressing and sintering data for the Series 3 pellets are given in Table A-3.

A. 2. 2. 2 Fuel Pin Component Fabrication

As seen in Figure A-2, all the components for the fuel pins were made from austenitic stainless steel, with the cladding specified as Type 347. Inspection of the cladding consisted of ultrasonic testing for wall defects and burst tests of sample pieces, as well as dimensional measurements. The burst pressure of the three samples tested was ~11,200 psig at room temperature.

7

-

4

7

7

Figure A-2. Fuel Specimen Details

TABLE A-1
DATA FOR SERIES 3 FUEL PINS

TREAT SERIES NUMBER	3-A	3-B	3-C	3-D	3-E
Powder Source - Batch No.	B-4	B-4	B-4	B-4	B-4
(1) Ratio Pu/Pu + U in Mixed Oxide (%)	28.1	28.1	28.1	28.1	28.1
Inside Diameter of Cladding (in.)	0.221	0.221	0.221	0.221	0.221
Maximum Diameter Range of Pellets (in.)					
Uranium Oxide	0.2157-0.2163	0.2158-0.2161	0.2158-0.2162	0.2157-0.2160	0.2157-0.2161
Mixed Oxide	0.215 -0.2195	0.215 -0.2185	0.215 -0.219	0.2155-0.2195	0.216 -0.2195
Average Diameter of Pellets (in.)					
Uranium Oxide	0.2159	0.2160	0.2161	0.2158	0.2160
Mixed Oxide	0.217	0.2165	0.217	0.2175	0.2185
Diametrical Gap (in.)					
Range	0.002-0.006	0.003-0.006	0.002-0.006	0.002-0.006	0.002-0.005
Average	0.004	0.0045	0.004	0.0035	0.0025
Fuel Length (in.)	6.013	6.010	6.0125	6.0375	6.110
Fuel Weight (g)	38.51	38.84	38.82	38.90	39.525
Fuel Density (% of Theoretical)					
Range	93.9-96.9	94.4-97.8	94.5-97.2	94.1-97.2	94.3-97.5
Average	95.0	96.8	96.0	95.2	95.0
(2) Fuel O/M Ratio	1.979 ± 0.006	1.979 ± 0.006	1.979 ± 0.006	1.979 ± 0.006	1.979 ± 0.006
Total Weight of Fuel Pin (g)	74.63	75.20	75.10	75.31	75.88
Date Loaded	8/22/62	8/22/62	8/22/62	8/30/62	8/30/62

EXPLANATORY NOTES FOR TABLE A-1

- (1) Ratio of Pu/Pu + U is based upon plutonium and uranium analyses of sintered pellets. Precision of plutonium and uranium analyses is ± 1 percent; therefore, the precision of the calculated ratio is ± 2 percent (i. e., 20.0 percent is 19.6 to 20.4 percent).
- (2) Fuel O/M ratio was determined for reference pellets from the batch (B-4). Determination was made by technique established by W. L. Lyon and described in a report entitled "The Measurement of Oxygen to Metal Ratio in Solid Solutions of Uranium and Plutonium Dioxides" (GEAP-4271).

TABLE A-2

ISOTOPIC ANALYSIS OF PLUTONIUM FEED SOLUTION
FOR
SERIES 3 FUEL PELLETS

<u>Isotope</u>	<u>Atom %</u>
239	91.60
240	7.65
241	0.72
242	0.03

TABLE A-3

PRESSING AND SINTERING DATA FOR SERIES 3 PELLETS

(28% Pu - 72% U Mixed Oxide)

