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SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Thirty-Fifth Quarterly Report

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4

5

6

7

8

9

10

TABLE OF CONTENTS

	Page
1. Introduction	1
2. Summary	1
2.1 Task B—Defected Fuel Behavior	1
2.2 Task C-1—Safety Experimentation (Transient Overpower)	1
2.3 Task C-2—Safety Experimentation (Loss of Flow)	1
2.4 Task E—Fuel Materials Properties and Performance Evaluation	7
2.5 Task F—Fast Flux Tests and Support	7
2.6 Task G-6—Core Design Analysis	10
2.7 Task G-10—Kinetics Analysis	10
2.8 Task L—Cladding Development	10
2.9 Task N—Core Physics Analysis	10
2.10 Task P—Fuel Element Design Analysis	10
3. Task B—Defected Fuel Behavior	17
3.1 General	17
3.2 Defected Fuel Testing	17
3.3 Sodium/Fuel/Cladding Compatibility Experiments	28
3.4 Fission Product Distribution	28
4. Task C-1—Safety Experimentation (Transient Overpower)	33
4.1 General	33
4.2 High Burnup Fuel Experiments	33
4.3 Defected or Sodium-Bonded Fuel Experiments	33
5. Task C-2—Safety Experimentation (Loss of Flow)	42
5.1 General	42
5.2 Mockup 1A	42
5.3 C10X	42
5.4 C10A/C10B	42
5.5 Planning and Design	43
6. Task E—Fuel Materials Properties and Performance Evaluation	44
6.1 General	44
6.2 Material Property Studies	44
6.3 Center Temperature Limits and Fuel Behavior	44
6.4 Fission Product—Fuel-Cladding Reactions	49
7. Task F—Fast Flux Tests and Support	65
7.1 General	65
7.2 Group F2 (Medium Power Scoping Irradiations)	65
7.3 Low Power Fuel Swelling Irradiations	68
7.4 High Power Intermediate Burnup Irradiations	74
7.5 Elevated Cladding Temperature Tests	75
7.6 Unencapsulated Fuel Pin Irradiations	75
7.7 Pre-Irradiation Transient Tests	77

Table of Contents (Continued)

	Page
7.8 Capsule F12B, In-Pile Axial Expansion and Gas Release of Mixed Oxide Fuel	77
7.9 Capsule F12P and F12Q, Effect of Radial Restraint of Fuel Swelling	77
7.10 F20 Series, Establishment of Fuel Pin Performance Limits	78
7.11 Comparison of Burnup Methods on Irradiated EBR-II Driver Fuel	78
7.12 Determination of Fission Yields for Cs-137 and Six Neodymium Nuclides from U-235 in the EBR-II Fast Neutron Reactor Spectrum	78
7.13 Characterization of Capsule Hardware	81
8. Task G-6—Core Design Analysis	82
8.1 Objective	82
8.2 Fuel Support System	82
8.3 Steady-State Irradiation Evaluation	89
8.4 Fuel Failure Effects	89
8.5 Coolant Boiling and Operating Safety Limits	98
9. Task G-10—Kinetic Analysis	103
9.1 General	103
9.2 Model Development	103
10. Task L—Cladding Development	110
10.1 General	110
10.2 Material Irradiation Performance	110
10.3 Cladding Alloy Swelling	112
10.4 Cladding Alloy Mechanical Properties	117
10.5 Materials Management	118
11. Task N—Core Physics Analysis	122
11.1 General	122
11.2 Cooperative Evaluation	122
11.3 Compilation and Distribution of Evaluated Data	122
11.4 Supplementary Analytical Interpretation of Integral Data	124
11.5 Improvements of Existing Methods for Generating Group Constants	125
11.6 Improvements of Existing Transport Theory Codes	125
11.7 Sensitivity Studies	127
12. Task P—Fuel Element Design Analysis	127
12.1 General	127
12.2 Experimental Data Analysis	127
12.3 Performance and Failure Mode Studies	128
12.4 Integral Performance and Failure Modeling	132
Appendix A — Prior Reports	A-1
Contributors	-1-
Distribution	-3-

LIST OF TABLES

Table	Title	Page
2-1	Task B—Summary of Current Mixed Oxide Capsule Irradiations	2
2-2	Task C—Current Experimental Test Series	4
2-3	Task E—Summary of Current Mixed Oxide Capsule Irradiations	8
2-4	Summary of Task F Fast Flux Testing Program in EBR-II	11
2-5	Summary of Task F Thermal Flux Irradiations	15
2-6	Summary of Task L Materials Irradiated in EBR-II	16
3-1	Fraction of Selected Fission Products Remaining in Miniloop Test Fuel Sample	30
3-2	Measured Deposition of Fission Products Inside Selected Areas of Miniloop Tubing	30
4-1	Summary of Treat Operating Parameters of Capsules C6A and C6B	33
5-1	THTE Coding and Checkout Status	43
7-1	Group F-2 Design and Irradiation Parameters	66
7-2	Maximum Diameter Changes Measured on Group F2 Capsule Tubes	68
7-3	Burnup and Fission Gas Release Fractions for the Group F6 Fuel Pins	74
7-4	Burnup and Fission Gas Release Fractions for the Group F8A Fuel Pins	75
7-5	Final Results of Burnup Exchange Analysis	80
7-6	U-235 Fission Yields in Fast Reactor Neutron Spectra	81
8-1	Comparison of Computer Usage Costs for Various Cobra-II Bundle Models	83
8-2	F9E Experimental Condition for Position 6N2 and 6N5	84
8-3	Design Loads	87
8-4	Grid Spacer Design Loads and Stresses	87
8-5	F9E Fuel Rod Bending for Annealed and Cold-Worked Type-316 Stainless Steel	88
8-6	Channel Wall Temperature Increase Due to Trapped Fuel Debris	92
8-7	Local Fuel Clad Temperature Increase Due to Trapped Fuel Debris	93
10-1	Volume Change Observed in MT-3 Specimens	112
10-2	Bimetallic Thermometer Gap Measurements	114
10-3	Specimen Identification for the L16B Capsules	116
10-4	Identification and Operation Characteristics for L16B Capsules	118
11-1	Example for Determining Accuracy Ranges of Design Enrichments	124
11-2	Results of Rapid Cross Section Adjusting Calculations Using 29- and 60-Groups for ZPR-3, Assembly 48	126

•

•

•

•

•

•

LIST OF ILLUSTRATIONS

Figure	Title	Page
3-1	B9C Final Proof Test of Defect Mechanism	18
3-2	Pump Performance and System Losses Versus Temperature	19
3-3	Break Strength of Defect Nub Versus Forging Force and Energy Input (Projection Welding)	20
3-4	A Schematic of the Test Apparatus Used to Determine Bubble Diameter	21
3-5	Water Velocity Versus Bubble Diameter Results from Gas De-entrainer Studies	22
3-6	A Sketch of the Centrifugal De-Entrainer	23
3-7	A Schematic of the Test Apparatus	24
3-8	Water Flow Rate Versus Efficiency of the Centrifugal De-Entrainer	24
3-9	Water Flow Rate Versus the Rate of Gas Entrainment of the Centrifugal De-Entrainer	25
3-10	Water Flow Rate Versus Efficiency of the Currently Used De-Entrainer	26
3-11	Post-Irradiation Appearance OD Defect Area of E1J-B2 Fuel Pin	27
3-12	Cross Sectional Appearance of E1J-B2 Fuel Pin through the Defect Area	27
3-13	Profilometer Data for E1J-B2 Fuel Pin (O/M 2.00)	28
3-14	Appearance of Fuel in the Defected and Non-Defected Regions OD E1J-B2 Fuel Pin	29
3-15	Post-Test Appearance of Miniloop Fuel Sample	31
3-16	Locations Selected for Leach Sample Analysis	32
4-1	Calculated Versus Actual Coolant Temperature TT1308 442 MW—seconds	34
4-2	Calculated Versus Actual Coolant Temperature TT1312 485 MW—seconds	35
4-3	Neutron Radiograph of Capsule C6B Following TREAT Transient (TT1308) Irradiation	37
4-4	Transverse Sections from C6B-1, Sodium-Filled Pellet Fuel Pin	38
4-5	Transverse Sections from C6B-2, Sodium-Filled Powder Fuel Pin	39
4-6	Transverse Sections from C6B-3, Gas-Filled Pellet Fuel Pin	40
4-7	Neutron Radiograph of Capsule C6A Following TREAT Transient (TT1312) Irradiation	41
6-1	The Effect of Density on the Creep Rate of (U-20 w/o Pu)O _{1.95} at 1550°C	45
6-2	The Effect of Plutonium Content on the Creep Rate of Mixed Oxides with a Constant O/Pu Ratio	46
6-3	Variations in Steady-State Creep Rate of 94% TD (U-20 w/o Pu)O _{2-x} with Stoichiometry.	47
6-4	Thermal Conductivity of 97.6% TD Stoichiometric Uranium Dioxide	48
6-5	Effect of Stoichiometry on Thermal Conductivity of (U-20 wt % Pu)O _{2-x}	48
6-6	Thermal Conductivity of Coprecipitated and Physically Mixed Pellets	49
6-7	Montage of Fuel-Cladding Interface of Pin F12C.	51
6-8	Specimen Current Images Showing the Distribution of the Elements by Relative Atomic Number for the Fuel, Cladding and Reaction Zone in Fuel Pin F12C	52
6-9	X-Ray Images Showing Distribution of Fe, Cr and Ni in Enlarged Area of F12C Shown in Figure 6-8(c). Intensity is Directly Related to Concentration.	53
6-10	X-Ray Images Showing the Distribution of Cs, Si, Te and Mo in the Area of F12C	54
6-11	Comparison of Specimen Current Image with Fe, Cr and Ni X-Ray Images in a Fuel-Clad Reaction Zone in Sample F12C	55

LIST OF ILLUSTRATIONS (Continued)

Figure	Title	Page
6-12	X-Ray Images Showing the Distribution of Mo, Cs and Si in the Area of F12C Shown in Figure 6-11.	56
6-13	Comparison of Positive and Negative Specimen Current Images with Optical Photographs of Crack Area and Cladding "Finger" of Powder Pin F12D.	57
6-14	X-Ray Images Showing the Distribution of U, Fe, Cr and Ni in Area of Powder Pin F12D Shown in Figures 6-13	58
6-15	Specimen Current and X-Ray Images Showing the Depletion of Alloy Elements in the Grain Boundaries of the Cladding Shown in Figure 6-13 of Powder Pin F12D	59
6-16	Specimen Current and X-Ray Images Showing Concentration of Cs and Si in the Grain Boundaries in an Area of Fuel-Clad Reaction and Intergranular Attack in the Powder Pin F12D	60
6-17	Radial Plutonium Concentration Across Specimen E1J-2 (1.94 O/M).	61
6-18	Radial Plutonium Concentration Across Sample E1J-6 with O/M = 1.97.	62
6-19	Radial Plutonium Concentration Across E1J-9 with O/M = 2.00	63
6-20	Pu Concentration Across Grains in the Once Molten Area of E1J Pins	64
7-1	Power Histories for Group F2 Capsules F2D, F2G, F2V, and F2X	67
7-2	Diameter Measurements from F2D Capsule Tube (F2D Fuel Pin Failed)	69
7-3	Diameter Measurements from F2G Capsule Tube	70
7-4	Diameter Measurements from F2V Capsule Tube	71
7-5	Diameter Measurements from F2X Capsule Tube	72
7-6	Photomicrographs of Transverse Section from Failed Group F6 Fuel Pin FOM	73
7-7	Maximum Diameter Changes Measured on Group F3A Fuel Pins After Fuel Burnups of ~3 at. % and ~5 at. %	77
7-8	Profilometer Traces of Fuel Pins F12P and F12Q at 10,000, 20,000, 30,000 and 40,000 MWd/Te Burnup	79
7-9	Diameter and Volume Increase for Pins F12P and F12Q. (Fuel is axially restrained)	80
8-1	Computer Model Describing Local Flow Blockage	91
8-2	Channel Response to Internal Pressure Loads	96
8-3	Channel Internal Pressure Load—Deflection vs Thickness	97
8-4	Channel Response to Externally Loaded Flat	99
8-5	Fuel Pin Beam Response to External Loads	100
8-6	Impact Velocity, V, (μ /sec)	102
9-1	Fuel Failure Model Results	105
9-2	Fuel Failure Model Results	106
9-3	Fuel Failure Model Results	107
9-4	Fuel Failure Model Results	108
9-5	Computed Voiding Rate Due to Fission Gas Injection	109

LIST OF ILLUSTRATIONS (Continued)

Figure	Title	Page
10-1	Typical Microstructures for Each Temperature Region	111
10-2	Temperature Dependence of Void Diameter	113
10-3	Bimetal Thermometer of the Type to be Used in the L-16A Capsules	115
10-4	Calibration Curve for Bimetal Thermometer	117
10-5	Capsule Assembly (L16B)	119
10-6	L16B Specimen Temperature Profiles EBR-II Row 4 Position 62.5 MW	121
10-7	Biaxial Creep Curves for Specialty Tubing	123
12-1	BEHAVE Model Predictions Versus Experimental Results	129
12-2	Radial Displacement Rate vs Radial Stress at Fuel-Clad Interface	131
12-3	BEHAVE Analysis of an F9E Fuel Pin—Subroutines Used	133
12-4	BEHAVE Analysis of Low Power Pin	135

4

7

7

7

7

7

HIGHLIGHTS

- Three out of four pins are intact after the highest known fast flux exposure of mixed oxide pins to 13.2 at. % burnup. Post-irradiation examination of four fuel pins irradiated in EBR-II to burnups up to 118,000 MWd/Te has started. The cladding fluence corresponds to an exposure near 40,000 MWd/Te in a power reactor or FFTF ($\sim 1 \times 10^{23}$ nvt, $E > 0.1$ MeV).
- Pins clad with 316 stainless steel showed substantially lower swelling than 304 clad pins at burnup near 50,000 MWd/Te. Post-irradiation examination of unencapsulated (Group F3A) fuel pins irradiated side-by-side in EBR-II to a burnup of 46,000 MWd/Te indicates that the 304L cladding increased in diameter significantly more than the 316 clad pins, 1.2% and 0.35% diameter change, respectively.
- Side-by-side high powered transient tests of gas-bonded and sodium-bonded fuel show satisfactory behavior for both. Over 50% melting occurred for the gas-bonded fuel and only 30% melting for sodium-bonded fuel. This is in accord with increased gap conductance. No deformations attributable to pressurization by sodium or sodium vapor were found.
- Modeling analysis indicates that fission-induced creep of mixed oxide results in clad restraint of fuel at high power, unlike the effect on the basis of thermal induced creep alone. At 16 kW/ft the analytical code START predicts that the fuel is sufficiently plastic under steady state irradiation conditions that it will not deform the cladding as long as a center void exists. At low power, however, the cladding may or may not deform depending on the rate of fission-induced creep.
- The maximum temperature at which clad swelling has been observed in 304 stainless steel is 680°C. Immersion density measurements on fast flux irradiated 304 stainless steel from specimen MT-3 indicate that no swelling occurs over about 680°C. This observation suggests significant change in clad swelling correlations and models.
- Erosion tests of irradiated mixed oxide fuel in a jet of flowing sodium show minor levels of washout of uranium and plutonium, but extensive leaching of fission products. Fuel from a fuel pin with a burnup of 57,000 MWd/Te was exposed to a jet of sodium at 1145°F for 18 hours in a miniature loop. Destructive examination of the miniloop indicated that a significant fraction of the fission products was leached from the fuel. The dominant soluble activity in the sodium was Cs-137.

1. INTRODUCTION

The Fast Ceramic Reactor Development Program is an integrated analytical and experimental program directed toward the development of fast reactors employing ceramic fuels, with particular attention to mixed plutonium-uranium oxide. Its major objectives follow:

- Development of a reliable, high performance fast reactor having nuclear characteristics which provide stable and safe operation.
- Demonstration of low fuel cycle cost capability for such a reactor, primarily through achieving high burnup of ceramic fuels operating at high specific power.

Progress during the period of May 1970 through July 1970 on the currently active tasks of this program is described in subsequent sections.

This is the thirty-fifth in a series of quarterly progress reports written in partial fulfillment of Contract AT(04-3)-189, Project Agreement No. 10, between the United States Atomic Energy Commission and the General Electric Company. Prior progress reports to the Commission are listed in the Appendix.

2. SUMMARY

2.1 TASK B--DEFECTED FUEL BEHAVIOR

Capsule B9C was sodium filled and inserted in the GETR pool. During capsule checkout prior to reactor start-up, water was detected in one of the flexible hoses surrounding the capsule instrument leads. Because of this off-standard condition, the capsule was removed from the pool to permit correction of this problem.

Design of capsule B9D was completed and capsule assembly is presently in progress.

Development of a projection welding technique for attachment of capsule B9B fuel pin defect nubs was initiated. A prototype centrifugal gas de-entrainer for B9B was built which is 90% efficient at removing gas bubbles from high velocity water (20 to 30 ft/sec).

Work continued on the preparation of the hazard analysis which is required before capsule B10A can be irradiated incore.

Metallographic examination of E1J-B2 fuel pin showed an average diametral increase of 0.026 in. measured at the axial mid-plane of the defect. The defect notch cut in the fully cold worked 316 SS cladding was propagated at both ends approximately 1/4 in.

Preliminary analysis of the B8C and B8D fuel microstructures indicate gap conductivity values comparable to or less than values measured for gas bonded pins with diametral gaps equal to the widths of the grey phase measured in the present pins.

Analysis of a control fuel sample and the miniloop test sample showed the following fraction of fission products remaining in the fuel after 18 hours operation: Cs-137 0.39, Ce-144 0.79, Ru-106 0.18, Sr-90 0.87, and

Pm-147 0.86. U/Pu ratios ranging from 100 to 3000 were found in the piping leach samples as compared to the original ratio of 4.

The current status of the Task B irradiation program is summarized in Table 2-1.

2.2 TASK C-1--SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)

Nondestructive examination was initiated on the final four TREAT specimens that were irradiated in EBR-II to a burnup of $\approx 60,000$ MWd/Te.

Prototype capsule C4X was completed and shipped to TREAT for transient irradiation. Preliminary analysis of the results of two calibration transients indicate a power factor some 30 to 40% lower than predicted.

Capsules C6A and C6B were subjected to test transients in TREAT. There were no fuel pin failures in C6B (3 pins) at calculated melt volumes in excess of 50% for the gas-filled pin, and 30% for the sodium-filled pin. C6A (3 pins) test results were similar (no failures) but reflected the increased severity of the test transient (485 MW-sec versus 442 MW-sec).

The current status of the Task C-1 irradiation test program is given in Table 2-2.

2.3 TASK C-2--SAFETY EXPERIMENTATION (LOSS OF FLOW)

Fabrication of the C10X fuel pin is complete and capsule fabrication has begun. The detailed thermal analysis for C10X has been completed and work continued on the analysis of the sodium voiding behavior. Preliminary specifications have been generated for capsule C10A. Work

Table 2-1
TASK B—SUMMARY OF CURRENT MIXED OXIDE
CAPSULE IRRADIATIONS

Capsule Designation	Purpose and Major Parameters	Design Parameters							Operating Conditions					Status
		Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in.-diam)	Composition			Cladding Material Stainless Steel Type	Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)		
					Density (Smear) (% TD)	PuO ₂ (%)	O/M							
B3D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	0.006	85	25	1.97	316	V-Raft GETR	1100	18	700	Capsule in disassembly prior to disposal	
B3E	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.213	23	0.008	83	25	1.98	316	V-Raft GETR	1100	18	8,700	Capsule in disassembly prior to disposal	
B3F	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.212	23	0.008	84	25	1.98	316	V-Raft GETR	1100	15	23,500	Examination completed	
B4D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	0.007	84	25	1.98	316	V-Raft GETR	1100	15	47,500	Examination completed	
B8B (2 pins)	Fuel-sodium compatibility. sodium-bonded fuel central temperature measured versus exposure time.	0.216	2	0.010	80	25	2.00	347	V-Raft	800	14	14,500	Examination completed	
		0.216	2	0.010	83	25	1.93		GETR					
B8C (2 pins)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel-sodium-clad compatibility.	0.144	2.2	0.006	86	25	1.97	316	V-Raft	1100	21	113,500	Examination initiated	
		0.204	2.25	0.017*	84	25	1.97		GETR			57,000		

GEAP-10028-35

Table 2-1 (Continued)

		Design Parameters											
Capsule Designation	Purpose and Major Parameters	Fuel							Operating Conditions				
		Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in -diam)	Composition			Cladding Material Stainless Steel Type	Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)	Status
					Density (Smear) (% TD)	PuO ₂ (%)	O/M						
B8D (3 pins)	Fuel-sodium compatibility versus density and stoichiometry and Pu content of fuel	0.214	1.75	0.0065	80-88	40	1.97	316	V-Raft	950	17.5	41,000	Examination completed
		0.214	1.75	0.0065	80-88	25	1.97		GETR			43,500	
		0.214	1.75	0.0065	80-88	25	1.94					41,400	
B9A (2 pins axially aligned)	Defected fuel pin. Fuel swelling versus stoichiometry in flowing sodium.	0.216	5	0.005	89	25	1.99	316	Special	1019	23.4	8,600	Examination completed
		0.216	5	0.005	89	25	1.95		V-Raft	971	23.4	8,600	
									GETR				
E1J-B1	Test thermal expansion clad defect mechanism	0.221	5	0.0007	94	25	2.00	316	Trail Cable	835	17.5	50	Examination completed
E1J-B2	Test thermal expansion clad defect mechanism	0.221	5	0.0007	92	25	2.00	316	Trail Cable	860	19.0	300	Examination completed

* Large gap required to accommodate sampling device

Table 2-2
TASK C—CURRENT EXPERIMENTAL TEST SERIES

		Design Parameters												TREAT		Nondestructive			
				Fuel Density (% TD)		Fuel Composition						Pre-Irra- diated		Transient Irradiation		Exam†			
Capsule	Test Objective	Diameter (inch)	Fuel Length (inch)			%UO ₂	%PuO ₂	%U-235	O/M	Cladding Material	Fabri- cation	Yes	No	Calib- ration	Test	Before Trans.	After Trans.	Destruc- tive Exam.	Final Report
				Pellet	Smeared														
C4A	Axial restraint at fuel melting (≈5000°F).	0.2159	14.23	93.8	90.4	80	20	0	1.983	316SS	C			X	C	C	C	C	C
C4B	Axial restraint similar to C4A; test to identify percent melting necessary to cause cladding failure.	0.2155	14.21	94.5	91.2	80	20	0	1.983	316SS	C			X	C	C	C	C	C
C4C	Axial restraint similar to C4A; test to observe effect of fuel density on incipient failure point identified in C4A.	0.2164	14.19	83.1	80.4	80	20	0	1.983	316SS	C			X	C	C	C	C	C
C4D	Axial restraint designed at ≈3000°F - simulating average FCR fuel temperatures; test to determine if cladding deformation occurs in transient to fuel melting (≈5000°F).	0.2154	14.22	83.7	80.1	80	20	0	1.983	316SS	C			X	C	C	C	C	C
C4E	Axial restraint designed to simulate the restraint of C4A after steady-state irradiation; specific test goals to be determined after nondestructive examination of pin following removal from EBR-II.	0.2159	13.46	83.2	80.5	72	28	92.7	1.974	316SS	C	C*		P	P	NC	P	P	
C4F	Identical to C4E (transient magnitude will be varied).	0.2160	13.51	84	80.5	72	28	92.8	1.975	316SS	C	C*		P	P	NC	P	P	

GEAP-10028-35

Table 2-2 (Continued)

		Design Parameters										Pre-Irra- diated		TREAT Transient Irradiation		Nondestructive Exam†		Destruc- tive Exam.	Final Report
Capsule	Test Objective	Diameter (inch)	Fuel Length (inch)	Fuel Density (% TD)		Fuel Composition				Cladding Material	Fabri- cation	Yes	No	Calib- ration	Test	Before Trans.	After Trans.		
				Pellet	Smeared	%UO ₂	%PuO ₂	%U-235	O/M										
C4G	Axial restraint similar to C4A. This test (and C4H) will be compared to C4A and B to determine the effect of burnup on failure threshold.	0.2154	13.50	94.4	90.5	72	28	89.3	1.973	316SS	C	C*		P	P	C	P	P	
C4H	Identical to C4G (transient magnitude will be varied).	0.2152	13.45	94.4	90.4	72	28	89.3	1.976	316SS	C	C*		P	P	C	P	P	
C4I	Typical powder fuel specimen. Test to determine failure threshold relative to pellet pins.	0.2204	13.48	powder	83.5	80	20	0	1.973	316SS	C		X	C	C	C	C	C	C
C4J	Identical to C4I (transient magnitude will be varied).	0.2205	13.45	powder	83.5	80	20	0	1.973	316SS	C		X	C	C	C	C	C	C
C4K	This test (and C4L) will be compared to C4I and J to determine effect of burnup on failure threshold of powder fuel.	0.2205	13.50	powder	84.1	72	28	93.0	1.972	316SS	C	C*		P	P	NC	P	P	
C4L	Identical to C4K (transient magnitude will be varied).	0.2206	13.47	powder	84.2	72	28	93.0	1.972	316SS	C	C*		P	P	NC	P	P	
C4M	Determination of failure threshold for low-power (6–10 kW/ft), high-burnup fuel.	0.215/ 0.217	13.5	84/94	80/90	70/80	30/20	~93	1.97/ 1.98	316SS	—††	X		P	P	P	P	P	
C4X	Checkout of the redesigned C4 TREAT capsule (remote assembly techniques and TREAT characteristics) intended for the transient irradiation of preirradiated specimens C4E–H and K–M.	0.2186	13.48	96.2	91.0	100	0	92.8	2.00	316SS	C		X	C	P	C	P	P	

Table 2-2. (Continued)

		Design Parameters										TREAT									
				Fuel Density (% TD)		Fuel Composition						Pre-Irra- diated		Transient Irradiation		Nondestructive Exam†			Destruc- tive		Final Report
Capsule	Test Objective	Diameter (inch)	Fuel Length (inch)	Pellet	Smeared	%UO ₂	%PuO ₂	%U-235	O/M	Cladding Material	Fabri- cation	Yes	No	Calib- ration	Test	Before Trans.	After Trans.	Exam			
C6A,-1	Test to evaluate transient performance of defected or sodium bonded fuel; determine relative performance limits and failure mechanisms of pellet and powder fuel under transient irradiation conditions. Sodium logged powder fuel is to maximize fuel-sodium heat transfer.	0.212	24	86	80	100	0	10	2.000	316SS	C		X	C	C	C		C	I		
-2		0.2203	24	powder	80	100	0	10	2.000	316SS											
-3		0.212	24	86	80	100	0	10	2.000	316SS											
C6B,-1	Identical to C6A (transient magnitude will be varied).	0.212	24	86	80	100	0	10	2.000	316SS	C		X	C	C	C		C	I		
-2		0.2203	24	powder	80	100	0	10	2.000	316SS											
-3		0.212	24	86	80	100	0	10	2.000	316SS											

P — Planned

I — Initiated

NC — Nearly completed

C — Completed

* EBR-II Fast flux, burnup goal $\approx 50,000$ MWd/Te** GETR thermal flux, burnup goal $\approx 20,000$ MWd/Te

† Consists of X-rays, neutron radiographs, deencapsulation, cladding, diametral measurements, gamma transmission scan

†† Actual fuel pin has yet to be chosen.

continues on the updating of the THTE program, as well as on the comparison study of the Mark II loop and the Forced Convection Capsule. Diagnostic examination of the heaters from Mockup 1A is being performed to determine potential for changes to permit more extended operation under loss-of-coolant conditions.

2.4 TASK E--FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

- Compressive creep test data on coprecipitated mixed oxides indicates that creep rates increase with: increasing O/M ratios, increasing plutonium contents, and decreasing densities.
- Thermal conductivity measurements on a mixed oxide pellet at several stoichiometries showed that the stoichiometric compound had higher conductivity than hypostoichiometric compounds at temperatures below about 1500°C. At higher temperatures the differences became insignificant.
- Thermal conductivity measurements on pellets prepared by coprecipitation and by mechanical mixing showed no significant differences over the temperature range of 800 to 2300°C.
- Microprobe examination of the fuel-cladding interface of pins F12C and F12D, which operated with cladding temperatures $\geq 1300^\circ\text{F}$, indicated that Cs, Si, Te and Mo had concentrated at the cladding interface, with chromium being the most prevalent cladding element at that location. In some cases, iron and nickel appear to have migrated furthest beyond the reaction layer into the fuel, while in other instances, chromium has undergone greater transport.
- Electron microprobe examination of F12C and F12D cladding that exhibited limited intergranular attack revealed that the grain boundaries in the region of attack were depleted in iron, chromium and nickel, and that silicon, tellurium and cesium were present. Cladding that showed no microstructural evidence of intergranular attack showed no such compositional changes.
- Electron microprobe examination of fuel pins that had melted during operation in GETR revealed no sharp composition changes at the apparent radius of melting.

The current status of the Task E irradiation test capsule program is given in Table 2-3.

2.5 TASK F--FAST FLUX TESTS AND SUPPORT

The Group F2 and F3A experiments were disassembled and an initial nondestructive examination of the pins

was performed at the EBR-II site. The Group F2 and F3A pins, respectively represent the highest burnup encapsulated and unencapsulated fuel pins to be examined under the FCR program. The peak burnup of the peak pin in the Group F2 and F3A tests was 118,000 MWd/Te (13.2 a/o)** and 52,000 MWd/Te (5.79 a/o),** respectively.

Neutronradiographs of the Group F2 capsules indicate that only F2D has failed. This pin was clad in 347 stainless steel and contained fuel of 94% smeared density having an initial oxygen-to-metal ratio of 2.00. The fuel in F2D was axially restrained. Peak $\Delta D/D$ of the 304 stainless steel capsule bodies was $\sim 2.5\%$.* Peak exposure was over 0.9×10^{23} nvt ($E > 0.1$ MeV).**

Diameter measurements on the Group F3A fuel pins indicated that the 304L clad fuel pins increased in diameter significantly more than the 316 clad pins. The diameter increases of the 304L and 316 clad pins were $\sim 1.2\%$ and 0.35% , respectively. Peak exposure was $\sim 0.4 \times 10^{23}$ nvt ($E > 0.1$ MeV).**

The eighteen Group F3B (encapsulated) fuel pins were removed from their subassembly for an interim examination. The average burnup and peak fluence on the peak exposure Group F3B pin was 78,800 MWd/Te (8.86 at. %)** and $\sim 0.7 \times 10^{23}$ nvt ($E > 0.1$ MeV).** Neutronradiography of the F3B capsules indicated that one pin, E1H-33, failed. Pin E1H-33 was clad with 304 stainless steel and contained fuel of 90% smeared density and 1.98 stoichiometry.

The design concept for incorporating triangular flow restrictors in the outer coolant channels of the Group F10A, F10B, and F10C subassemblies was approved by the ANL Reactor Engineering and Fuel Examination Facility. Fabrication of the triangular restrictors was initiated.

Approval to fabricate F9E fuel pins was received from AEC-RDT. Fabrication of fuel and cladding was initiated.

Examination of the results from capsules F12P and F12Q after burnups of 10,000, 20,000, 30,000, and 40,000 MWd/Te has shown that 0.015 inch wall cladding restrains mixed oxide fuel expansion more effectively than 0.010 inch wall cladding.

Results of the cooperative burnup analysis exchange indicate that experimentally determined burnup of samples irradiated in the EBR-II are approximately 16% lower than calculated using fission rates published for the EBR-II.

* Based on diameter measurements taken at EBR-II using a dial gauge.

** All fuel burnup and fluence values are those reported by the EBR-II project or calculated from values supplied by EBR-II.

Table 2-3
TASK E—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

Capsule	Description	Design Parameters							Operating Conditions				Status
		Diameter (inch)	Length (inch)	Fuel		PuO ₂ (%)	O/M	Cladding Material	Irradiation Facility	Cladding		Burnup (MWd/Te)	
				Pellet	Smeared					Surface Temp (°F)	Peak Power (kW/ft)		
E1AA	Continuous in-pile thermal conductivity.	0.219	6.0	92.0	91.0	20	1.97	316	GETR	1300	15	0	Report in publication
E1E	Relative smeared conductivity at high burnup irradiation to high burnup and rabbit test E1F.	0.218	1.9	94.7	94.5	20	2.00	347	MTR	875	20	85,000	Report in review
E1F		0.218	1.9	94.6	94.8	20	2.00		MTR	654/875	20/28	77,000	
E1J-1	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.94	316	GETR	1100	23	2,000	Report in process
E1J-2	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.94	316	GETR	1100	23	2,000	Report in process
E1J-3	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.94	316	GETR	1100	23	2,000	Report in process
E1J-4	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	316	GETR	1100	23	2,000	Report in process
E1J-5	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	316	GETR	1100	24	2,000	Report in process
E1J-6	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	316	GETR	1100	23	2,000	Report in process
E1J-7	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	316	GETR	1100	24	2,000	Report in process
E1J-8	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	316	GETR	1100	23	2,000	Report in process

Table 2-3 (Continued)

Capsule	Description	Design Parameters							Operating Conditions				Status
		Diameter (inch)	Length (inch)	Fuel		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temp (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)	
				Density (% TD)		PuO ₂ (%)	O/M						
				Pellet	Smeared								
E1J-9	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	316	GETR	1100	23	2,000	Report in process
E2C-1	High burnup. Reference fuel irradiated at near reference thermal conditions.	0.218	5	84.5	82.2	25	2.00	347	GETR	1060	20	121,000	Report in review
E2C-2		0.218	5	94.3	91.4	25	2.00		GETR	1060	20	113,000	
E2E	Fuel swelling. Four separate pins at three power levels	0.100	3	98.5	95.4	25	2.00	304	MTR	785	11.6	199,000	Report in review
E2F		0.100	3	96.9	95.1	25	2.00		MTR	655	9.4	172,000	
E2G		0.100	3	97.0	95.2	25	2.00		MTR	560	7.9	255,000	
E2H		0.100	3	96.6	94.8	25	2.00		MTR	785	11.6	330,000	
E2J-1	High burnup. Annular versus solid pellet.	0.218	5	90.4	89.5	25	1.98	316	GETR	1100	21	124,000	Report in review
E1J-2		0.218	5	96.0	88.5	25	1.98			1100	21	125,000	
E2R	Center void formation rate at lower burnup.	0.216	10	94.0	91.0	25	1.98	316	GETR	1000	20	0	Fabrication in process
E5A-1	Powder and pellet fuel comparison at low burnup.	0.218	4.9	86.8	84.3	25	2.00	347	GETR	1100	27	2,000	Report in review
E5A-2		0.218	4.9	84.1	84.1	25	2.00		GETR	1100	27	2,000	
E5B-1	Powder and pellet fuel comparison at low burnup.	0.218	4.9	85.8	84.1	25	2.00	347	GETR	1400	20	138,000	Report in review
E5B-2		0.218	4.9	84.1	84.1	25	2.00		GETR	1400	20	138,000	

The current status of the Task F irradiation program is summarized in Tables 2-4 and 2-5.

2.6 TASK G-6—CORE DESIGN ANALYSIS

The F9E grid assembly test conditions in EBR-II have been selected. Based on operation of EBR-II at 62.5 MWt, the assembly will be located in the 6N2 or 6N5 core location.

Calculations of fuel rod bending have been made and corresponding rod deflection, stresses and grid loads were determined for the F9E assembly for various swelling and irradiation induced creep correlations. Typical values after 10,200 hours are 8000 psi bending stress, 2 pound grid load and 0.002 inch fuel rod deflection.

The assessment of the needs for design information regarding fuel failure and sodium boiling has continued with sensitivity studies in the following areas: The thermal and mechanical effects of fuel swelling and local flow blockage; rod and channel response to pressure pulses; molten fuel/clad sodium thermal interaction; and pressures due to sodium hammer and vapor bubble collapse.

2.7 TASK G-10—KINETICS ANALYSIS

The FREADM coupled kinetics-thermal analysis code has been coupled to a revised version of the MARS two dimensional modified Bethe-Tait calculation. The coupled code permits calculation of design basis accidents for fast reactor containment from the point of initiation through core disassembly. The checkout of the coupled FREADM-MARS code is in progress. Wrapup and restart options have been added to FREADM which permit problem restart from pre-computed transient or steady state conditions.