Pellet Numbers	Pelletizing Pressure (psig)	Green density (% theoretical)		Sintering		Sintered Density (% theoretical)		Firing Shrinkage (%)		Comments
		Range	Avg.	Temp (°C)	Time (hrs)	Range	Avg.	Diametric	Axial	
1-4	9,500	46.4-46.6	46.5	1580-1600	4-1/2	95.4-96.5	96.0	22.3	21.4	Warped (a) Warped Warped and chipped Warped and chipped Warped Warped and chipped (b) green data for three pellets (c) green data for two pellets (d) green data for seven pellets (e) green data for five pellets
5-8	11,000	47.4-47.9	47.7			95.9-97.1	96.7	21.7	20.7	
9-12	14,000	48.4-48.6	48.5			95.9-96.3	96.1			
13-16	20,000	51.1-51.5	51.4			96.0-96.3	96.1	19.6	18.8	
17-20	10,000	46.6-46.8	46.8			94.3-95.1	94.7	21.8	20.8	
21-30	11,000	47.0-47.6	47.3	1580-1610	4-1/2	93.9-95.2	94.5	21.5	20.3	
31-34	12,000	48.5	48.5			94.2-95.4	94.8	21.1	19.7	
35-37	18,000	50.5-50.8	50.6			93.6-94.6	94.2	19.7	18.3	
38-40	23,000	52.8-53.0	52.9			92.3-92.9	92.5	18.0	16.7	
41-42	16,500	49.8-50.1	49.9			91.4-92.9	92.2	19.0	18.9	
44-64	10,500	46.7-48.0	47.3	1560-1620	4	89.9-92.2	91.4	20.5	19.6	
65-66	19,500	50.2-50.8	50.5			91.5-92.0	91.8	19.2	17.5	
67-68	13,500	47.6-50.0	48.8			90.0-91.5	90.8	19.6	18.3	
69-74	11,000	44.7-47.1	46.3			88.8-94.6	91.4	21.3	19.8	
75-89 ^(b)	11,000	47.5-48.6	47.9			96.3-97.8	97.3	22.0	20.7	
90-99 ^(c)	11,000	47.7	47.7	1560-1590	4	95.9-97.0	96.5	22.0	20.5	
100-120 ^(b)	11,000	48.1-48.4	48.2	1560-1590	4	92.3-97.2	94.9	21.2	20.1	
121-144 ^(d)	11,000	47.6-48.3	47.8	1560-1590	4	93.0-95.7	94.5	21.2	19.8	
145-168 ^(e)	11,000	47.7-48.3	48.0	1570-1600	4	93.2-95.3	94.7	21.1	20.2	
169-177	10,500	46.8-48.0	47.5	1480-1530	4	96.2-97.5	96.8	22.0	20.9	

NOTES: (a) hour-glass configuration
 (b) green data for three pellets
 (c) green data for two pellets
 (d) green data for seven pellets
 (e) green data for five pellets

APPENDIX B

SAMPLE HEAT TRANSFER CALCULATIONS

B. 1 BASIS

Fission heating	=	22.9 kW/ft
Gamma heating	=	1.8 watts/gram
Coolant - water	=	8 gal/min at 11 ft/sec

B. 2 CAPSULE DIMENSIONS

D_1	=	0.250 inch	=	fuel cladding outside diameter
D_2	=	0.370 inch	=	thermal dam inside diameter
D_3	=	0.967 inch	=	thermal dam outside diameter
D_4	=	1.027 inches	=	capsule tube inside diameter
D_5	=	1.125 inches	=	capsule tube outside diameter

B. 3 CAPSULE MATERIALS

$D_1 - D_2$	NaK
$D_2 - D_3$	Aluminum (6061)
$D_3 - D_4$	NaK
$D_4 - D_5$	304 Stainless Steel

B. 4 FILM COEFFICIENT

$$\begin{aligned}
 T_w &= 110^\circ \text{F} \\
 h &= 170 \left(1 + t \times 10^{-2} - t^2 \times 10^{-5} \right) \frac{v^{0.8}}{D_e^{0.2}} * \\
 h &= 170 \left[1 + (110)(10^{-2}) - (110)^2(10^{-5}) \right] \frac{11^{0.8}}{0.125^{0.2}} \\
 h &= 3460 \text{ Btu/h-ft}^2\text{-}^\circ\text{F} .
 \end{aligned}$$

* Glasstone, "Principles of Nuclear Engineering," p. 678.

B.5 GAMMA HEATING OF COMPONENTS

$$(1.8) \frac{\text{watts}}{\text{gm}} \times (3.415) \frac{\text{Btu}}{\text{h-watt}} \times (454) \frac{\text{gm}}{\text{lb}} = 2790 \frac{\text{Btu}}{\text{h-lb}}$$

$$\left(\frac{\text{in.}^3}{\text{ft}} \right) \frac{\text{volume}}{\text{ft-length}} \times \left(\frac{\text{lb}}{\text{in.}^3} \right) \text{sp. wt.} \times 2790 \frac{\text{Btu}}{\text{h-lb}} = \frac{\text{Btu}}{\text{h-ft}}$$

$$(12) \left(\frac{\pi}{4} \right) \left(D_o^2 - D_i^2 \right) (\rho) (2790) = \frac{\text{Btu}}{\text{h-ft}}$$

$$\frac{\text{Btu}}{\text{h-ft}} = (26,300) \left(D_o^2 - D_i^2 \right) (\rho) = {}_oQ'_i \gamma$$

For $D_2^2 - D_1^2$ (NaK), $\rho = 0.0307 \text{ lb/in.}^3$,

$$\begin{aligned} {}_1Q'_2 \gamma &= (26,300) (0.370^2 - 0.250^2) (0.0307) \\ &= 60 \text{ Btu/h-ft.} \end{aligned}$$

For $D_3^2 - D_2^2$ (Aluminum), $\rho = 0.098 \text{ lb/in.}^3$,

$$\begin{aligned} {}_2Q'_3 \gamma &= (26,300) (0.967^2 - 0.370^2) (0.098) \\ &= 2060 \text{ Btu/h-ft.} \end{aligned}$$

For $D_4^2 - D_3^2$ (NaK), $\rho = 0.0307 \text{ lb/in.}^3$,

$$\begin{aligned} {}_3Q'_4 \gamma &= (26,300) (1.027^2 - 0.967^2) (0.0307) \\ &= 95 \text{ Btu/h-ft.} \end{aligned}$$

For $D_5^2 - D_4^2$ (304 Stainless Steel), $\rho = 0.29 \text{ lb/in.}^3$,

$$\begin{aligned} {}_4Q'_5 \gamma &= (26,300) (1.125^2 - 1.027^2) (0.29) \\ &= 1630 \text{ Btu/h-ft.} \end{aligned}$$

B.6 LINEAR HEAT RATES, INCLUDING GAMMA HEATING

$${}_1Q'_2 = 22.9 \text{ kW/ft} = 78,200 \text{ Btu/h-ft},$$

$${}_2Q'_3 = {}_1Q'_2 + {}_1Q'_2 \gamma = 78,200 + 60 = 78,260 \text{ Btu/h-ft},$$

$${}_3Q'_4 = {}_2Q'_3 + {}_2Q'_3 \gamma = 78,260 + 2,060 = 80,320 \text{ Btu/h-ft},$$

$${}_4Q'_5 = {}_3Q'_4 + {}_3Q'_4 \gamma = 80,320 + 95 = 80,415 \text{ Btu/h-ft},$$

$$Q'_5 = {}_4Q'_5 + {}_4Q'_5 \gamma = 80,415 + 1,630 = 82,045 \text{ Btu/h-ft}.$$

B.7 CAPSULE SURFACE HEAT FLUX

$$Q''_5 = \frac{Q'_5}{\pi D} = \frac{(82,045)(12)}{(\pi)(1.125)} = 278,000 \text{ Btu/h-ft}^2.$$

B.8 CAPSULE SURFACE TEMPERATURE

$$\begin{aligned} T_5 &= T_w + \frac{Q''_5}{h} \\ &= 110 + \frac{278,000}{3,460} \\ &= 110 + 81 \\ &= 191^\circ\text{F}. \end{aligned}$$