The improved core heat transfer and voiding computational model has reached its operational state as an independent model. The extended model will simulate flow anomalies involving loss-of-coolant and flow-blockage. Several subsequent bubbles can be described as well as back-flow, slug-ejection, and re-entry phenomena.

A parametric analysis is underway to examine the voiding rate due to fission gas release from a ruptured fuel pin. The simplified analysis considers the dynamics of the coolant after a large rupture in a fuel pin has caused a reversal of coolant flow.

2.8 TASK L—CLADDING DEVELOPMENT

Irradiation of three L-6 series capsules continued in S/A X018B. The L-15 series and MT-4 have been assigned to a new subassembly, X080. The four L-2 series capsules have been discharged and returned to VNC for post irradiation examination. The total fluences are 9.0, 7.9, 6.9, and

$0.94 \times 10^{22} \text{ n/cm}^2$ for the L-2, L-6, MT-4, and L-15 series respectively. Fabrication of the L-16B capsules has been completed and 19 capsules shipped to EBR-II. Design of the L-16A capsules has been completed. A bimetal thermometer has been tested in the temperature range 900°F to 1350°F. Test results show that temperature can be measured within $\pm 15^\circ\text{F}$ with this design in this temperature region. These thermometers will be used in the L-16A capsules. Transmission electron microscopy examination of highly irradiated Type 304 showed that void size increases with increasing temperature of irradiation in the temperature range 400 to 650°C. Above 682°C voids were not observed. The upper temperature limit for void formation was estimated to be between 650 and 682°C. The maximum void size observed exceeded the thickness of the foils $> 2000 \text{ \AA}$. Stress-rupture testing of Type 304H and Type 316H tubing with various degrees of cold working is continuing. Results indicate that an initial strength advantage for Type 304H may be realized with light to moderate cold working but that the long-term stress-rupture properties of such material is inferior to that of annealed material.

The status of the current Task L irradiation program is summarized in Table 2-6.

2.9 TASK N—CORE PHYSICS ANALYSIS

Topical reports were issued describing the Pu-239 cross section evaluation performed for ENDF/B, Version II, and the nuclear data code ENDRUN-1.

Analyses have been started for the purpose of using integral quantities measured in ZPPR, Assembly 2 and the SEFOR reactor to evaluate the accuracies of current nuclear design methods. Emphasis will be placed on the computation of design quantities with uncertainties extrapolated from the analysis of integral measurements.

Preliminary comparisons of 29- and 60-group calculations using the rapid cross section adjusting procedure indicate good accuracy with 29 groups for k_{eff} and breeding ratio, but a higher Doppler coefficient and softer low energy spectrum with the 60 groups.

Modification of the DOT2DB code to standard interfaces was completed and has been distributed to three members of the *de facto* Committee on Computer Code Coordination. Conversion to the GE635 of the other sections of coding which were modified to the standard interfaces has begun.

2.10 TASK P—FUEL ELEMENT DESIGN ANALYSIS

The START model has been formulated to predict the mean interference between fuel and cladding during the first startup of a fresh fuel pin. This model has been refined

Table 2-4
SUMMARY OF TASK F FAST FLUX TESTING PROGRAM I, R II*

Group	Purpose and Major Parameters	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins in S/A	S/A	Position	Range of Average Pin Powers ⁽¹⁾ kW/ft	Target Burnup		Ave. Calc. Burnup ⁽²⁾		Peak Calc. Fluence ⁽³⁾ (nvt X 10 ²³)		Status
										At.%	MWd/T _e	At. %	MWd/T _e	Total	E>0.1 MeV	
F-0	Lower power swelling rate of high density fuel at various burnup levels	5	07/16/65		08/15/68	1	XG02	7A1	8.71	3.91	35,000	3.88	34,702	0.322	0.2370	Post exam complete
			07/16/65	10/03/71	-----	2	XG03	7D1	5.39-5.84	8.95	80,000	6.79	60,730	0.585	0.4269	In reactor
			07/16/65	06/22/72	-----	2	XG04	7B1	5.35-6.15	11.18	100,000	7.52	67,272	0.654	0.4812	In reactor
F-1	Short time performance proof test.	6	05/06/65		05/23/65	6	XG01	4F2	12.17-14.31	0.22	2,000	0.21	1,904	0.017	0.0149	Reported in GEAP-5570
F-2	Medium power scoping irradiation. Major Parameters include smear density, stoi- chiometry, cladding- coolant interactions and burnup.	21	09/03/65		03/06/68	9	XG05	4C2	11.18-13.74	8.39	75,000	6.69	59,800	0.569	0.5011	Report in Review
			09/03/65		02/20/67	12	XG06	4E2	8.34-14.65	5.59	50,000	5.02	44,897	0.422	0.3613	Reported in GEAP-13549
			02/23/60		05/18/70	4	X050	4C2	10.50-12.58	11.19	100,000	11.75	105,058	1.034	0.8861	Post exam initiated
F-3A	Bare pin irradiation to evaluate fuel clad- ding and cladding- coolant interaction	16	08/15/68		06/30/69	16	X040	5B2	12.59-14.32	5.62	50,000	2.88	25,601	0.236	0.2008	Interim exam
			10/04/69		05/18/70	16	X040A	5B2	12.25-13.91	5.62	50,000	5.17	45,978	0.430	0.3658	Post exam initiated
F-3B	Density and stoi- chiometry effects for encapsulated fuel pins.	18	11/21/67		05/18/70	18	X027	4B3	11.20-12.49	8.99	80,000	8.86	78,819	0.754	0.6459	Interim exam
F-4	Low density fuel behavior at high heat ratings	9	05/09/66		06/28/67	9	X011	2F1	13.57-14.30	5.64	50,000	3.64	32,255	0.298	0.2562	Post exam complete
			08/15/68		12/24/68	7	X039	3C2	13.12-13.55	9.03	80,000	4.85	42,964	0.400	0.3436	Interim exam
			01/10/70	10/07/70	-----	7	X070	3E1	12.47-13.03	9.03	80,000	8.04	71,215	0.676	0.5812	In reactor
F-5	Fuel performance at peak FCR cladding temperatures.	19	05/28/69	11/20/71	-----	19	X064	4F2	12.29-13.08	11.25	100,000	5.51	42,022	0.458	0.3929	In reactor
F-6	Fuel-cladding inter- actions under accele- rated swelling.	4	03/24/66		04/24/69	4	X010	7F3	8.45-9.17	5.60	50,000	5.71	51,014	0.480	0.3936	Post exam complete
F-7	Pre-irradiated axially restrained fuel prior to transient test. Main variable is fuel density.	6	12/23/67		12/23/67	2	X015	4A2	-----	-----	-----	---	-----	-----	-----	Terminated
			01/23/69		05/18/70	4	X050	4C2	-----	5.48	50,000	5.86	53,400	---	-----	Terminated
F-8A	Effect of density on fuel swelling rate.	16	01/13/67		08/02/69	7	X019	6D2	8.36-8.95	5.63	50,000	4.97	44,161	0.412	0.3420	Post exam complete
			01/13/67		01/30/70	9	X020	6B5	8.20-8.91	11.26	100,000	6.5	54,615	0.524	0.4377	Interim exam
			05/24/70	04/09/72	-----	9	X081	7C5	7.06-8.06	11.26	100,000	6.67	60,207	0.580	0.4831	In reactor
F-8B	Effect of void deployment on fuel cladding reaction.	19	07/25/68	01/27/71	-----	19	X036	7E1	4.60-6.47	5.63	50,000	4.47	39,748	0.378	0.2835	In reactor
F-9A	High power, high burnup, unencapsulated fuel pin irradiations.	37	02/20/69	11/21/71	05/18/70	37	X043	4D2	11.78-13.03	11.26	100,000	5.57	49,501	0.464	0.3974	Interim exam and reconstitution

Table 2-4 (Continued)

Group	Purpose and Major Parameters	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins in S/A	S/A	Position	Range of Average Pin Powers (1) kW/ft	Target Burnup		Ave. Calc. Burnup (2)		Peak Calc. Fluence (3) (nvt X 10 ²³)		Status
										At. %	MWd/T _e	At. %	MWd/T _e	Total	E>0.1 MeV	
F-9B	Medium power, high burnup, unencapsulated fuel pin irradiations.	37	05/23/69	08/12/72	----	37	X062	6F3	8.81-10.61	11.26	100,000	4.15	36,922	0.342	0.2894	In reactor
F-9C	High power, high burnup, unencapsulated fuel pin irradiations.	37	04/03/69	01/06/72	07/15/70	37	X056	5C2	10.26-11.27	11.24	100,000	5.67	50,455	0.472	0.4022	Interim exam and reconstitution
F-9D	Medium power, high burnup, unencapsulated fuel pin irradiations.	37	04/24/69	12/07/72	----	37	X058	6F1	7.37-8.75	11.25	100,000	4.18	37,194	0.348	0.2885	In reactor
F-9E	High power, high burnup.	37	----	----	----	37	----	---								Design in progress
F-10A	Cladding performance at peak cladding temperatures.	19	----	----	----	19	----	---								Fabrication in progress
F-10B	Cladding performance at peak cladding temperatures.	19	----	----	----	19	----	---								Fabrication in progress
F-10C	Fuel pin performance at peak cladding temperatures.	19	----	----	----	19	----	---								Design in progress
F-11a	Fuel pin performance at peak cladding temperatures.	37	----	----	----	37	----	---								Fabrication in progress

* All calculated power, burnup, and fluence numbers are based on the fission rates provided in the "Guide for Irradiation Experiments in EBR-II."

(1) Values are the calculated axial average powers. Ratio of peak to average power is 1.12.

(2) Burnup values are axial average values for the pin in the subassembly with the highest fuel burnup regardless of position in subassembly.
All values are corrected for fuel depletion. Burnup values in MWd/T_e are based on a value of 192 MeV/fission and calculated from data supplied by EBR-II.

(3) Fluence values reported are calculated for the peak axial flux for the pin in the subassembly located closest to the center of reactor core. Based on values supplied by EBR-II.

Table 2-5

SUMMARY OF TASK F THERMAL FLUX IRRADIATIONS

Capsule New No. Former No.		Description	Design Parameters						Operating Conditions					Status
			Fuel				Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)			
			Diameter (inch)	Fuel Length (inch)	Density (%TD)							Composition		
					Pellet	Smearcd	PuO ₂ (%)	O/M						
F12A	(E5F)	Linear fuel movement versus burnup	0.214	10.0	92.0	87.1	25	1.98	Type-316 stainless steel	GETR	850	17	44,000	REPORT IN PROCESS
F12B	—	Axial fuel movement and fission gas release versus burnup	0.214	10.0	92.0	87.1	25	1.97	Type-316 stainless steel	GETR	1200	20	100,000*	IN FABRICATION
F12C	(E5C-1)	Powder, pellet, and annular fuel comparison at high burnup	0.215	16.0	87.5	82.9	25	1.96	Type-316 stainless steel	GETR	1160	19.5	137,000	REPORT IN PROCESS
F12D	(E5D-1)	Powder, pellet, and annular fuel comparison at high burnup	Powder	16.0	81.8	81.8	25	1.97	Type-316 stainless steel	GETR	1160	19.5	141,000	
F12E	(E5E-1)	Powder, pellet, and annular fuel comparison at high burnup	0.215 (Annular)	16.0	95.0	83.3	25	1.96	Type-316 stainless steel	GETR	1160	19.5	130,000	
F12P	(E2P)	Effect of radial restraint on fuel swelling (15 mil tubing)	0.219	2.0	92.0	91.0	25	1.98	Type-316 stainless steel	GETR	1160	15	~50,000	Irradiation
F12Q	(E2Q)	Effect of radial restraint on fuel swelling (10 mil tubing)	0.229	2.0	92.0	91.0	25	1.98	Type-316 stainless	GETR	1160	15	~50,000	

*Target

Table 2-6
SUMMARY OF TASK-L MATERIALS IRRADIATED IN EBR-II

Series	Row-S/A	Capsules	Capsule Materials	Number of Specimens		Total Fluence			Projected Completion Date**
				Biaxial	Tensile	Target, 10 ²³ nvt	July 1970 10 ²³ nvt† 10 ²³ nvt‡		
Irradiation Completed									
L-2'	4 – XG06	5	One each of Types 304, 316, 321, and 347 stainless steels, and Incoloy-800.	15	80	0.38	0.34	0.28	Feb., 1967
L-4	4 – X009	2	Type 316 stainless steel	10	20	0.20	0.19	0.16	Dec., 1966
L-4'	2 – X014	5	One each of Types 304, 321, and 347 stainless steels; and two of Incoloy-800.	25	50	0.20	0.17	0.14	Apr., 1967
L-10	4 – X016	10	One each of Type 321 and 347 stainless steels; two of Type 304 stainless steel; three each of Type 316 stainless steel, and Incoloy-800.	70	60*	0.20	0.29	0.24	Oct., 1968
L-14	4 – X025	19	Five each of Type 316 stainless steel and Incoloy-800; four of Type 304 stainless steel; three of Type 321, two of Type 347 stainless steel.	133	114*	0.20	0.31	0.26	Dec., 1968
L-21	1 – XG05 & 3 – X039	1	Type 321	3	16	0.76	0.57	0.48	Dec., 1968
MT-3‡	4 – XA08	1	16 Incoloy-800, 16 Inconel-625 & 16 Hastelloy-X tensile specimens	None	48	1.00	0.69	0.58	June, 1968
L-2	4 – XG05 4 – X050	4	One each of Types 304, 316, and 347 stainless steels, and Incoloy-800.	12	64	3.2	0.90	0.75	May, 1970
Irradiation in Progress									
L-6	2 – X018 4 – X018A 4 – X018B	9	One each of Types 321 and 347 stainless steels; two each of Type 304 stainless steel and Incoloy-800; three of Type 316 stainless steel.	63	126*	1.0	0.79	0.66	Sept., 1970
L-15	3 – X039	4	All five alloys	None	288	0.50	0.09	0.07	1971
MT-4‡	4 – XA08	1	16 Incoloy-800, 16 Inconel-625 & 16 Hastelloy-X Tensile Samples	None	48	3.2	0.69	0.58	1972

* Includes specimens of experimental alloy of base composition Fe, (9-14)Ni, (9-14)Cr, (0.004-0.05)C.

** Assuming EBR-II continues to operate at 50 MW with 50% load factor.

† Fluence based on EBR-II core physics conversion factors without any correction factors.

‡ Fluence based on EBR-II core physics conversion factors with 1/1.2 correction factor.

† Capsules from previous AEC program under Contract AT(04-3)-189, PA-29.

and now is capable of predicting the local deformation along the entire length of the fuel pin. After addition of the experimentally determined component of cladding swelling, this model gave excellent agreement with the observed profilometry results from pin F2H which was irradiated in EBR-II.

A simplified model for calculation of fuel-cladding mechanical interaction during steady-state operation has been encoded (SS1) and is now operational within the STEADY subroutine of the BEHAVE fuel element simulation code. This model incorporates the effects of flux induced creep in the cladding and fission induced creep in the fuel. An advanced version of a steady-state interaction

model has been encoded (SS2) and is currently being debugged. This advanced model includes provisions for cracks and crack healing of the fuel and for axial flow of the more plastic fuel. Both items may constitute important effects.

The BEHAVE fuel element simulation model and computer code is now operative with the three most important of the subroutines (START, RESTRU, and STEADY) and a number of test cases have been run. The STATUS code for predicting local deformation and lifetime of the cladding is still being debugged and is currently being used to calculate the effects of thermal stresses and the differential clad swelling stresses.

3. TASK B—DEFECTED FUEL BEHAVIOR

3.1 GENERAL

The purpose of Task B is to determine analytically and experimentally the manner and consequences of propagation of fuel defects, the behavior and performance of the cladding interface in intimate contact with sodium, the distribution and rate of release of fission products from high performance mixed-oxide fuel, and the identity of the fission product species which can escape to the primary coolant and cover gas.

The technical feasibility of fast reactors depends on quantitative knowledge of these factors and on designs to limit the consequences of the release of fission products. It appears attainable to limit long-term activity release from vented fuel to essentially Kr-85 so that primary system maintenance conditions are equivalent to (or possibly better than) nonvented designs.

The technologies of defected, vented, and sodium-bonded fuels are closely related. While Task B is directed primarily at studying the behavior of defected fuel, it has also included development of vented and sodium-bonded fuels. Although vented pins are a prospective reference design for 300 MW and 1000 MW plants, funding limitations have made it necessary to suspend this aspect of the work.

3.2 DEFECTED FUEL TESTING

3.2.1 Forced Convection Capsule B9C

Capsule B9C is designed to investigate the performance of defected hypostoichiometric mixed oxide fuel at prototypic LMFBR conditions in flowing sodium.

Capsule fabrication and assembly were completed on this capsule.

Concurrent with capsule assembly, final proof-testing of the entire system for producing fuel pin defects was successfully conducted in the FCC sodium fill system furnace at 1000°F. As shown in Figure 3-1, the top fuel pin nub defected with a force of 16 pounds; the bottom fuel pin nub defected with a force of 18 pounds. In addition, a capsule transfer test was conducted with a mockup that duplicated the capsule weight and size. This mockup also incorporated a dummy fuel pin with a defect nub.

The practice transfer included the removal of the capsule mockup from the sodium fill furnace into the heater can, transfer of the heater can and capsule to GETR, and transfer through the GETR equipment airlock to the third floor of the reactor. After this practice transfer was completed, a check of the defect nub showed no deformation.

Final quality assurance of capsule B9C was performed using X-radiography and neutrography. Results of analysis for argon in the dummy pins tagged as standards with the fuel pins indicate that an argon content of ~ 11.5% was achieved in the B9C fuel pins (argon-36 in the upper pin gas plenum and argon-40 in the lower pin gas plenum).