B.9 INTERNAL TEMPERATURES

$$\Delta T = \frac{Q' \ln \frac{D_o}{D_i}}{2 \pi k} ,$$

$${}_4 \Delta T_5 = \frac{{}_4 Q'_5 \ln \frac{D_5}{D_4}}{2 \pi k_{s/s}}$$

$$= \frac{(80,415) \ln \frac{1.125}{1.027}}{(2 \pi) (9.4)} \quad {}_4 k_5 = 9.4 \text{ (Stainless Steel)}$$

$$= 125 F^\circ .$$

$${}_3 \Delta T_4 = \frac{{}_3 Q'_4 \ln \frac{D_4}{D_3}}{2 \pi k_{NaK}}$$

$$= \frac{(80,320) \ln \frac{1.027}{0.967}}{2 \pi (14.4)} \quad {}_3 k_4 = 14.4 \text{ (NaK)}$$

$$= 53 F^\circ .$$

$${}_2 \Delta T_3 = \frac{(78,260) \ln \frac{0.976}{0.370}}{2 \pi (115)} \quad {}_2 k_3 = 115 \text{ (Aluminum)}$$

$$= 104 F^\circ$$

$${}_1 \Delta T_2 = \frac{(78,200) \ln \frac{0.370}{0.250}}{2 \pi (14.8)} \quad {}_1 k_2 = 14.8 \text{ (NaK)}$$

$$= 330 F^\circ$$

$$\Delta T_{\text{cladding}} = \frac{Q'_1 \ln \frac{D_o}{D_i}}{2 \pi k} \quad k_{\text{cladding}} = 12.6 \text{ (347 Stainless Steel)}$$

$$= \frac{(78,200) \ln \frac{0.250}{0.220}}{2 \pi (12.6)}$$

$$= 126 \text{ F}^\circ .$$

$\Delta T_{\text{fuel-cladding gap}}$

$$\Delta T_{\text{gap}} = \frac{Q''_{\text{fuel}}}{h_{\text{gap}}} .$$

Assume $h = 1500 \text{ Btu/h-ft}^2\text{-F}^\circ$.

$$Q''_{\text{fuel}} = \frac{Q'_{\text{fuel}}}{\pi D} = \frac{(22.9) (3415) (12)}{\pi (0.220)} ,$$

$$Q''_{\text{fuel}} = 1.36 \times 10^6 \text{ Btu/h-ft}^2 ,$$

$$\Delta T_{\text{gap}} = \frac{1.36 \times 10^6}{1500}$$

$$= 905 \text{ F}^\circ .$$

$$T_5 = 191^\circ \text{F}$$

$${}_4\Delta T_5 = 125 \text{ F}^\circ$$

$$T_4 = 316^\circ \text{F}$$

$${}_3\Delta T_4 = 53 \text{ F}^\circ$$

$$T_3 = 369^\circ \text{F}$$

$${}_2\Delta T_3 = 104 \text{ F}^\circ$$

$$T_2 = 473^\circ\text{F}$$

$${}_1\Delta T_2 = 330^\circ\text{F}$$

$$T_1 = 803^\circ\text{F}$$

$$\Delta T_{\text{clad}} = 126^\circ\text{F}$$

$$T = 929^\circ\text{F}$$

$$\Delta T_{\text{gap}} = 905^\circ\text{F}$$

$$T_s = 1834^\circ\text{F} = \text{fuel surface temperature.}$$

B. 10 FUEL SURFACE TEMPERATURE FOR 29.3 kW/ft ROD POWER

$$\Sigma \Delta T \text{ at } 22.9 \text{ kW/ft} = \Delta T_{\text{film}} + {}_4\Delta T_5 + {}_3\Delta T_4 + {}_2\Delta T_3 + {}_1\Delta T_2 + \Delta T_{\text{clad}} + \Delta T_{\text{gap}}$$

$$= 81 + 125 + 53 + 104 + 330 + 126 + 905$$

$$= 1724^\circ\text{F.}$$

$$\Sigma \Delta T \text{ at } 29.3 \text{ kW/ft} = (1724) \left(\frac{29.3}{22.9} \right)$$

$$= 2210.$$

Fuel surface temperature at 29.3 kW/ft

$$T_s = T_w + \Sigma \Delta T$$

$$= 110 + 2210$$

$$= 2320^\circ\text{F.}$$

APPENDIX C

CYCLE INTEGRATED RELATIVE FLUX CURVES

Figure C-1. Integrated Cycle Average Relative Flux - Cycle 42

Figure C-2. Integrated Cycle Average Relative Flux - Cycle 44

Figure C-3. Integrated Cycle Average Relative Flux - Cycle 45

Figure C-4. Integrated Cycle Average Relative Flux - Cycle 46

Figure C-5. Integrated Cycle Average Relative Flux - Cycle 47

Figure C-6. Integrated Cycle Average Relative Flux - Cycle 48

Figure C-7. Integrated Cycle Average Relative Flux - Cycle 49

Figure C-8. Integrated Cycle Average Relative Flux - Cycle 50

Figure C-9. Integrated Cycle Average Relative Flux - Cycle 51

Figure C-10. Integrated Cycle Average Relative Flux - Cycle 52

Figure C-11. Integrated Cycle Average Relative Flux - Cycle 53

APPENDIX D

POWER GENERATION VERSUS TIME CURVES

4

7

7

4

4

Figure D-1. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 42

Figure D-2. Series III - Irradiations in GETR -
Linear Power Generation versus
Time - Cycle 44

Figure D-3. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 45

Figure D-4. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 46

Figure D-5. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 47

Figure D-6. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 48

Figure D-7. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 49

Figure D-8. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 50

Figure D-9. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 51

Figure D-10. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 52

Figure D-11. Series III - Irradiations in GETR - Linear Power Generation versus Time - Cycle 53

APPENDIX E

MULTICHANNEL PULSE HEIGHT ANALYZER.
RESULTS ON CAPSULE C3D

During the initial phases of the post-irradiation examination of specimen C3D, an analysis of the fission product gamma spectrum was made. The results of this analysis indicated extensive mobility of fission products usually assumed to be quite immobile; viz., Zr Nb⁹⁵.

The gamma scan on which the spectrum analysis was based is shown in Figure E-1. The spectrum data were obtained with a 0.010-inch-slit collimator assembly located in the wall of the RML cell. The gamma radiation was detected with a 1-3/4 × 2 inch NaI (T1) scintillation crystal coupled to a photomultiplier tube. The resultant signal was amplified and transmitted to a Nuclear Data 512 channel pulse height analyzer. Data were plotted directly with a Moslay x-y recorder. Spectrum analyses for the activities given at each of the lettered positions in Figure E-1 are listed in Table E-1.

The high enrichment in Ru¹⁰³ at location D corresponds to the position of the metallic core in the specimen (Figure 3-20). Similar analyses were performed for specimens C3B, C3C, and C3E. These data are given in Figures E-2, E-3, and E-4, and Tables E-2, E-3, and E-4, respectively.

Figure E-1. Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3D

TABLE E-1

GAMMA SPECTRUM ANALYSIS CAPSULE C3D

<u>Position</u>	<u>Major Activity*</u>	<u>Other Activities</u>	<u>Comments</u>
A	Cr ⁵¹	Co ⁶⁰	No fission products.
B	Cr ⁵¹ , Co ⁶⁰	FP	Mixture of stainless steel and fission products.
C	FP	-	Expected FP distribution.
D	Ru ¹⁰³	ZrNb ⁹⁵ BaLa ¹⁴⁰	Highly enriched in Ru ¹⁰³ , somewhat enriched in ZrNb ⁹⁵ , and depleted in BaLa ¹⁴⁰ compared to location C.
E	FP	-	About same distribution as C but lower in intensity.
F	Ru ¹⁰³	ZrNb ⁹⁵ BaLa ¹⁴⁰	Similar distribution as D but lower in intensity.
G	ZrNb ⁹⁵	Ru ¹⁰³ BaLa ¹⁴⁰	Similar to C but depleted in Ru ¹⁰³ -- also lower over-all intensity.
H	Ru ¹⁰³	ZrNb ⁹⁵ BaLa ¹⁴⁰	Similar to D but not so enriched in Ru ¹⁰³ .
I	FP	-	Almost identical to C.
J	FP	Co ⁶⁰ , Cr ⁵¹	FP activity down a factor of 3 to 6 from C. Stainless steel activation products visible.
K	FP	-	About same distribution as C but 50% higher in intensity.
L	FP	-	Similar to K.
M	Ru ¹⁰³	ZrNb ⁹⁵ BaLa ¹⁴⁰	Same as H.
N	Cr ⁵¹ , Co ⁶⁰	FP	Fission products are definitely present.
O	Cr ⁵¹ , Co ⁶⁰	FP	Same distribution as N but twice as much activity.
P	FP	-	Same as K.
Q	Cr ⁵¹ , Co ⁶⁰	FP	Fission products are present but low compared to the Cr ⁵¹ and Co ⁶⁰ .
R	Cr ⁵¹ , Co ⁶⁰	-	No fission products observed.