The sodium in the cleanup and fill system was circulated and cold-trapped for two days to ensure high purity sodium. A plugging run made immediately after valving sodium into capsule B9C indicated a plugging temperature of 210°F (< 5 ppm oxide in the sodium). Sodium was then circulated through the capsule for approximately 24 hours at 1000°F. During this period, approximately 10% of the sodium flow exiting from the capsule was bypassed to a sample holder containing niobium and vanadium wire (10 mil diameter). These wire samples will be analyzed for oxygen content to confirm the indicated low oxygen level

Figure 3-1. B9C Final Proof Test of Defect Mechanism

in the sodium. Bulk sodium samples will also be analyzed. After the capsule cleanup operation, a second plugging temperature run indicated 210°F.

The capsule sodium level was adjusted by back purging the upper plenum with helium and was verified by X-radiography. No alpha contamination of the loop sodium was detected. This established that the capsule fuel pins had not defected during sodium fill.

Capsule B9C was successfully transported from the sodium fill system to the GETR and inserted into the special V-RAFT in the reactor pool. Sodium flow was established and equilibrium temperatures of ~ 600°F were measured by capsule thermocouples. The flowmeter indicated the expected 16 gpm sodium flowrate. Figure 3-2 shows the B9C system loss curve plotted against the manufacturer's predicted pump performance curves at 1200°F and 600°F. During a heat balance calibration, the main (3 KW rated) capsule heater failed at 70% rated power.

During final capsule checkout prior to reactor start-up, water was detected in one of the flexible hoses surrounding the capsule instrument leads. Because of this off-standard condition, the capsule was removed from the reactor pool to permit correction of the leak and was rescheduled for later irradiation.

Water was drained from the instrument lead hose and pressurization with helium showed a crack in a weld where the hose connected to a junction box bulkhead. The weld was repaired and the instrument lead system dried by heating and evacuating approximately for 1 week. All thermocouple leads and the flowmeter leads were intact and operable after completion of the repair. Continuity checks made on the four solenoid valves on the failure detection system showed integrity and the valves were cycled on and off to verify operability.

An X-ray of the failed heater showed the leads had shorted and burned out probably due to the ingress of the water noted above. The arcing also burned a hole in the stainless steel tube which shielded the leads and comprised part of the capsule's secondary containment. In order to repair the secondary containment, the heater leads were cut and seal welded at the next containment bulkhead. This re-established double containment but eliminated the heater.*

* Loss of the heater is acceptable since it is only used as a backup heat source and for calorimetric calibration of the flowmeter and is not required for capsule insertion or operation.

Figure 3-2. Pump Performance and System Losses Versus Temperature

To determine if the fuel had been inadvertently defected during the increased capsule handling encountered in the repair operations, the failure detection system was operated using a temporary sampling station. This system directly samples the plenum gas above the capsule sodium to detect the A-36 and A-40 tag gas which had been fabricated into the fuel specimens. Analysis of the sample showed no argon but did indicate high nitrogen. Since air contamination was a definite possibility, a second sample was taken using the primary sampling station. This second sample also showed contamination by air as evidenced by the presence of oxygen in the sample. Since oxygen cannot exist in any significant quantity in equilibrium above hot sodium, the air must have been aspirated through a leak in the fuel failure detection system lines or in chemistry. To determine the source of the leak, the capsule flex hoses were pressurized to 50 psig with Ne (they were previously filled with ambient air) and a third sample taken and analyzed. Analysis showed the presence of Ne indicating that the flex hose internals were open to the gas sample lines. Such communication could lead to the spread of fission gases throughout the entire instrument lead system and cause gross operator exposure and handling difficulties during capsule irradiation.

The capsule is being examined to determine the cause of failure and to determine the extent of repairs that will be required before capsule irradiation is begun.

3.2.2 Forced Convection Capsule B9D

Design of forced convection capsule B9D has been completed. This experiment will investigate the performance of defected mixed oxide fuel in flowing sodium at 1200°F. Capsule B9D has the same basic design as B9C but will operate with sodium temperatures of 1200°F rather than 1000°F. A summary of the important test parameters have been previously reported.¹ The capsule design incorporates two axially-aligned fuel pins. Both fuel pins will be defected at the same time after a fuel burnup of ~1000 MWd/Te. This short period of irradiation (3 to 4 days) will allow an equilibrium fuel microstructure to develop prior to defecting.

The B9D defect mechanism is identical to that of B9C and consists of a mechanical plug design. The development and reliability testing of this design were presented previously in conjunction with capsule B9C. The defect in the top pin is located in the gas plenum region, and in the bottom pin at the mid-plane of the active fuel.

A failure detection system identical to that in B9C has also been incorporated into B9D. Analysis of the capsule cover gas during irradiation for traces of argon-36 and -40 will permit positive identification of each fuel pin failure.

Fabrication and testing of capsule components have been completed and capsule assembly is presently in progress. The fuel pin was also completed, and an analysis

of dummy samples filled at the same time as the fuel pins indicate that the fuel pins contain 12% argon tag gas. The remainder of the gas in each fuel section is helium.

3.2.3 Forced Convection Capsule B9B

Work continued on the design of forced convection capsule B9B. This capsule contains one fuel pin which will be defected after an exposure of approximately 25,000 MWd/Te.

Development was initiated to determine the feasibility of using a projection welding technique for attachment of the fuel pin defect nubs. To date, twenty-seven specimens have been fabricated and tested in a parameter study. Figure 3-3 shows the breaking strengths measured for the defect nubs as a function of the forging force and total electrical energy input during welding. The results indicate that leak tight welds can be made routinely and that room temperature breaking strengths comparable to the previous notched plug defect design can be obtained reproducibly ($\pm 25\%$). Further testing is continuing to fully qualify this fabrication technique.

Because of the possible sudden release of accumulated fission gas from the long, high burnup fuel pin in capsule B9B, which would result in a stream of fission gas being injected into the flowing sodium adjacent to the fuel pin, tests were performed to measure the relationship between the rate of gas injection into the sodium, sodium

velocity and the bubble diameter. A plexiglass test section was constructed modeling a single fuel pin within a concentric circular liner with a 0.093 in. flow annulus. Figure 3-4 shows a schematic of the test apparatus used. Gas was injected into the water stream through a 0.063 in. diameter hole in the wall of the liner. Both the water velocity and the gas injection rate were varied while high speed photographs were taken of the resulting bubbles. The average of the six to ten bubbles measured in each photograph was taken as the bubble size. The highest water velocity for which photographs could be taken with the currently available equipment was approximately 16 ft/sec (sodium velocity past the fuel pin for B9B is approximately 30 ft/sec). As shown in Figure 3-5, the data obtained indicates that the bubble diameter decreases with increasing water flowrate. For high water velocities and low gas injection rates, the bubble diameter is approximately 30 to 40 mils.

Following the evaluation of the present forced convection capsule de-entrainer, as reported in the 34th Quarterly Report, a new centrifugal type de-entrainer was built. A sketch of the design is shown in Figure 3-6. The de-entrainer functions in the following manner. The sodium passes from the central tube through the slots in the side which directs the sodium flow tangentially onto the surface of the vanes. The centrifugal force resulting from the fluid

Figure 3-3. Break Strength of Defect Nub Versus Forging Force and Energy Input (Projection Welding)

Figure 3-4. A Schematic of the Test Apparatus Used to Determine Bubble Diameter

Figure 3-5. Water Velocity Versus Bubble Diameter Results from Gas De-entrainer Studies

following the curvature of the vanes forces the gas to the sodium surface due to the difference in densities.

A plexiglass mockup of the top end of the FCC was constructed to perform the tests. Figure 3-7 shows a schematic of the test apparatus. For a typical test, the water and gas flowrates were set and the system was allowed to reach an equilibrium condition. The water level in the capsule was maintained at the level of the bottom of the vanes throughout all the tests. A graduated cylinder, filled with water, was inverted in the reservoir over the exit line from the FCC model. In this way any gas bubbles in the exit line (i.e., bubbles not de-entrained) were collected and measured. By measuring the time to collect a given amount of gas (corrected for pressure differences between the gas injection port and the collection cylinder), the rate of gas not de-entrained was determined. The efficiency of the de-entrainer is defined as the volume de-entrained divided by the volume injected. The results of these tests are shown in Figure 3-8.

In operation it appeared that there was always 100% separation of gas and water on the vanes of the de-entrainer but that there was also a constant entrainment of a small amount of gas in the pool below the vanes. The rate of this entrainment was constant for a given water flowrate and completely independent of the gas injection rate. This

entrainment causes the apparent decrease in the overall efficiency of the de-entrainer with a decrease in gas injection rate and can also explain the shape of the efficiency curves. There are two phenomena occurring in the pool. One is the entrainment of gas due to the turbulence in the water coming off the vanes. The other is the separation of gas and water due to the vortex created by the velocity of the water coming off the vanes. From 0 to 4 gpm the vortex has a negligible effect while the turbulence is entraining air proportional to the velocity. Between 4 and 6 gpm, the vortex is increasing in strength sufficiently to cause separation and thus the net rate of gas entrainment is reduced. Above 6 gpm the turbulence increases at a faster rate than does the strength of the vortex and the net rate of gas entrainment increases. These effects are visible in the graph of water flowrate versus gas entrainment as shown in Figure 3-9.

Tests were also conducted to examine the performance of the centrifugal de-entrainer for a large and sudden release of fission gas as might occur in capsule B9B when the high burnup pin is failed. For these tests 100 and 200 cc of gas were suddenly (4 to 8 seconds duration) injected into the water upstream of the de-entrainer model. The tests indicate the slug flow efficiency to be similar to the steady state rate (95 to 100% efficient).

Figure 3-6. A Sketch of the Centrifugal De-Entrainer

Figure 3-7. A Schematic of the Test Apparatus

Figure 3-8. Water Flow Rate Versus Efficiency of the Centrifugal De-Entrainer

Figure 3-9. Water Flowrate Versus the Rate of Gas Entrainment of the Centrifugal De-Entrainer

After completion of the initial testing of the centrifugal type de-entrainer, the currently used de-entrainer was retested using the new plastic mockup and the improved technique for measuring entrained gas. The tests were conducted similarly to those described above for the centrifugal gas de-entrainer. During the tests the water level was initially set even with the top of the vanes of the de-entrainer.

The results of these tests are shown in Figure 3-10. The current de-entrainer is 15 to 30% efficient when operated at water flowrates of 5 to 10 gpm. Previous tests using less precise methods of flow and gas measurement indicated efficiencies of only 0 to 10%. No re-entrainment of gas was measured during the tests.

Tests were also conducted to determine the performance of the present de-entrainer for slug flow conditions. Similar to the centrifugal type de-entrainer, slug flow efficiency was found to be similar to that of the steady state rate (i.e., 15 to 30% efficient at 20 to 30 ft/sec water velocities).

3.2.4 Forced Convection Capsule B10A

Forced convection capsule B10A is planned for irradiation in the core of GETR. The capsule is being designed to explore the short term effects of the failure of the cladding of one fuel pin on neighboring pins. Work continued on the

preparation of the safety analysis which is required before this capsule can be irradiated in-core. Work completed to date on establishing detailed capsule performance criteria includes GETR and FCC decay heat calculations, establishment of the experimental parameters, fixing of capsule dimensions, preliminary pump design, calculation of the natural circulation in the FCC, development of a model for analysis of reactivity transients and startup accidents, and updating of the THTD code in preparation for running steady state cases of capsule operation. Completion of the safety analysis is scheduled for October 1970.

3.2.5 Defect Development

Differential Fuel/Cladding Expansion

Destructive examination of capsule E1J-B2 was completed. This inpile experiment was conducted to demonstrate the use of differential fuel-clad interaction as a defecting mechanism. This test provided further information on the differential expansion defect mechanism as a means of defecting fuel pins, as well as providing short-term swelling kinetics of ruptured mixed-oxide fuel specimens. This specimen was irradiated for time periods of 2 minutes, 1/2 hour, 2 hours and 24 hours.

The fuel pin was fabricated with fully cold worked 316 stainless steel cladding. The initial length of the defect notch cut in the cladding was 1 inch. Figures 3-11 and 3-12 are post-irradiation photographs of the defect area.

Figure 3-10. Water Flowrate Versus Efficiency of the Currently Used De-Entrainer

Figure 3-11. Post-Irradiation Appearance OD Defect Area of E1J-B2 Fuel Pin

Figure 3-12. Cross Sectional Appearance of E1J-B2 Fuel Pin through the Defect Area

Post-irradiation diametral measurements of the cladding are shown in Figure 3-13. The fuel pin showed substantial swelling in the fabricated defect area. An average diametral increase of 0.026 in. was measured at the axial mid-plane of the defect. This corresponds to an average diametral increase of approximately 11%.

Sectioning and metallographic examination of the fuel pin were completed. Specimens were examined from the defected and nondefected regions of the pin. As can be observed (Figure 3-11), the notch was propagated (approximately 1/4 in.) at both ends.

Figure 3-14 shows the appearance of the fuel in the defected and nondefected regions of the pin. The fuel contained 25 wt/% Pu and was of stoichiometric composition. The fuel in the defect region was difficult to retain during handling. Certain areas showed a pronounced tendency to "pull-out" compared to the remaining portions of the fuel indicating greater attack by the sodium in these areas. Only minor traces of reaction product were observed at the periphery of the fuel. No columnar grain growth was observed in the fuel.

3.3 SODIUM/FUEL/CLADDING COMPATIBILITY EXPERIMENTS

3.3.1 Capsules B8C and B8D

Preliminary analysis of the microstructures of the sodium bonded fuel pins from Capsules B8C and B8D indicate gap conductivity values comparable to or less than the values measured for gas bonded pins with diametral gaps equal to the widths of the grey phase measured in the present pins. At 113,000 MWd/Te, a gap conductivity of approximately $1175 \text{ Btu/hr-ft}^2 \cdot ^\circ\text{F}$ was calculated (for 0.140 in. o.d. fuel). This compares with values of approximately $450\text{-}700 \text{ Btu/hr-ft}^2 \cdot ^\circ\text{F}$ calculated at lower burnups (0.220 in. o.d. fuel).

A fission gas release rate of 78% was measured for fuel pin B8C-1 which went to 113,000 MWd/Te. This release rate is similar to values measured for gas bonded fuel with comparable center temperatures and burnups.

3.4 FISSION PRODUCT DISTRIBUTION

3.4.1 Miniloop (Out-of-Pile Radioactive Fission Product/Sodium Loop)

Destructive examination was continued to determine the fission product distribution in a small pumped sodium loop in which hot sodium was allowed to flow by a defected mixed-oxide fuel specimen which had been

Figure 3-13. Profilometer Data for E1J-B2 Fuel Pin (O/M 2.00)

Figure 3-14. Appearance of Fuel in the Defected and Non-Defected Regions OD E1J-B2 Fuel Pin

previously irradiated to $\sim 57,000$ MWd/Te in EBR-II. Figure 3-15 shows the cross sectional appearance of the test fuel sample. As reported in the 34th Quarterly Report, the test sample showed a 3% volume increase of the fuel. This can be physically observed by the increasing width of the slot in the cladding toward the axial mid-plane of the fuel sample, and by the fuel extruded into the slot area of the cladding.

A control fuel sample and the test fuel sample were analyzed for selected fission products. Table 3-1 shows the fraction of the fission products remaining in the test fuel sample after 18 hours operation. The sodium velocity and temperature at the fuel sample were 7 ft/sec and 1145°F , respectively (loop ΔT 650°F). The ruthenium shows the greatest change. However, it remains in the metallic state in the fuel and tends to agglomerate to larger particles within the fuel. This larger change could be due to nonuniformity of distribution between the control and test sample.

A sample of the loop sodium was analyzed for fission products. Only Cs-137 was found in the sample.

Selected sections of the loop piping were sectioned and their inner surface leached with a nitric acid solution. The leached samples were analyzed for the same fission products as the test fuel sample. Table 3-2 shows the results

of the analysis. Figure 3-16 shows the area of the loop from which the samples were obtained. From the present partial results, it can be seen that the Cs-137 activity remains dissolved in the sodium and does not deposit on the tubing walls. The scatter in the Cs-137 distribution appears to be due to the varying amounts of residual sodium found in the tubing samples. For the remaining elements, the cooler inlet sections show the highest concentration of fission products.

The leached samples were also analyzed for uranium and plutonium. The ratio of uranium to plutonium found in the samples ranges from 100 to 3000. Thus, particulate fuel (which would have a U/Pu ratio of 4) is not the predominant form present.

Table 3-1
FRACTION OF SELECTED FISSION PRODUCTS
REMAINING IN MINILoop TEST FUEL SAMPLE

Fission Product	Fraction Remaining
Cs-137	0.39
Ce-144	0.79
Ru-106	0.18
Sr-90	0.87
Pm-147	0.86

Table 3-2
MEASURED DEPOSITION OF FISSION PRODUCTS
INSIDE SELECTED AREAS OF MINILoop TUBING

Sample Number	Cs-137	Ce-144	Zr-95	Ru-106	Pm-147	Sr-90
1	2.0×10^{14}	6.4×10^{12}	3.3×10^{11}	5.9×10^{12}	1.2×10^{13}	6.7×10^{13}
3a	9.7×10^{14}	8.3×10^{12}	1.1×10^{12}	1.6×10^{13}	2.8×10^{13}	1.5×10^{14}
3b	4.0×10^{14}	1.7×10^{13}	1.9×10^{12}	2.5×10^{13}	7.3×10^{13}	2.3×10^{14}
3c	1.2×10^{16}	3.1×10^{13}	4.6×10^{12}	2.6×10^{13}	1.1×10^{14}	3.6×10^{14}
5a	1.6×10^{16}	ND	ND	ND	2.7×10^{12}	3.9×10^{12}
5b	3.0×10^{15}	1.2×10^{13}	8.8×10^{11}	7.4×10^{12}	3.0×10^{12}	1.1×10^{14}
7	1.6×10^{16}	ND	ND	ND	4.8×10^{12}	9.0×10^{12}
8	4.7×10^{15}	5.0×10^{12}	4.3×10^{11}	ND	7.2×10^{12}	1.5×10^{14}
8a	1.1×10^{16}	ND	5.3×10^{11}	ND	4.0×10^{12}	6.8×10^{12}
9b	1.0×10^{16}	7.6×10^{11}	6.3×10^{10}	4.0×10^{11}	4.8×10^{12}	1.7×10^{13}

All values are in terms of atoms per square inch of inside tubing area.