*FP refers to fission products Ru¹⁰³, ZrNb⁹⁵, and BaLa¹⁴⁰. All identifications were made by observations of the gamma spectra since it was impossible to verify the identifications by radiochemistry.

Figure E-2. Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3B

Figure E-3. Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3C

Figure E-4. Gamma Scan Showing Positions for Gamma Spectrum Analysis - Specimen C3E

TABLE E-2

GAMMA SPECTRUM ANALYSIS CAPSULE C3B

<u>Position</u>	<u>Activities Present^(a)</u>	<u>Comments</u>
A	RuRh ZrNb-95 Co-60 Sb-125	
B	RuRh ZrNb-95 Co-60 Sb-125	Similar to A
C	RuRh ZrNb-95 Co-60 Sb-125	RuRh low in intensity, ZrNb-95 down a factor of 3 Co-60 up a factor of 3 -- relative to B
D	Co-60	Co-60 up a factor of 2 from C
E	Co-60 Sb-125	

^(a)0.42 MeV gamma peak is reported as Sb-125

TABLE E-3

GAMMA SPECTRUM ANALYSIS CAPSULE C3C

<u>Position</u>	<u>Activities Present</u>	<u>Comments</u>
A	Cr-51 Co-58 Co-60	
B	Cr-51 RuRh ZrNb-95 Co-60 BaLa-140	
C	Cr-51 RuRh Cs-137 ZrNb-95 Co-60 BaLa-140	RuRh and ZrNb-95 higher in intensity than in B
D	Same as C	Similar to C
E	RuRh ZrNb-95 BaLa-140	RuRh and ZrNb-95 major activities
F	Same as E	Slight increase of ZrNb-95, slight decrease of BaLa-140 -- relative to E
G	Same as E	Similar to E
H	Same as E	Slight increase of ZrNb-95 -- relative to G
I	Cr-51 RuRh ZrNb-95 Co-60 BaLa-140	FP down a factor of 4 -- relative to H. RuRh up a factor of 4, ZrNb-95, Co-60, BaLa-140 up a factor of 3 -- relative to B

TABLE E-3 (Continued)

<u>Position</u>	<u>Activities Present</u>	<u>Comments</u>
J	Same as I	Similar to I except Co-60 down a factor of 2
K	Same as I	Similar to 3
L	Same as I	Similar to 3
M	Cr-51 RuRh Cs-137 ZrNb-95 Co-60 BaLa-140	RuRh, 2rNb-95, BaLa-140 down a factor of 1-1/2, Co-60 up a factor of 1-1/2 -- relative to B
N	Cr-51 Co-60 Co-58	Co-60 up a factor of 2 from M
O	Same as N	Similar to N
P	Same as N	Similar to N
Q	Same as N	Similar to N

ACKNOWLEDGMENTS

This experimental work was a joint effort of several working groups and many individuals of the Advanced Products Operation, Irradiation Processing Operation, and the Vallecitos Atomic Laboratory of the General Electric Company.

The work was performed within the Fast Ceramic Reactor Development Program, Task C - Transient Testing of Fuel, under the direction of F. J. Leitz, FCR Project Engineer, and J. E. Hanson, Task C Leader who replaced J. H. Field when Mr. Field assumed the position of task leader for SEFOR Task 3.1 - Fuel-Clad Transient Performance.

Major contributions to the program were made by:

Program Management at IPO	-	L. E. Wilkinson
Irradiations in GETR	-	R. D. Robertshaw
Post-Irradiation Examinations	-	R. W. Darmitzel
Fuel Fabrication	-	E. F. Kurtz.

REFERENCES

1. McNelly, M. J., "Liquid Metal Fast Breeder Reactor Design Study (1000 MWe UO_2 - PuO_2 Fueled Plant)." GEAP-4418, January 1964.
2. Osborne, R. B., and Sherer, D. B., "Analytical Studies of Transient Effects In Fast Reactor Fuels," GEAP-4058, August 1962.
3. "Ceramics Research and Development Operation Quarterly Report, October - December 1962," HW-76300.
4. Field, J. H., "Experimental Studies of Transient Effects In Fast Reactor Fuels, Series I, UO_2 Irradiations," GEAP-4130, November 15, 1962.
5. Cohen, K. P., McNelly, M. J., Wolfe, B., "Experimental Fast Oxide Reactor," GEAP-3856, November 1961.
6. Hanson, J. E., et al., "Experimental Studies of Transient Effects In Fast Reactor Fuels, Series II - Mixed Oxide (PuO_2 - UO_2) Irradiations," GEAP-4804, June 1965.
7. Freund, Elias, MacFarlane, Geier, Boland, "Design Summary Report on the Transient Reactor Test Facility - TREAT," ANL-6034, June 1960.
8. "Fast Ceramic Reactor Development Program, Twelfth Quarterly Report, July - September 1964," GEAP-4723, pp 3-7, 3-13.
9. "Fast Ceramic Reactor Development Program, Thirteenth Quarterly Report, October 1964 - January 1965," GEAP-4795, to be issued.
10. Ruzak, P. P., and Rischall, H., "Further Studies on Stainless Steel Hot Cracking," Welding Journal, 36, 57S-61S, February 1957.
11. Hart, R. G., Lounsbury, M., Jones, R. W., and Notley, M. J. F., Nuclear Science and Engineering, 18, 6 (1964).
12. Weinberg, A. M., and Wigner, E. P., "The Physical Theory of Neutron Chain Reactors," The University of Chicago Press, Illinois (1958) p 138.
13. Rabin, S. A., and Baily, W. E., Unpublished Memo dated April 15, 1965.

REFERENCES (Continued)

14. Baily, W. E., and Gilbertson, J. C., Unpublished Memo dated November 12, 1964.
15. Hines, D. P., Unpublished Memo dated April 24, 1963.
16. Horst, K. M., Hutchins, B. A., Leitz, F. J., Wolfe, B., "Core Design Study for a 500 MW_e Fast Oxide Reactor," GEAP-3721, December 1961.
17. Lyons, M. F., et al., "UO₂ Fuel Rod Operation with Gross Central Melting," GEAP-4264, October 1963.

DISTRIBUTION LIST

	<u>No. of Copies</u>
Dr. James Gilbreath Assistant Director for Argonne National Laboratory Argonne, Illinois	4
Dr. S. Lawroski Argonne National Laboratory Argonne, Illinois	1
Mr. A. P. Donnell Atomic Power Development Associates 1911 First Street Detroit 26, Michigan	1
Nuclear Development Corp. of America 5 New Street White Plains, New York Attn: Library (For Dr. K. Goldsmith)	1
General Directorate, Research and Training Euratom Rue Belliard 51-53 Brussels, Belgium Attn: A. Stordeur	20
Mr. D. T. Eggen Atomics International P. O. Box 309 Canoga Park, California	1
M. C. Edlund The Babcock and Wilcox Company Atomic Energy Division Lynchburg, Virginia	1
Dr. John C. Woodhouse 1 Guest Lane Wilmington, Delaware	1
H. W. Savage Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee	1
W. R. Simmons Argonne National Laboratory Argonne, Illinois	1
W. R. Wykoff Los Alamos Scientific Laboratory Los Alamos, New Mexico	1
O. E. Dwyer Brookhaven National Laboratory Upton, New York	1