REFERENCES

1. *Sodium-Cooled Reactors, Fast Ceramic Development Program, Thirty-Fourth Quarterly Report, February-April 1970, June 1970, (GEAP-10028-34).*

Figure 3-15. Post-Test Appearance of Miniloop Fuel Sample

Figure 3-16. Locations Selected for Leach Sample Analysis

4. TASK C-1 SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)

4.1 GENERAL

The purpose of Task C-1 is to establish the behavior of fuel and cladding under transient casualty and accident situations. Both analytical techniques and an experimental program are used to determine the response of fuel to severe transients. These investigations include a study of the transient performance of fuel and cladding as a function of axial restraint, density, condition (defected, high burnup), and fuel form (powder and pelleted fuel). Additional tests were conducted to determine the potential for movement of molten fuel and to observe the effect of the prototypic range of fuel and blanket lengths on movement.

4.2 HIGH BURNUP FUEL EXPERIMENTS

4.2.1 Capsule Fabrication and Assembly

Capsule C4X was completely assembled and checked for thermocouple, heater, and pressure transducer response, all of which were successful. During the final leak check of the assembly, leakage was found in the vendor-supplied pressure vessel head. The main leakage occurred through the potting compound which encapsulated the instrumentation leads. The resin epoxy was apparently improperly mixed, and was still curing (the head had previously passed a 200 psi helium leak check), resulting in continued shrinkage. The C4X pressure vessel head was partially repotted and a leak tight seal obtained (leakage $< 4.3 \times 10^{-8}$ cc/sec). The capsule heating system and instrumentation were successfully rechecked, and the capsule shipped to TREAT.

4.2.2 Treat Irradiation

An addendum to the Series 4 TREAT Hazard Analysis was submitted to TREAT requesting extension of the Series 4 tests to include the new capsule design and the additional fuel specimens C4K, C4L (powder fuel), and C4X (zero-burnup enriched UO_2 prototype specimen). At the request of TREAT, the addendum was revised to incorporate potential capsule effects resulting from any occurrence of sodium hammer, and to increase the energy release considered for maximum accident conditions.

The C4X capsule was subjected to two calibration-size transient irradiations with apparently successful operation of all instrumentation (13 thermocouples, pressure transducer, and EM flowmeter). Preliminary results indicate that the experimental capsule power factor is 30 to 40% below the predictions based on the calculated nuclear characteristics of the Series IV TREAT capsule. These cal-

culations are being reanalyzed in an attempt to resolve the differences between the predicted and experimental power factor.

4.2.3 EBR-II Irradiations

The final four TREAT specimens to be irradiated in EBR-II (specimens C4E, F, K, and L) have completed their preirradiation with an accumulated peak burnup of $\sim 60,000$ MWd/Te. These specimens have been returned to Vallecitos Nuclear Center and are presently being non-destructively examined in preparation for remote insertion into their respective TREAT capsules. Examination of post-irradiation neutrographs and gamma scans indicates that all four of the specimens are in excellent condition.

4.3 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

This test involved two three-pin capsules to investigate the transient performance of defected or sodium-bonded fuel. Each of these capsules contained two sodium-filled pins (24-inch length fuel, 15-inch length upper blanket), using both pellet and powder fuel, plus a normal gas-filled pin. The three pins in each assembly were separately encapsulated and were arranged side-by-side in a triangular array (see 31st Quarterly Report GEAP-10028-31). Test transients on both capsules C6B and C6A were performed during this report period. Recorded TREAT parameters for the transients are shown in Table 4-1.

Table 4-1

SUMMARY OF TREAT OPERATING PARAMETERS
OF CAPSULES C6A AND C6B

	<u>C6B</u>	<u>C6A</u>
TREAT transient No.	1308	1312
Integrated power, MW-sec	442	485
Peak power, MW	924	1064
Period, sec	0.088	0.083
Reactivity, %	1.71	1.78

Analysis of the test data indicated satisfactory performance. Figures 4-1 and 4-2 which show the sodium coolant temperatures (thermocouple data shown from fuel

Figure 4-1. Calculated Versus Actual Coolant Temperature TT1308 442MW—seconds

Figure 4-2. Calculated Versus Actual Coolant Temperature TT1312 485 MW-seconds

column midplane locations) during the transients, illustrate good correlation between calculated and actual test data. Based on these correlations, the volume of transient-melted fuel for each of the pins is estimated to be:

	C6B	C6A
Gas-filled pellet	≈ 54%	≈ 59%
Sodium-filled pellet	≈ 32%	≈ 42%
Sodium-filled powder	≈ 20%	≈ 22%

Both capsule assemblies have been returned to Vallecitos for destructive examination. Capsule C6B has been completely disassembled and metallographic examination of select fuel specimens is in progress. Neutron radiographs of the inner capsules are shown in Figure 4-3. None of the fuel pins failed. Numerous central voids are evident in the gas-filled pellet pin (C6B-3); 50 to 100 mils wide and of varying shapes. With the exception of a few pellet separations, there are no gross changes evident in the sodium-filled pellet pin (C6B-1) as a result of the transient irradiations. The sodium-filled powder fuel column (C6B-2)

has separated in several places and has also increased in length ≈ 1.6 inches, compressing the hold-down spring to its fully collapsed position. A narrow central void is barely visible.

Profilometer traces of the C6B fuel pins do not indicate any gross distortion (≈ 1 mil diametral maximum with a few peaks as great as 2.5 mils diametral), but pronounced ovality is evident for the sodium-filled pellet pin.

The C6B fuel pins were sectioned and samples selected for detailed examination. Transverse sections from the top, middle, and bottom (T/C locations) from each of the pins are shown in Figures 4-4, 4-5, and 4-6.

Capsule C6A has been partially disassembled. Neutron radiographs of the inner capsules are shown in Figure 4-7. Fuel pin failure is not evident and the results appear similar to C6B. The sodium-filled pellet fuel column (C6A-1) increased axially ≈ 1 inch due to pellet separations and a small central void is evident. The sodium-filled powder fuel column (C6A-2) increased axially ≈ 2 inches and central void of ≈ 60 mils in diameter was noted. A continuous central void (≈ 100 mils wide) was noted in the upper half of the gas-filled pellet fuel pin (C6A-3).

Figure 4-3. Neutron Radiograph of Capsule C6B Following TREAT Transient (TT1308) Irradiation

10 X

GROUND

A504-02

TOP

10 X

GROUND

A505-01

MIDDLE

10X

GROUND

A500-01

BOTTOM

Figure 4-4. Transverse Sections From C6B-1, Sodium-Filled Pellet Fuel Pin

10X

GROUND

A511-01

TOP

10X

GROUND

A509-01

MIDDLE

10X

GROUND

A507-01

BOTTOM

Figure 4-5. Transverse Sections From C6B-2, Sodium-Filled Powder Fuel Pin

10X

GROUND

A514-01

TOP

10X

GROUND

A515-01

MIDDLE

10X

GROUND

A516-01

BOTTOM

Figure 4-6. Transverse Sections From C6B-3, Gas-Filled Pellet Fuel Pin

Figure 4-7. Neutron Radiograph of Capsule C6A Following TREAT Transient (TT1312) Irradiation

5. TASK C-2—SAFETY EXPERIMENTATION (LOSS OF FLOW)

5.1 GENERAL

A major safety-related area which is of concern to both FFTF and demonstration plant programs is the possibility of rapid reduction or loss of coolant to localized portions of the core. Such coolant starvation might be a result of local blockages, fuel distortion, debris, or failure of the pumps. The questions of fuel response to such starvation, potential for propagation to neighboring pins, and the time characteristics of signals such as temperature, flow, etc., are a primary concern in the design of the fuel bundles and instrumentation system. In addition, answers to these questions must be available to provide reliable input to accident analyses for licensing purposes.

Various theoretical studies have been conducted on the effects of loss of flow, ranging from estimates of core voiding reactivity to the local heat transfer and dynamics of single channel boiling. Very little experimental data, however, are available on the local effects of transient undercooling and boiling in a confined channel. A need exists to perform experiments in-pile, with oxide fuel and flowing coolant, to verify models based on out-of-pile experiments and to assess secondary events produced by sodium re-entry and subsequent contact with hot or molten fuel. The purpose of this task is to develop and proceed on a program of such tests, utilizing the Forced Convection Capsule and/or the Mark II Loop as the basic vehicle. Task C-2 was formally initiated in the latter half of 1969.

5.2 MOCKUP 1A

There is a continuing limitation on the usefulness and interpretability of out-of-pile tests such as Mockup 1A due to the loss of continuity or possible arcing of heaters which occur relatively early in the expulsion-reentry sequence. The Mockup 1A heaters have been removed from the test rig and are being examined to determine the distortions which occur on continued operation during loss-of-flow tests. The examination of the heaters is being conducted with a view to determine (1) possible changes in specifications which could increase the time-to-failure under these extreme conditions, and (2) whether the mode of heater failure itself contributed to the observed pressure and flow effects. The results of the examination to date confirm the fact that the heaters failed internally and that the flow and pressure oscillations reported in GEAP-10028-34 were valid observations of expulsion and reentry effects. Further x-ray and destructive examinations are in progress. The results of these investigations will be applied to possible improved specifications for heaters for subsequent out-of-pile transient tests.

5.3 C10X

This experiment is designed to establish the pressures and energies associated with molten fuel-sodium interactions. Fabrication of the C10X capsule is under way. Machining has been completed on most major components. A proof test of the burst diaphragm operation is being prepared. This burst diaphragm is used to initiate the temperature transient and its operation must be factored into the transient thermal analysis. The sodium fill procedure is also undergoing a proof test which will minimize the possibility of trouble in the final fill. The components for this test are being prepared and the fill system readied.

The fuel pin has been fabricated and is ready for final assembly.

A detailed thermal analysis, describing the transient response of the fuel and capsule, has been completed. The transient behavior of the burst diaphragm was assumed and will be confirmed by the previously mentioned test. Pertinent design and operation information is being prepared for insertion into the data package. The safety analysis for this data package has been completed.

In order to analyse the voiding behavior, the VOID code was modified to handle the unique geometry and zero-flow conditions of capsule C10X. Some meaningful results have been obtained which show that even for low superheats ($< 50^{\circ}\text{F}$) the vapor pressure is sufficient to cause significant displacement of the upper sodium slug. The combination of low sodium inertia and plenum gas pressurization results in an oscillatory motion of the expelled sodium slug shortly after boiling has been initiated. These results are in agreement with the experimental work of Spiller, Perschke, and Grass.¹

5.4 C10A/C10B

Test parameters and capsule requirements for the first loss-of-flow test (C10A) have been formulated and preliminary capsule specifications generated. Work has been initiated on a conceptual design of the in-pile test facility. This concept will permit an objective evaluation of the capabilities of the facility to meet specific test requirements.

Work continued on the checkout and updating of the THTE program. The major options of the program worked on to date and their status are listed in Table 5-1.

The Dynamic Memory Allocation feature of the program allows adjusting the computer memory requirements of the program to the particular problem being executed. The direct cylindrical geometry feature allows

direct input of cylindrical geometry and eliminates the need for a precomputation of irregular geometry for cylindrical nodes. The remote terminal summary option provides a data processing error summary and edited temperature output at the remote terminal to minimize job turnaround time. The data translating program (DTE) will convert THTD input data decks to the THTE data format allowing existing THTD models to be executed on the THTE program with a minimum of hand coding.

The capability of the THTE program to perform multi-channel series-parallel hydraulics and heat transfer calculations will allow a much more rigorous thermal analysis for the C10A/C10B experiments than is currently possible using THTD. These capabilities will allow a detailed analysis, including the prediction of the radial temperature distribution up to the time of voiding, to be carried out for the test section proper.

5.5 PLANNING AND DESIGN

A thermal-hydraulic study of the ANL Mark II loop is in progress. This study is intended to determine the feasibility of using the loop to perform flow blockage and flow coastdown experiments in TREAT. The basis for the analysis was the unmodified Mark II loop geometry with a test section selected to be as representative as possible of a reactor system. Preliminary results indicate that TREAT could be expected to provide approximately 20 seconds of steady-state power under conditions required for loss-of-flow testing. This relatively short operating period will limit the usefulness of this facility when loss-of-flow tests with low flow decay rates are required (e.g., loss-of-pump accident) or if significant cooling is found to occur following initial voiding.

Table 5-1
THTE CODING AND CHECKOUT STATUS

<u>Option/Feature</u>	<u>Coding</u>	<u>Checkout</u>
1. Dynamic memory allocation	95% completed	95% completed
2. Direct cylindrical geometry input	completed	completed
3. Binary tape usage (for edited temperature summary)	completed	completed
4. Remote summary	90% completed	90% completed
5. Isothermal phase change	completed	40% completed
6. Computational error correction (errors discovered in THTD and corrected)	completed	completed
7. THTD to THTE data translator	70% completed	50% completed

REFERENCES

1. Von Spiller, K. H., Perschke, D., and Grass, G., *Messungen der Restfilmdicke bei der Einzelblasenejektion von flüssigem Natrium in einem Rohr*, 1969, (ATKE 14-21).

6. TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

6.1 GENERAL

The purpose of the task is to establish the physical, mechanical, and thermal properties of ceramic fuels and fuel elements. These properties are obtained by both in- and ex-reactor testing. These properties are then utilized to model or predict the behavior of fuel elements under steady-state or casualty conditions. The reactions of fuel and fission products with cladding are investigated using an electron probe microanalyzer. Methods of limiting the reaction are being explored. The migration of impurities and fission products is examined in irradiated samples.

6.2 MATERIAL PROPERTY STUDIES

6.2.1 Mechanical Properties

The steady-state compressive creep testing of mixed-oxide samples continued with samples incorporating the variables of stoichiometry, density, and composition. A semi-empirical expression developed previously¹ described the behavior of 93% TD (U-20 wt % Pu)O_{1.95} and was similar in form to the equation derived for UO_{2.00}.

Studies of the effect of density on creep behavior showed that creep rates increased with decreasing density. This behavior is shown in Figure 6-1. The creep rates at initial stresses of 1000 and 6000 psi are plotted for samples with densities ranging from 89 to 94% TD. The similarity of the two curves illustrates that density variations have similar effects at both high and low stresses. All data were obtained on samples of (U-20 wt % Pu)O_{1.95}.

The testing of mixed-oxide samples with 20, 30, and 100% plutonia showed that composition significantly affects steady-state creep rates. Increasing the plutonium content from 20 to 100 wt % increased the creep rate at both the 1000 and 6000 psi stress levels. This is shown in Figure 6-2. The creep rate of UO_{2.00} at these stresses is included for comparison.² These data were obtained in identical test furnace atmospheres which enabled the O/Pu ratios to be maintained at about 1.78 in all samples. It is of interest to note that the creep rate nearly doubles as the plutonia content is increased from 20 to 30 wt %.

Based upon the data thus far obtained, the following general statements may be made about the creep strength of mixed-oxide pellets prepared from coprecipitated powders. For identical test conditions the steady-state creep rate increases with:

1. An increase in the oxygen-to-metal ratio from the hypostoichiometric to the stoichiometric state. This is illustrated in Figure 6-3.*

2. An increase in plutonia content.
3. A decrease in the density of the sample.

6.2.2 Thermal Conductivity Studies

Thermal conductivity measurements of mixed oxides continued. A radial heat flow technique is being used that allows temperatures to be varied from 800 to 2300°C. The results of tests performed on high density (97.6% TD) UO_{2.00} are presented in Figure 6-4. These tests were conducted to calibrate the measuring technique. A comparison of the previously reported data by Asamoto, et al.,³ indicates that these data are within the estimated experimental error of $\pm 10\%$.

Figure 6-5 illustrates the effect of stoichiometry on mixed oxide thermal conductivity. The 91.7% TD sample of (U-20 wt % Pu)O_{2-x} was prepared from physically mixed powders. The stoichiometry was varied *in situ* by changing the water content of the test furnace atmosphere. The oxygen-to-metal ratios were verified by measuring weight changes in the test pellets. At 900°C the conductivity of pellets with an O/M ratio of 2.00 is about 30% greater than those with an O/M ratio of 1.94. The effect of stoichiometry decreases with increasing temperature and when the data are extrapolated, stoichiometry is found to have a negligible effect above 1500 to 1600°C. Thermal diffusivity data reported by Gibby⁴ for (U-25 wt % Pu)O_{2-x} coprecipitated samples are shown in Figure 6-5 for comparison. The difference between the two groups of data is probably the result of different experimental techniques (radial heat flow versus thermal diffusivity) rather than differences in the method of sample preparation (physically mixed samples in contrast to coprecipitated powders).

Figure 6-6 compares the thermal conduction of (U-20 wt % Pu)O_{2-x} samples prepared from coprecipitated and physically mixed powders. The slight difference is attributed primarily to the density differences between the samples. The data show that the thermal conductivity curve may have reached a minimum at 2000 to 2200°C and may be increasing slightly at higher temperatures. Additional data are being obtained at temperatures above 2300°C.

6.3 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

6.3.1 Capsule Series E1J (Effects of Stoichiometry on Integral Thermal Conductivity of Restructured, Melted Fuel)

All nine capsules in this series embodying triplication of tests at three stoichiometries have been examined after irradiation under comparable fuel pin power conditions

* This is similar to Figure 6-1 of the 34th Quarterly Report¹ except that test temperatures indicated in the latter figure were in error and have been corrected in Figure 6-3.

Figure 6-1. The Effect of Density on the Creep Rate of (U-20 w/o Pu) $O_{1.95}$ at 1550°C

Figure 6-2. The Effect of Plutonium Content on the Creep Rate of Mixed Oxides with a Constant O/Pu Ratio

Figure 6-3. Variations in Steady State Creep Rate of 94% T.D. (U-20 wt% Pu)O_{2-x} with Stoichiometry. This is similar to Figure 6-1 of Ref. 1 Except that Test Temperatures Have Been Corrected

Figure 6-4. Thermal Conductivity of 97.6% T.D. Stoichiometric Uranium Dioxide

Figure 6-5. Effect of Stoichiometry on Thermal Conductivity of (U-20 wt% Pu)O_{2-x}

Figure 6-6. Thermal Conductivity of Coprecipitated and Physically Mixed Pellets

which resulted in fuel restructuring and subsequent melting. Analysis of the microstructural sections is being performed and will be completed during the third quarter of 1970. Fuel specimens of each of the three stoichiometries (1.94, 1.97, and 2.00 O/M ratios) have been prepared for electron microprobe analysis to study the plutonium concentration changes across the radius of melting. The microprobe analysis of three samples has been completed and is described in Section 6.4.2.