EXTERNAL DISTRIBUTION (Continued)

	<u>No. of Copies</u>
U. S. Atomic Energy Commission New York Operations Office 376 Hudson Street New York, New York Attn: Jules Wise	1
U. S. Atomic Energy Commission Canoga Park Area Office P. O. Box 591 Canoga Park, California Attn: C. W. Richards	1
H. A. Wagner The Detroit Edison Company 2000 Second Avenue Detroit 26, Michigan	1
United Nuclear Corporation 5 New Street White Plains, New York Attn: Library (For Dr. C. Graves)	1
F. A. Thalgott Argonne National Laboratory Idaho Division P. O. Box 1096 Idaho Falls, Idaho	1
Mr. Arthur S. Griswold Power Reactor Development Co. 1911 First Street Detroit 26, Michigan	1
Westinghouse Electric Corporation Bettis Atomic Power Laboratory P. O. Box 79 West Mifflin, Pennsylvania 15122 Attn: D. C. Dehuff	1
Dr. Karl H. Puechl Nuclear Materials and Equipment Corp. Apollo, Pa.	1
R. C. Werner Mine Safety Appliances Research Corp. Callery, Pa.	1
Director, Reactor Engineering Division Attn: C. A. Pursel Chicago Operations Office 9800 South Cass Avenue Argonne, Illinois	2

EXTERNAL DISTRIBUTION (Continued)

	<u>No. of Copies</u>
Mr. W. H. Brummett, Jr. Director, Contracts Division U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley 4, California	2
Col. J. B. Radcliffe, Jr. Director, Reactor Division U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley 4, California	2
Dr. G. W. Wensch Director, Sodium Reactors Branch Division of Reactor Development U. S. Atomic Energy Commission Washington 25, D. C.	3
Mr. F. A. Robertson Chief, California Patent Group U. S. Atomic Energy Commission San Francisco Operations Office P. O. Box 808 Livermore, California	1
J. K. Davidson Allis-Chalmers Company 508 Kennedy Street, NW Washington 11, D. C.	2
S. Visner Contract No. AT(11-1)-1317 Combustion Engineering Corporation Nuclear Division Prospect Hill Road Windsor, Connecticut	2
Mr. Samuel C. Nordlinger United States Embassy London, England	1
Dr. Peter A. Morris United States Embassy Tokyo, Japan	1
Dr. Abraham S. Freidman United States Embassy Paris, France	1
Dr. Martin Biles U. S. Mission to the European Communities c/o United States Embassy Brussels, Belgium	1

EXTERNAL DISTRIBUTION (Continued)

	<u>No. of Copies</u>
Director Research and Development Division U. S. Atomic Energy Commission Richland Operations Office Post Office Box 550 Richland, Washington 99352	1
Mr. D. Glenn Boyer USAEC Technical Representative U. S. Atomic Energy Commission White Shell Branch Pinawa, Manitoba, Canada	1
Dott, Ing. F. Pierantoni CNEN Via Mazzini 2 Bologna, Italy	4
Prof. W. Haefele Kernforschungszentrum Karlsruhe 7500 Karlsruhe Karlsruhe, Germany	10
Mr. G. Vendryes CEN Saclay Boite Postale 2 Gif-sur-Yvette (S et O), France	10
D.T.I.E. U. S. Atomic Energy Commission P. O. Box 62 Oak Ridge, Tennessee	3
Dr. I. Zartman, Chief Reactor Physics Branch Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1
Mr. S. A. Szawlewicz, Chief Research & Development Branch for Nuclear Safety Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1
Mr. J. M. Simmons, Chief Fuels & Materials Branch Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1
Dr. E. Anderson, Chief Chemical Separation & Development Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1

EXTERNAL DISTRIBUTION (Continued)

	<u>No. of Copies</u>
Mr. M. Whitman Assistant Director for Reactor Engineering Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1
Mr. T. Beresoviski Core Design Branch Division of Reactor Development & Technology U. S. Atomic Energy Commission Washington, D. C. 20545	1