6.3.2 Capsule E2R (Rate of Central Hole Formation at Startup)

The E2R capsule irradiation test is intended to experimentally determine the rate of center void formation in a pin containing 88 and 94% TD (U-25 wt % Pu)O_{1.97} pellets at fuel temperatures approximating those predicted for startup of an LMFBR. The test will be conducted in the GETR trail cable facility to allow brief, carefully positioned capsule insertions and to provide accurately controlled capsule operating conditions. The capsule will be irradiated at a peak fuel center temperature of 2500°C for a series of exposures lasting 10 minutes, 30 minutes, 1 hour, 5 hours, 24 hours, and 1 week. After each irradiation period the capsule will be removed from the trail cable facility and neutrographed to determine center hole existence or size. Capsule assembly is scheduled to be completed in early September 1970.

The experimental neutrography effort related to the E2R capsule program continued. Neutrographs were prepared using epithermal neutrons only, with cadmium shields to exclude thermal neutrons. This improved the definition of the internal structure of pellets fully enriched in U 235, as used in Task F pins irradiated in EBR-II; pellet diameters and cladding features were less distinct using this technique, however.

Photographic enhancement, preparing a negative on Type SCR film from an original negative on Type T film, was found to significantly improve contrast. This method provides a refined image without the prolonged exposure normally associated with the use of "slower" Type SCR film for neutrography.

6.4 FISSION PRODUCT--FUEL-CLADDING REACTIONS

6.4.1 Examination of Pins F12C and F12D

These pins had been irradiated in the GETR (thermal flux) to a burnup of about 135,000 MWd/Te; fuel fabrication parameters, irradiation test conditions, and fuel and cladding metallography have been reported previously.¹ Pin F12C contained vi-packed powder fuel while the F12D fuel was solid pellets. Both capsules operated with cladding i.d. temperatures $\geq 1300^\circ\text{F}$. The post-irradiation microstructures of both pins were similar,

with considerable reaction evident at the fuel-cladding interface all around the pellet circumference. The reaction zones of both pins (~ 0.004 inch of cladding reacted) were examined by electron probe microanalysis. Figure 6-7 shows a region from pin F12C which was examined at high magnification. Specimen current images of the fuel and cladding in this region are shown in Figure 6-8. The positive specimen current images indicate intergranular penetration of the cladding by elements with a lower atomic number, as shown in Figure 6-8(b) and 6-8(c). Limited reaction of the cladding or reaction phase with the fuel is indicated by the negative specimen current image, Figure 6-8(a). More specific identification of the elements in the reaction zone is shown in Figure 6-9 by the X-ray pulse images for iron, chromium, and nickel. The reaction phase is primarily nickel, with chromium concentrated more than the other elements at the edge of the reaction layer, nearest the fuel. Figure 6-10 shows that cesium, silicon, tellurium, and molybdenum are minor components of the metallographically observed reaction phase.

Another region of the reaction phase was examined, with the results presented in Figures 6-11 and 6-12. As in the initial location, this region contains molybdenum, cesium, and silicon, with the cladding elements concentrated at the fuel edge. In this region, the iron and nickel seem to be present in higher concentrations than chromium.

The examination of powder pin F12D revealed a 1-mil-thick reaction layer at the fuel-cladding interface and an extension of cladding into the fuel as shown in Figures 6-13 and 6-14. Uranium and plutonium are not present in this extension and the fuel is displaced. More iron and chromium than nickel appear to have migrated into the fuel with the chromium segregated at the fuel cladding interface.

The cladding reaction and grain boundary attack is shown in Figure 6-15 and at higher magnification in Figure 6-16. The negative specimen current image in Figure 6-15 shows that grain boundaries near the reaction phase are filled with an element(s) that has significantly different atomic number(s) from the cladding, while the composition of grain boundaries about one grain farther away from the reaction zone is indistinguishable from the matrix. The attacked grain boundaries near the reaction phase are depleted in iron and nickel as shown in Figure 6-15. Within the cladding but away from the reaction zone, the grain boundaries have alloy contents equivalent to the matrix. Figure 6-16 shows that silicon and

cesium are concentrated in the grain boundaries of the attacked region next to the fuel but are not present at detectable concentrations farther inward.

6.4.2 Examination of Pins E1J-2, E1J-6, E1J-9

A group of three samples irradiated in the E1J capsule series at power levels to cause fuel center melting were examined with the electron microprobe to determine radial variations in plutonium content that correspond to distinct microstructural features, particularly the limit of melting. These samples have been described previously and consisted of specimens with O/M ratios of 1.94 (E1J-2), 1.97 (E1J-6), and 2.00 (E1J-9). Two radial concentration profiles were determined across each fuel section and plots of the results are shown in Figures 6-17, 6-18, and 6-19. The Pu/(U+Pu) ratio was established across the entire radius of E1J-2 (Figure 6-17). No significant changes in plutonium concentration were noted in the cooler regions of the fuel. An increase in plutonium content was found near the center of the fuel beginning in the high-density structure and rising in the porous structure within the center hole. No notable increase in plutonium was noted at the microstructural boundary assumed to be the radius of melting. The initiation of fine porosity in the high-density fuel is normally considered the molten boundary and is shown as radius (a) in Figures 6-17, 6-18, and 6-19. Significant, but not maximum, plutonium concentrations were observed near the boundary of high-density fuel and the porous region at the fuel center. Maximum plutonium concentrations were recorded within this porous fuel at the center. Similar observations of plutonium concentration apply to the sample with 1.97 O/M ratio fuel, Figure 6-18.

The radial plutonium concentration within pin E1J-9 with an O/M ratio of 2.00 did not vary significantly, as shown in Figure 6-19. No variation is evident at the assumed molten boundary radius.

Plutonium concentration traces were taken across single grains in the porous centers of pins E1J-2 and E1J-9 with O/M ratios of 1.94 and 2.00, respectively. The results (Figure 6-20) correspond to earlier findings, with the stoichiometric compound showing uniform chemistry and the hypostoichiometric sample showing variations in plutonium that may be the result of coring, occurring during solidification from the melt.

Similar microprobe studies will be conducted on another set of three E1J samples to provide further information on these microstructural phenomena.

AS POLISHED

500X

Figure 6-7. Montage of Fuel-Cladding Interface of pin F12C. Typical of Area Examined with the Shielded Electron Microprobe. Cladding Temperature was Calculated to be $\geq 1300^{\circ}\text{F}$.

250X

250X

A. NEGATIVE SPECIMEN CURRENT IMAGE SHOWING MIXED OXIDE FUEL.

B. POSITIVE SPECIMEN CURRENT IMAGE OF CLADDING AND REACTION PHASE.

500X

FUEL

REACTION ZONE

CLADDING

C. POSITIVE SPECIMEN CURRENT IMAGE AS ABOVE RIGHT AT HIGHER MAGNIFICATION.

Figure 6-8. Specimen Current Images Showing the Distribution of the Elements by Relative Atomic Number for the Fuel, Cladding and Reaction Zone in Fuel Pin F12C

FUEL | REACTION ZONE | CLADDING

A. Fe-K α X-RAY PULSE IMAGE

B. Cr-K α X-RAY PULSE IMAGE

C. Ni-K α X-RAY PULSE IMAGE

500X

Figure 6-9. X-Ray Images Showing Distribution of Fe, Cr and Ni in Enlarged Area of F12C Shown in Figure 6-8 (c). Intensity is Directly Related to Concentration. Low Intensity Areas Represent Background Signal

FUEL | REACTION ZONE | CLADDING

A. Cs-K α X-RAY PULSE IMAGE

B. Si-K α X-RAY PULSE IMAGE

C. Te-L α X-RAY PULSE IMAGE

D. Mo-K α X-RAY PULSE IMAGE

500X

Figure 6-10. X-Ray Images Showing the Distribution of Cs, Si, Te and Mo in the Area of F12C Shown in Figures 6-8 (c) and 6-9. Intensity is Directly Related to Concentration. Low-Intensity Areas Represent Background Signal.

Figure 6-11. Comparison of Specimen Current Image with Fe, Cr and Ni X-Ray Images in a Fuel-Clad Reactor Sample F12C. Width of Reaction Zone is About 100 μ . Intensity is Directly Related to Concentration. Low-Intensity Areas Represent Background Signal.

CLADDING | REACTION ZONE | FUEL

A. Mo- K_{α} X-RAY PULSE IMAGE

B. Cs- K_{α} X-RAY PULSE IMAGE

500X

C. Si- K_{α} X-RAY PULSE IMAGE

Figure 6-12. X-Ray Images Showing the Distribution of Mo, Cs and Si in the Area of F12C Shown in Figure 6-11. Intensity is Directly Related to Concentration. Low-Intensity Areas Represent Background Signal.

POSITIVE SPECIMEN CURRENT IMAGE EMPHASIZING CLADDING AND LOW-ATOMIC-NUMBER ELEMENTS (250X)

NEGATIVE SPECIMEN CURRENT IMAGE OF HIGH-ATOMIC NUMBER ELEMENTS, FUEL. (250X)

Figure 6-13. Comparison of Positive and Negative Specimen Current Images With Optical Photographs of Crack Area and Cladding "Finger" of Powder Pin F12D.

A. U-M α X-RAY PULSE IMAGE

B. Fe-K α X-RAY PULSE IMAGE

C. Cr-K α X-RAY PULSE IMAGE

D. Ni-K α X-RAY PULSE IMAGE

250X

Figure 6-14. X-Ray Images Showing the Distribution of U, Fe, Cr and Ni in Area of Powder Pin F12D Shown in Figure 6-13. Intensity is directly related to Concentration. Low-Intensity Areas Represent Background Signal.

A. PHOTOMICROGRAPH OF AREA SHOWN
IN SPECIMEN CURRENT IMAGE

B. NEGATIVE SPECIMEN CURRENT IMAGE

C. Ni-K α X-RAY PULSE IMAGE

D. Fe-K α X-RAY PULSE IMAGE

500X

Figure 6-15. Specimen Current and X-Ray Images Showing the Depletion of Alloy Elements in the Grain Boundaries of the Cladding Shown in Figure 6-13 of Powder Pin F12D. Surrounding Regions are shown in Figure 6-14. Intensity is Directly Related to Concentration. Low-Intensity Areas Represent Background Signal.

FUEL
 REACTION ZONE
 CLADDING

A. POSITIVE SPECIMEN CURRENT IMAGE (250X)

B. Fe-K α X-RAY IMAGE WITH Cs READOUT SUPERIMPOSED. STRAIGHT HORIZONTAL LINE IS LOCATION OF TRACE AND BACKGROUND IS THE LOWER LINE. INTENSITY OF Cs INCREASES IN GRAIN BOUNDARY DELINEATED BY DECREASE IN Fe INTENSITY.

C. Si-K α X-RAY PULSE IMAGE (500X)

D. Cs-K α X-RAY PULSE IMAGE (500X)

Figure 6-16. Specimen Current and X-Ray Images Showing Concentration of Cs and Si in the Grain Boundaries in an Area of Fuel-Clad Reaction and Intergranular Attack in the Powder Pin F12D. Intensity is Directly Related to Concentration. Low-Intensity Areas Represent Background Signal.

Figure 6-17. Radial Plutonium Concentration Across Specimen E1J-2, (1.94 O/M) Region (a) Represents The Microstructural Interface Between Solid Fuel and Origin of Porosity, Usually Considered the Melting Limit; Region (b) Represents the Radius of Solid, Intact Fuel. The Two Radial Illustrations Represent a Single Diametral View.

Figure 6-18. Radial Plutonium Concentration across Sample E1J-6 with O/M = 1.97. Region (a) represents the microstructural interface between solid fuel and the origin of porosity, usually considered the melting limit; Region (b) represents the radius of solid intact fuel. The Two Radial Figures Represent a Single Diametral View

Figure 6-19. Radial Plutonium Concentration across E1J-9 with O/M = 2.00. Region (a) represents the microstructural interface between solid fuel and the origin of porosity usually considered the melting limit; Region (b) represents the radius of solid, intact fuel. The two Radial Sections are from a Single Diametral View

a. Pu CONCENTRATION IN E1J-2 (1.94 O/M)

b. Pu CONCENTRATION IN E1J-9 (2.00 O/M)

Figure 6-20. Pu Concentration Across Grains in the Once Molten Area of E1J Pins. Note Variations in E1J-2, in Contrast to Uniformity of Stoichiometric Pin E1J-9.

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4. Gibby, R. L., *The Effect of Oxygen Stoichiometry on the Thermal Diffusivity of $U_{0.75}Pu_{0.25}O_{2-x}$* , January 1969, (BNWL-927).

7. TASK F—FAST FLUX TESTS AND SUPPORT

7.1 GENERAL

The purpose of Task F is to apply the more promising results of developments in basic fuel technology to the design, fabrication, and integral testing of fuel elements in fast fluxes. Important fuel performance characteristics, such as fission gas release, fuel-cladding interactions, and fuel thermal performance, are determined under conditions simulating, as closely as possible, those expected in a fast ceramic reactor. Testing of fuel pins in a fast flux, in principle, will permit subsequent testing of fuel bundles under a fair approximation of prototypical conditions.

The test parameters for Task F are selected by a combination of design analyses which relies on basic properties data and on the results of pertinent experiments (such as those in Task E and Task L). More recently, model analysis (Task P) has been applied quantitatively to interpretation of the results, and to aid in the selection of ranges of parameters for further tests. The results of prior fast-flux irradiation tests are interpreted as quantitatively as possible to guide the design of future tests. The selection of test parameters is also influenced and guided by the work under Task B on defected fuel testing; Task C, which focuses on the preferred range of parameters for safety considerations of LMFBR designs; Task G, which is concerned with subassembly design and performance; and Task P, involving fuel element design and analysis. Demonstration of high fuel burnup (50,000 to 100,000 MWd/Te) is required over a range of power levels (4 to 24 kW/ft). Fuel parameters that are under study in a fast flux environment include density, configuration, fabrication process history, stoichiometry, diametral gap, axial gap, and external restraint. Fast flux tests are performed with both encapsulated and unencapsulated pins.

Selected irradiation tests in thermal test reactors are being performed in support of the fast flux irradiations in EBR-II. The areas being explored are high-burnup

irradiations at high-power levels which contribute to the understanding of the performance of mixed-oxide fuel under irradiation and which cannot currently be performed in the EBR-II.

These tests include high-risk tests, instrumented capsules, capsules operated at high-power with fuel temperatures at or above melting, and capsules designed to determine mixed-oxide-fuel swelling by measuring fuel pin diameters without the complications of cladding swelling.

A summary of the fast flux irradiation tests in the EBR-II is given in Table 2-4. Calculated values for total fluence, greater than 0.1 MeV, and fuel burnup values in MWd/Te and atom percent (taking into account fuel depletion) are incorporated into the summary table. The fluence, power, and burnup values given in Table 2-4 were calculated using the fission rates provided in the latest revision of the "Experimenters' Guide for Irradiations in EBR-II" as a basis.

Capsule irradiation tests performed in thermal spectrum test reactors which are being performed under Task F and capsule tests which were transferred from Task E to Task F, are listed in Table 2-5.

7.2 GROUP F2 (MEDIUM POWER SCOPING IRRADIATIONS)

The irradiation of the four Group F2 capsules in EBR-II subassembly X050 was terminated at the end of run 42. The average burnup and peak total fluence received by the highest burnup capsule, F2V, was calculated* to be 105,050 MWd/Te (11.75 at. %) and 1.03×10^{23} nvt. The peak burnup in pin F2V was 118,200 MWd/Te (13.2 at. %). These capsules have attained the highest fuel burnup and fast neutron fluence of any mixed-oxide fuel irradiation in the EBR-II under the FCR Program.

* All burnup and fluence values reported were calculated based upon fission rates in the EBR-II Experimenters' Guide (latest revision) unless noted otherwise. The burnup values reported include correction for fuel depletion.

After a decay period the capsules were removed from the EBR-II storage basket, the X050 subassembly was disassembled, and an initial examination of the capsules was performed at the EBR-II site.

Each fuel pin contained approximately 14.2 inches of coprecipitated mixed-oxide fuel. The fuel composition was nominally 20% plutonium and 80% uranium enriched fully in U-235. The cladding was welded and drawn Type-316 or -347 stainless steel, 0.250-inch in diameter by 0.015-inch wall. Fabrication and irradiation parameters for each fuel pin are listed in Table 7-1.

The power history of the Group F2 pins is shown in Figure 7-1. During the time the pins were in the reactor three significant changes occurred which affected the power of the fuel pins.

1. After 393 MWd of reactor operation the XG05 subassembly was rotated 180 degrees to place it in its

proper orientation. This resulted in an increase in power of fuel pins F2D, F2G, and F2V. The operating power of pin F2X decreased.

2. A stainless-steel reflector was installed in the reactor which flattened the radial flux profile. This resulted in essentially no change in the power of pins F2D and F2G. The power of pins F2V and F2X decreased and increased, respectively, as a result of the position of these pins within the subassembly.
3. The reactor power was increased from 45 to 50 MW. Coincidentally, the pins were placed in different locations in a new subassembly. A small net power decrease resulted.

The general decrease in operating power which occurred over the lifetime of the irradiation resulted from burnup depletion of the fuel.

Table 7-1
GROUP F2 DESIGN AND IRRADIATION PARAMETERS

<u>Pin</u>	<u>F2D(a)</u>	<u>F2G(a)</u>	<u>F2V</u>	<u>F2X</u>
Fuel Type	Pellet	Pellet	Pellet	Powder
Fuel Density, %	96.6	95.5	89.8	83.9
Fuel-Cladding Gap, mils	1.8-4.1	0.9-6.0	2.8-4.5	---
Fuel Smeared Density, % TD	94.4	92.9	86.7	83.9
Fuel O/M Ratio	2.000	1.975	1.988	1.992
Cladding Type	347	347	316	316
Peak Linear Power, kW/ft(b)	16.5	16.1	15.7	15.3
Peak Fuel Burnup, at. %(b)	12.8	12.7	13.1	11.9
Average Fuel Burnup, at. %(b)	11.5	11.4	11.8	10.7
Peak Cladding Fluence, > 0.1 MeV ($\times 10^{23}$)	0.92	0.92	0.95	0.85

(a) Fuel pins F2D and F2G contained axially restrained fuel. Pin F2D failed at some burnup above 7 at. %.

(b) All burnup and fluence values reported were calculated based upon fission rates in the EBR-II Experimenters' Guide (latest revision). The burnup values reported include correction for fuel depletion. The peak linear power reported occurred at the beginning of the irradiation of the pins.

Figure 7-1. Power Histories for Group F2 Capsules F2D, F2G, F2V, and F2X

Neutronradiographs of the capsules indicated that one pin, F2D, had failed. This pin contained solid-pellet fuel, which was clad in Type-347 stainless steel. The fuel density and oxygen-to-metal ratio were 96.6% and 2.00, respectively. The diametral clearance between the fuel and cladding initially ranged from 1.8 to 4.1 mils, resulting in an over all fuel smeared density of 94.4%. Neutronradiographs of the F2D pin at a burnup of approximately 70,000 MWd/Te indicated that the fuel pin was intact. Pin F2D contained axially restrained fuel.

Diameter measurements were taken at two orientations on the 0.375-in.-diameter Type-304 stainless steel capsule tubes at the EBR-II site. The maximum diameter increases measured on the capsules are listed in Table 7-2. Plots of the measurements are presented in Figures 7-2 through 7-5. The measured diameter increases agree well with calculated values obtained using the empirical equation for stainless steel swelling proposed by General Electric.¹

7.3 LOW POWER FUEL SWELLING IRRADIATIONS

7.3.1 Group F0 (Low Power Swelling of High Density Fuel)

The Group F0 capsules in EBR-II subassemblies XG03 and XG04 reached average burnups of 60,750 MWd/Te (6.79 at. %) and 67,300 MWd/Te (7.52 at. %), respectively, at the end of run 44A. The target burnup for the fuel in the XG03 and XG04 subassemblies is, respectively, 80,000 MWd/Te (8.95 at. %) and 100,000 MWd/Te (11.18 at. %). It is anticipated that the normal operating power of the EBR-II will be raised from

50 to 62.5 MW prior to the Group F0 capsules reaching these burnup targets. In this event, the capsules will be reconstituted into new subassembly hardware and placed into new grid positions for operation at their design power levels.

Revised fuel temperature calculations for pin F0E* based on the power history published in GEAP-10028-34 were completed. The new calculations indicate a peak fuel temperature of 1400°C was attained at a linear power of 8.3 kW/ft based on a gap coefficient of 1500 Btu/h-ft²-°F.

7.3.2 Group F6 (Fuel-Cladding Interactions Under Accelerated Swelling)

Results of the destructive examination of the Group F6 fuel pins were reported previously^{2,3} with the exception of the metallographic examination of the failed fuel pin, F0M, and fuel burnup and fission gas release fractions.

A transverse metallographic specimen taken from the peak flux region of fuel pin F0M was examined. The fuel structure was similar to the unfailed Group F6 fuel pins except in the region adjacent to the cladding, where a 5-to-10-mil-thick zone of fuel-sodium reaction phase had formed. No fuel-fission product reaction with the I-800 cladding was observed and there was no evidence of abnormal fuel or cladding temperatures. Secondary cracks initiating from the inside surface of this cladding were found approximately 120 degrees from the major cladding failure. The cracks were intergranular in nature. Photomicrographs of the fuel pin section examined are shown in Figure 7-6.

* The examination of pin F0E was reported in GEAP-10028-34.

Table 7-2
MAXIMUM DIAMETER CHANGES MEASURED ON GROUP F2 CAPSULE TUBES

<u>Capsule</u>	<u>Maximum ΔD (mils)</u>	<u>Maximum $\Delta D/D$ (%)</u>
F2D(a)	9.6	2.6
F2G	10.1	2.7
F2V	7.8	2.1
F2X	9.2	2.5

(a) Fuel pin F2D failed

Figure 7-2. Diameter Measurements from F2D Capsule Tube (F2D Fuel Pin Failed)

Figure 7-3. Diameter Measurements from F2G Capsule Tube

Figure 7-4. Diameter Measurements from F2V Capsule Tube

Figure 7-5. Diameter Measurements from F2X Capsule Tube

METALLOGRAPHY FROM FAILED PIN FOM
60,000 MWd/Te, INCOLOY 800 CLAD, 1.98 O/M FUEL

Figure 7-6. Photomicrographs of Transverse Section From Failed Group F6 Fuel Pin FOM

The possibility that sodium may have entered the FOM fuel pin through a defective end plug weld was investigated by metallographically examining the end plug weld by progressive grinding. Some isolated weld porosity was found but no cracks or pin holes were discovered.

It was not possible to determine where the fuel cladding failed initially or the cause of the failure. The low fission gas release indicates that the pin may have failed early in life. The multiple cracks observed in the cladding over the fueled region of the pin are thought to be the result of the formation of a low-density fuel-sodium reaction product which would effectively increase the volume of the fuel and result in excessive cladding strain.

The fission gas fractions determined for the Group F6 fuel pins were reported previously² based on analytical burnup results.

The fractional fission gas releases are listed in Table 7-3 based on both calculated and experimental burnup values and on gas chromatograph analysis of fission gas samples. The gas release fractions for the unfailed Group F6 pins are similar to the Group F2 pins examined at approximately the same burnup but which operated at considerably higher fuel temperatures. Fission gas release from the failed pin, FOM, was 67 to 78% of that determined for the unfailed Group F6 fuel pins.

7.3.3 Group F8A (Effect of Fuel Density on Fuel Swelling Rate)

The post-irradiation examination of the Group F8A pins irradiated in EBR-II subassembly X019 was completed.

Final fuel burnup results and fission gas release fractions are listed in Table 7-4. All gas release fractions are based on fuel burnups and gas chromatograph analysis of fission gas samples.

The interim examination of the 9 Group F8A capsules removed from EBR-II subassembly X020 was completed at the EBR-II site. The maximum diameter changes measured on the capsules ranged from 0.26% to 0.79%. Based on the results of this examination, the capsules were judged acceptable for continued irradiation and were reassembled into EBR-II subassembly X081 for irradiation toward a target burnup of 100,000 MWd/Te (11.25 at. %). The capsules were reinserted in the EBR-II core at the start of run 43.

7.3.4 Group F8B (Effect of Void Deployment on Fuel Cladding Interactions)

Irradiation of the 19 Group F8B capsules in EBR-II subassembly X036 continued. The average burnup on the highest burnup capsule in the subassembly was 39,750 MWd/Te (4.47 at. %). Target burnup for these capsules is 50,000 MWd/Te (5.6 at. %).

7.4 HIGH POWER INTERMEDIATE BURNUP IRRADIATIONS

7.4.1 Group F3B (Density and Stoichiometry Effects)

The Group F3B capsules were removed from subassembly X027 and an interim examination was

Table 7-3
BURNUP AND FISSION GAS RELEASE FRACTIONS FOR THE GROUP F6 FUEL PINS

Fuel Pin	Calculated Burnup(a) (at. %)	Measured Burnup(b) (at. %)	% Fission Gas Released	
			Based on Calculated Burnup	Based on Measured Burnup
FOJ	6.29	5.74	57.5	62.9
FOK	6.07	5.57	51.3	56.0
FOL	6.15	6.36(c)	49.6	48.1
FOM	6.39	(d)	38.8	(d)

(a) Peak burnup based on EBR-II fission rates and corrected for burnup depletion

(b) Peak burnup based on neodymium isotope analysis

(c) Value is high compared to other samples. A second analysis is being run for confirmation.

(d) Fuel pin failed.

Table 7-4
BURNUP AND FISSION GAS RELEASE FRACTIONS FOR THE GROUP F8A FUEL PINS

<u>Fuel Pin</u>	<u>Calculated Burnup(a) (at. %)</u>	<u>Measured Burnup(b) (at. %)</u>	<u>% Fission Gas Released</u>	
			<u>Based on Calculated Burnup</u>	<u>Based on Measured Burnup</u>
F8A	5.43	4.65	57.9	67.6
F8C	5.28	4.49	57.0	67.0
F8E	5.40	4.69	54.0	62.2
F8G	5.45	4.82	55.3	62.5

(a) Peak burnup based on EBR-II fission rates and corrected for burnup depletion.

(b) Peak burnup based on neodymium isotope analysis.

performed at the EBR-II site. Neutronradiographs of the capsules indicated that pin E1H-33 failed. Pin E1H-33 contained pellet fuel of 90.1% smeared density clad in Type-304 stainless steel. The oxygen-to-metal ratio of the mixed-oxide fuel was 1.975. Six Group F3B capsules, pins E1H-21, -23, -26, -29, -30, and -33, were selected for destructive examination at the GE, Vallecitos Nuclear Center. If the interim examination does not indicate any problems related to further irradiation of the capsules, the twelve remaining Group F3B capsules will be reconstituted into a new subassembly for irradiation toward a burnup goal of 120,000 MWd/Te.

7.4.2 Group F4 (Low-Density-Fuel Behavior at High Heat Ratings)

The average burnup calculated for the pin with the highest exposure in the Group F4 capsules in EBR-II subassembly X070 was 71,200 MWd/Te (8.04 at. %). These capsules are being irradiated toward a target burnup of 80,000 MWd/Te (9.03 at. %).

7.5 ELEVATED CLADDING TEMPERATURE TESTS

7.5.1 Group F5 (Fuel Performance at Peak FCR Cladding Temperatures)

The 19 Group F5 capsules in EBR-II subassembly X064 continued their irradiation toward a target burnup of 100,000 MWd/Te (11.2 at. %). The average burnup on the highest burnup capsule was 42,000 MWd/Te (5.51 at. %). The X064 subassembly is scheduled to be removed from the EBR-II core at the end of run 44B for an interim examination and reconstitution into new subassembly hardware for irradiation at a reactor power of 62.5 MW.

7.5.2 Group F10A and F10B (Cladding Performance at Peak FCR Cladding Temperatures)

Sodium filling of the 19 Group F10A capsules was completed. Inspection of the sodium bonding is in progress. Work on the Group F10A data package to support the insertion of the capsules in the EBR-II was initiated. The Group F10A and F10B data packages may be combined with concurrence of the EBR-II Project because of the similarity of the experiments.

Approval of the method proposed by GE for attaching coolant flow equalizers on the twelve outer capsules of the Group F10A and F10B tests was obtained from the EBR-II project. Design of the parts required was completed and procurement was initiated. The purpose of the flow equalizers is to reduce differences in coolant flow rate and therefore differences in temperature between capsules in the peripheral row and those in the inner rows of the subassembly.

Wire wrapping and final inspection of the last six Group F10B capsules was completed. All of the Group F10B capsules are now complete.

7.6 UNENCAPSULATED FUEL PIN IRRADIATIONS

7.6.1 Group F3A (Unencapsulated Fuel Pin Irradiation to Evaluate Fuel-Cladding and Cladding-Coolant Interactions)

The 16 unencapsulated Group F3A fuel pins in EBR-II subassembly X040A were removed from the reactor at the end of run 42 and an examination of each pin was initiated at the EBR-II site. The average burnup on the

highest burnup pin was 45,980 MWd/Te (5.17 at. %) and the peak fluence was 0.37×10^{23} nvt ($E > 0.1$ MeV). Details on the fabrication parameters and operating conditions of the Group F3A fuel pins were reported previously¹ along with the results of an interim examination performed at approximately 3 at. % burnup. Results of the examination conducted at the EBR-II site at 5.17 at. % burnup are presented below.

7.6.1.1 Disassembly

The X040A subassembly was removed from the reactor basket and cleaned with moist argon followed by a water flush. The subassembly was then dried with argon. The hexagonal can was cut and easily removed from the assembly. The fuel pins appeared bright and clean through the hot cell window.

7.6.1.2 Visual Examination

Each of the fuel pins was visually examined with a stereomicroscope. The appearance of the pins was similar to that observed at 3 at. % and no visual evidence of cladding deterioration was observed. More residual sodium compounds were observed on the pins than during the earlier interim examination. The sodium compounds were most often observed in the vicinity of the top end plug weld.

The wire wrap on most of the pins was observed to be loose with clearance between the wire and pin estimated to be between 5 and 10 mils.

7.6.1.3 Diameter Measurements

Diameter measurements were performed on the pins at discrete axial locations and at the same orientations as used during the interim examination performed at approximately 3 at. % burnup. The measurements obtained during the interim examination indicated that the Type-316-stainless-clad pins had increased less in diameter than the Type-304L-clad pins. This trend was confirmed by the measurements obtained during the current examination. The maximum diameter changes measured at 3 at. % and 5 at. % burnup fall within the ranges plotted in Figure 7-7. The maximum diameter change for the Type-304L-clad pins was 2-1/2 times as great as the maximum diameter change measured for the Type-316-clad pins.

7.6.1.4 Neutronradiography

The neutronradiographs of the pins appeared similar to those taken at approximately 3 at. %. Center voids were observed in all pins and no abnormal fuel distributions were observed.

The fuel pins were shipped to the GE Vallecitos Laboratory for destructive examination.

7.6.2 Group F9A (High-Power, High-Burnup, Unencapsulated Fuel Pin Irradiations)

The 37 Group F9A (unencapsulated) fuel pins were removed from the EBR-II core and placed in the storage basket at the end of reactor run 42. Originally the X043 subassembly containing the Group F9A fuel pins was to be reinserted in the reactor if the results of the examination of the Group F3A pins were favorable and the EBR-II Project's analysis of the Group F3A examination data indicated that the risk associated with continued irradiation of the Group F9A pins was low. Presently a decision is pending to split all unencapsulated fuel pin subassemblies and continue their irradiation in 19-pin Mark E subassemblies designed for high-risk experiments. If this policy is adopted, the Group F9A fuel pins will remain in the reactor storage basket until they can be reconstituted into Mark E subassembly hardware. It is estimated that the Mark E hardware will be available in November 1970.

7.6.3 Group F9B (Medium-Power, High-Burnup, Unencapsulated Fuel Pin Irradiations)

The Group F9B fuel pins continued their irradiation in EBR-II subassembly X062. The calculated average burnup of the highest burnup pin in Group F9B is 36,900 MWd/Te (4.15 at. %). Target burnup for these pins is 100,000 MWd/Te (11.2 at. %).

7.6.4 Group F9C (High-Power, High-Burnup, Unencapsulated Fuel Pin Irradiations)

Calculated average burnup of the highest burnup pin in the 37-pin Group F9C experiment was 50,450 MWd/Te (5.67 at. %). These fuel pins are now the lead burnup unencapsulated fuel pins under the FCR Program. The Group F9C fuel pins were removed from the reactor core at the end of run 44A and placed in the storage basket. The subassembly will remain in the basket until a decision is made whether to allow unencapsulated subassemblies to continue irradiation to higher burnups or to be reconstituted into Mark E subassembly hardware.

7.6.5 Group F9D (Medium-Power, High-Burnup, Unencapsulated Fuel Pin Irradiations)

The calculated average burnup of the highest burnup pin in the Group F9D test was 37,200 MWd/Te (4.18 at. %). Target burnup for these pins is 100,000 MWd/Te (11.2 at. %).

7.6.6 Group F9E (High-Power, High-Burnup, Unencapsulated Fuel Pin Irradiations)

The request for approval-in-principle for the Group F9E test was granted by AEC-RDT and fabrication of the fuel pins was initiated. Fabrication of the mixed-

Figure 7-7. Maximum Diameter Changes Measured on Group F3A Fuel Pins After Fuel Burnups of ~ 3 at.% and ~ 5 at.%

oxide pellet fuel was completed and the cladding was prepared and inspected prior to fuel loading.

Discussions were held with ANL and EBR-II personnel concerning the conceptual design of the grid spacer subassembly for the Group F9E test. Modifications of the subassembly design to make disassembly and remote reassembly possible in the EBR-II Fuel Examination Facility were discussed and are being incorporated into the design. The design support for this subassembly is being carried out under Task G-6.

7.6.7 Group F11A (Cladding Performance at Peak FCR Cladding Temperatures)

Fabrication of the 37 Group F11A fuel pins was completed with the exception of wire wrapping the fuel pins. The wire wrapping of the pins will be performed after a fixture is fabricated to measure the tension in the wire after wrapping. Design of the fixture was completed and procurement of the necessary parts was initiated.

7.7 PRE-IRRADIATION TRANSIENT TESTS

7.7.1 Group 7 (Effect of Irradiation, Density and Axial Restraint)

The four Group F7 capsules were removed from subassembly X059 along with the Group F2 capsules at the

end of EBR-II run 42. The Group F7 capsules will be returned to the GE Vallecitos Nuclear Center for non-destructive examination. The fuel pins will be re-encapsulated for re-irradiation in TREAT under Task C. The calculated average burnup in the peak burnup capsule is 53,400 MWd/Te (5.86 at. %).

7.8 CAPSULE F12B, IN-PILE AXIAL EXPANSION AND GAS RELEASE OF MIXED OXIDE FUEL

Fabrication of capsule F12B was continued. Procurement of pressure and displacement transducers for the fuel pin continues to be a problem. The vendor has not been able to supply units which pass inspections for gas leakage and electrical continuity. The F12B capsule is to be irradiated in the GETR (thermal flux test reactor) and will provide information on the axial thermal expansion and gas release characteristics of mixed-oxide fuel during irradiation to burnups up to 100,000 MWd/Te.

7.9 CAPSULES F12P AND F12Q, EFFECT OF RADIAL RESTRAINT ON FUEL SWELLING

An interim examination was performed on the F12P and F12Q fuel pins after a fuel burnup of approximately 40,000 MWd/Te was reached. Previous examinations were carried out at burnups of approximately 10,000; 20,000; and 30,000 MWd/Te. The profile traces obtained during

each examination are plotted in Figure 7-8. The traces indicate that the 15-mil-wall cladding is more effective in restraining the fuel expansion (fuel is axially restrained) than the 10-mil-wall cladding.

Diameter and calculated volume changes obtained for each burnup increment are plotted in Figure 7-9. The values for the diameter and volume change were obtained by integrating the profile curves. The cross-hatched regions represent the spread in the data obtained from profile measurements made at four circumferential orientations around the pins. The data show that the 15-mil-wall cladding restrained the fuel more effectively, both during the initial startup which produced cladding strain due to differential thermal expansion between the fuel and cladding (as evidenced by the values obtained at 10,000 MWd/Te burnup), and the subsequent cladding strain produced by thermal cycling and fission product swelling in the fuel. A volume increase of 0.56% $\Delta V/V$ per 10,000 MWd/Te was calculated for the 10-mil-wall cladding between 10,000 and 40,000 MWd/Te ($0.24\%/10^{20}$ f/cc). This may be compared to 0.17% $\Delta V/V$ per 10,000 MWd/Te for the 15-mil-wall cladding during the same period ($0.08\%/10^{20}$ f/cc).

Capsules F12P and F12Q were re-encapsulated and inserted in the GETR on May 5, 1970. The next interim examination will be performed at a burnup of approximately 55,000 MWd/Te.

7.10 F20 SERIES, ESTABLISHMENT OF FUEL PIN PERFORMANCE LIMITS

Design of the F20 series of irradiations tests in EBR-II was initiated. These tests will be designed to provide data on the performance limits of high-burnup mixed-oxide fuels when subjected to a higher steady-state power or short-time overpower (not transient accidents) condition of operation typical of fuel shuffling. Specific objectives of the irradiations are:

1. Establish fuel rod power to produce fuel melting in EBR-II as accurately as possible as a function of various fuel pin parameters.
2. Obtain a measure of diameter change (cladding strain) which occurs when fuel pin power and temperatures are increased after accumulating appreciable exposure.

7.11 COMPARISON OF BURNUP METHODS ON IRRADIATED EBR-II DRIVER FUEL

Burnup analyses of GE fueled capsules irradiated in EBR-II (determined by Nd-148 analysis) have averaged 13 to 18% lower than predicted from calculated EBR-II fission rates. A cooperative exchange of irradiated EBR-II driver

fuel was arranged to compare results from the burnup methods employed at ANL-Chicago, ANL-Idaho, and GE-Vallecitos. A sample was cut from the axial center of the EBR-II fission driver fuel pin designated E82 of subassembly B-3045. This subassembly was placed in location 6C3 of the reactor grid on 9/27/68 where it remained until 5/23/69. It was predicted to have reached 1.1 ± 0.1 at. % burnup based on EBR-II calculated fission rates.

The pre-irradiation analysis of the casting batch which produced pin E82 was 94.72 ± 0.6 wt % uranium with an isotopic analysis of 52.00 wt % U-235, 47.45 wt % U-238, 0.269 wt % U-234, and 0.277 wt % U-236. A 1/4-in.-long section of the irradiated fuel pin was dissolved at ANL-Chicago and an ampul of this solution was sent to GE-Vallecitos for analysis. ANL-Idaho dissolved a 1/4-in.-section from each side of the ANL-Chicago section for comparison analysis. Burnup analyses by the three cooperating sites are now complete and are summarized in Table 7-5. The effective fission yields used for neodymium were adjusted for a small contribution from U-238 fission to the predominantly U-235 fission.

It is concluded that (1) burnup analysis of EBR-II driver fuel by three chemical methods (i.e., analysis for Nd-148, Cs-137, and Tc-99) are in agreement within $\pm 1\%$; (2) burnup analyses performed at GE-Vallecitos are not biased low with respect to the two other laboratories in this study; (3) analytical results for the three fission products Nd-148, Cs-137, and Tc-99 yield $\sim 16\%$ lower burnups than are predicted from calculations using EBR-II fission rates; and (4) the agreement of these results establishes the relative fission yields for U-235 of Nd-148, Cs-137, and Tc-99 with respect to each other in an EBR-II fast neutron spectrum.

7.12 DETERMINATION OF FISSION YIELDS FOR CS-137 AND SIX NEODYMIUM NUCLIDES FROM U-235 IN THE EBR-II FAST NEUTRON REACTOR SPECTRUM

Relatively few measurements have been made of fission yields in fast neutron spectra. Lisman⁴ has measured U-235 fission yields in EBR-I and Davies⁵ has measured a number of fission yields in the Dounreay Fast Reactor; their fission yields for most nuclides are in good agreement. Two of the nuclides for which the U-235 fast neutron fission yields of these authors differ are Cs-137 and Nd-150. The cooperative analysis of EBR-II driver fuel mentioned above, in which fissions occurred almost entirely in U-235, afforded an opportunity to evaluate U-235 fast neutron fission yields in the EBR-II fast reactor.

Figure 7-8. Profilometer Traces of Fuel Pins F12P and F12Q at 10,000, 20,000, 30,000, and 40,000 MWD/Te Burnup

Figure 7-9. Diameter and Volume Increase for Pins F12P and F12Q. (Fuel is axially restrained)

Table 7-5
FINAL RESULTS OF BURNUP EXCHANGE ANALYSIS

<u>Method of Analysis</u>	<u>Fission Yield (%)</u>	<u>At. % Fission</u>
EBR-II Calculated	—	1.0–1.2
GE Total Nd	17.81	0.935
Nd-148	1.77	0.945
Cs-137	6.20	0.941
ANL-Idaho		
Cs-137	6.20	0.927
Tc-99	5.85	0.933
ANL-Chicago		
Cs-137	6.20	0.931

Table 7-6 shows the Cs-137, Nd-143, Nd-144, Nd-145, Nd-146, Nd-148, and Nd-150 yields in U-235 fast fission for the EBR-II neutron spectrum along with the comparable results of Lisman for EBR-I and Davies for DFR. The Cs-137 result is the average of replicate analyses by four radiochemical methods including (1) NaI(Tl) detector analysis of Cs_2PtCl_6 , (2) NaI(Tl) detector analysis of CsClO_4 , (3) Ge(Li) detector analysis of the unseparated sample, and (4) Ge(Li) detector analysis of the sample from which all Zr-Nb-95 interference was chemically removed. The neodymium nuclides were analyzed on a single focusing solid emission mass spectrometer. The relative yields of these nuclides in this work were normalized to the absolute yield of 1.75% for Nd-148 reported by both Lisman and Davies.

It is concluded that the U-235 fast neutron fission yields in EBR-II for (1) Nd-143, Nd-144, Nd-145, and Nd-146 are in good agreement with those of Lisman for EBR-I and Davies for DFR, (2) Cs-137 is in close agreement with the value of Lisman for EBR-I, and (3) Nd-150 is in closer agreement with Davies for DFR than it is with Lisman for EBR-I.

7.13 CHARACTERIZATION OF CAPSULE HARDWARE

Characterization is an essential requirement on all materials slated for insertion in EBR-II. In response to this need, a large quantity of analytical work on wire, bar, cladding and capsule tubing was performed during the current report period. This work includes detailed chemical analyses for principal alloying elements and impurities, metallographic examinations for alloy structural condition and inclusion characterization, and, with cladding in particular, evaluation of mechanical properties.

A comprehensive evaluation of the chemistry, structure, and tensile properties at elevated temperature of the heat of Type-316H tubing (Heat No. M2357) recently procured, is under way. This material is 20% cold worked and is currently being nondestructively examined prior to its use in fuel pin assemblies.

Completed during this report period was an analysis of Heat No. 29640, capsule tubing, and a characterization of Heat No. 065226, round bar, the latter representing the first custom-quality bar procured for FCR experimental programs. The results of this characterization demonstrated that good chemical control and cleanliness can be maintained for mill run quantities of material.

Table 7-6
U-235 FISSION YIELDS IN FAST REACTOR NEUTRON SPECTRA

<u>Isotope</u>	<u>EBR-I⁴</u>	<u>DFR⁵</u>	<u>EBR-II (Row 6)</u>
Cs-137	6.20 ± 0.16	6.55 ± 7%	6.18 ± 0.18
Nd-143	5.80 ± 0.12	5.98 ± 6%	5.75 ± 0.12
Nd-144	5.26 ± 0.11	5.22 ± 6%	5.36 ± 0.25
Nd-145	3.85 ± 0.08	3.93 ± 6%	3.88 ± 0.08
Nd-146	3.00 ± 0.06	3.06 ± 6%	3.03 ± 0.06
Nd-148	1.75 ± 0.03	1.75 ± 6%	(1.75)*
Nd-150	0.832 ± 0.02	0.73 ± 6%	0.75 ± 0.02

* Normalizing value

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4. Lisman, F. L., et al., *Burnup Determination of Nuclear Fuels*, Project Report for the Quarter April 1-June 30, 1968 and Final Report In-1277, March 1969.
5. Davies, W., "Absolute Measurements of Fission Yields for 235-Uranium and 239-Plutonium in the Dounreay Fast Reactor," *Radiochimica Acta* 12 173-178, (1969).

8. TASK G-6- CORE DESIGN ANALYSIS

8.1 OBJECTIVE

The purpose of this task is to contribute to the understanding of fuel performance, core design and thermal hydraulic problem areas; to develop and use analytical and design tools to aid in this work; and to provide guidance for effective planning and coordination of experimental programs. The results of these studies provide direction for further development of analytical methods, and form a basis for establishing current design and safety criteria which, in turn, define the range of reactor characteristics and test parameters upon which the experimental programs must focus.

8.2 FUEL SUPPORT SYSTEM

The purpose of this subtask is the design and evaluation of a grid fuel rod support system suitable for use in a 1000 MWe fuel assembly, with the objective of defining typical grid-supported fuel assemblies for open-loop tests in FFTF. The feasibility of the design will be demonstrated by prior testing in EBR-II.

8.2.1 Thermal Hydraulic

COBRA-II code work has been completed. Sodium properties and heat transfer correlations are incorporated. An input option has been added to permit wetted perimeter variations for selected flow subchannels, as well as flow area variations. Dynamic programming techniques and core overlays are used to reduce the computer code requirements by 40 to 75% (as compared to the original BNWL version), depending on the size of the rod bundle investigated.

The following rod bundle models have been developed and are available for use with COBRA-II:

1. Full bundles of 7, 19, 37, 61, 91, and 127 rods.
2. One-half section of 37-rod bundle.
3. 30-degree section (1/12) of 127-rod bundle.

Computer core requirements and relative running costs of each are listed in Table 8-1. A nominal bundle configuration is assumed in each case.

A test case was run on the 37-rod half-bundle model to simulate a grid spacer in an EBR-II bundle. A 6-inch length of the bundle was considered with the 1-inch-high spacer centered at 3 inches. The area and perimeter variations in each channel due to the spacer were supplied as inputs to the program. Although the calculated flow redistribution appears reasonable, the spacer pressure drop is only 65% of that experimentally measured at Karlsruhe.¹ This difference is attributable to the inability of COBRA-II to consider form drag. For correct pressure drop calculations, spacer location and drag coefficients must be given as inputs to COBRA-II.

A study was performed to determine the pin-to-liner spacing that would minimize the circumferential thermal gradient for the edge and corner pins. The optimum spacing was found to be 42 mils, and the sensitivity of thermal gradients to changes in this spacing was evaluated as 8°F/mil for spacings less than 42 mils and was 5°F/mil for spacings greater than 42 mils.

8.2.2 Test Conditions for F9E Subassembly

The F9E subassembly is to be tested at a peak linear power density of 15.0 kW/ft. The EBR-II core positions chosen for computer analysis are the 6N2 and 6N5 positions, which are identical with respect to flux gradient.

The thermal data shown in Table 8-2 are based on fission rates given by the EBR-II project. It is assumed that no change in radial or axial power profiles will occur when the EBR-II power is increased from 50 to 62.5 MWt. The total flow through the assembly was chosen to give a mixed-mean outlet temperature of 1000°F. This temperature is within 100°F of the average of the outlet

temperatures of assemblies surrounding positions 6N2 and 6N5.

A computer code was used to generate the coolant temperature distribution within the subassembly. Thermal gradients across rods shown in Table 8-2 are conservatively large, since mixing between flow channels was assumed to be only 2%.

8.2.3 Structural Criteria

A set of interim structural criteria for use in design of the grid spacer was chosen. The purpose of these criteria is to establish design limits on stresses and strain levels for in-core components consistent with material changes at expected fluence levels. These criteria are based on a modified form of the criteria of Section III of the ASME Boiler and Pressure Vessel Code. Design stress levels to be used are chosen using recommended properties of materials under irradiation and the choice includes consideration of the effects of irradiation creep, reduced ductility, and reduced stress rupture.

8.2.4 Design Loads and Stresses in The Wave-Beam Grid Spacer

There are seven major load sources, listed in Table 8-3, which are being considered in the design of the wave-beam grid spacer fuel support system.

These loads result in four generalized stresses in the grid beams as shown in Table 8-4. The stresses resulting from each load are classified as either primary or secondary, depending on the nature of the load. Some loadings, such as thermal- and irradiation-induced pin bowing, which are normally considered secondary, result in primary stresses in the grid beams since small deflections will not substantially reduce the applied loads. This categorization of loads and stresses has been used in computer analysis of a typical grid beam spacer design, as described in subsection 8.2.5.

8.2.5 Grid Spacer Loads

Grid spacer loads have previously been computed using elastic assumptions which resulted in high loads near end-of-life when the effects of thermal and irradiation gradients are included. As a part of the rod support design effort, preliminary runs have been completed as an initial checkout and evaluation of the computer program CRASIB-1,² developed at PNL. This program is a finite-element procedure for the two-dimensional creep analysis of any statically indeterminate beam of arbitrary cross section with multiple supports. It accommodates creep, steady-state thermal loads, time-dependent loading conditions resulting from material swelling, concentrated loads, linear temperature gradients, and linear flux gradients. During the reporting period, this program has been used to investigate the sensitivity in end-of-life grid spacer loads to the various irradiation swelling and creep correlations. This has included the reported GE annealed correlation along with PNL annealed and PNL cold-worked correlations.

The results of this work are shown in Table 8-5, where a typical fuel rod (0.250-in. diameter X 0.015-in. wall) for EBR-II was analyzed. This rod was 45 inches long with a 6-inch grid pitch. The peak flux* used was 5.45×10^{18} n/cm²-h, with a peak flux gradient across the rod of 0.33×10^{18} n/cm²-h/in., and a peak temperature gradient of 217°F/in. These results show the influence of the various swelling and creep correlations on end-of-life grid spacer loads as well as the peak expected rod bending stress and in-core rod deflection.

8.2.6 Other Activities

Approval was received from AEC-RDT to utilize annealed Type-316 stainless steel material for fuel cladding

* Core position 4N1 at 50 MWt (>0.1 MeV)

Table 8-1
COMPARISON OF COMPUTER USAGE COSTS FOR VARIOUS COBRA-II BUNDLE MODELS

Model	No. of Rods	No. of Channels	Computer Core Required (K)	Relative Cost
7-rod full bundle	7	18	32	0.13
37-rod full bundle	37	78	77	6.7
37-rod half bundle	20	43	40	1.0
127-rod 1/12 section	16	25	36	0.3

Table 8-2
F9E EXPERIMENTAL CONDITION FOR POSITION 6N2 AND 6N5

<u>Geometry</u>	<u>Values</u>
Outer Channel i.d. across Flats (in.)	2.210
Gap between Outer Channel i.d. and Liner o.d. (in.)	0.068
i.d. of Liner across Flats (in.)	1.997
Wall Thickness of Liner (in.)	0.040
Rod-to-Liner Space (in.)	0.042
Pitch-to-Diameter Ratio	1.28
Cladding o.d. (in.)	0.250
Cladding i.d. (in.)	0.220
Number of Rods in S/A	37
Length of Fuel Region (in.)	13.5
Length of Lower Insulator (in.)	2.5
Length of Upper Axial Blanket (in.)	1.0
Total Rod Length (in.)	39.593
Total S/A Length (in.)	91.875
Flow Area of Central Cell (in. ²)	0.0396
Hydraulic Diameter of Central Cell (in.)	0.202
Flow Area Between Liner and Channel (in. ²)	0.494
Total Assembly Flow Area (including bypass between channel and Liner) (in. ²)	2.133
Axial Spacing of Grid Spacers (in.)	6

Table 8-2 (Continued)

<u>Thermal-Hydraulic Parameters*</u>	<u>Values</u>
Inlet Temperature ($^{\circ}\text{F}$)	700
Average Outlet Temperature ($^{\circ}\text{F}$)	1000
Hot Channel Coolant Temperature at the Top of Fuel Region ($^{\circ}\text{F}$) (including uncertainties)	1102
Maximum Mid-Cladding Temperature Averaged over 60° of Arc ($^{\circ}\text{F}$)	1064
Average Outlet Temperature of Surrounding S/A's ($^{\circ}\text{F}$)	937
Thermal Bowing Gradient on an Average Central Rod ($^{\circ}\text{F}$) (Top of Fuel Region)	12
Maximum Thermal Bowing Gradient on an Edge Rod ($^{\circ}\text{F}$)	39
Maximum Local Circumferential Thermal Gradient on an Edge Rod ($^{\circ}\text{F}/60^{\circ}$ Arc)	44
Maximum Local Circumferential Thermal Gradient on a Corner Rod ($^{\circ}\text{F}/60^{\circ}$ Arc)	72
S/A Flow Inside Liner (lb/h)	20,300
Coolant Velocity Central Flow Cell (ft/sec)	10.0
S/A Mass Flux Inside Liner (lb/h-ft ²)	1.78×10^6
Film Heat Transfer Coefficient at 1000°F for a Central Flow Cell (Btu/h-ft ² - $^{\circ}\text{F}$)	32,160
Maximum Heat Flux (Btu/h-ft ²)	7.83×10^5
Average Heat Flux (Btu/h-ft ²)	7.15×10^5
Maximum Power Density of Hottest Rod (kW/ft)	15.0
Maximum Power Density of Average Rod (kW/ft)	13.7
Maximum Power Density of Coldest Rod (kW/ft)	12.4
Total S/A Power (kW)	511
Fuel Bundle Pressure Drop (psi) (Fuel Rod Friction and 7 Grid Spacers only)	4.8

*Based on 62.5-MW reactor power level

Table 8-2 (Continued)

<u>Nuclear Characteristics*</u>	<u>Values</u>
Axial Peak-to-Average Power Density	1.12
Average Initial Burnup Rate (kW/kg)	189
Maximum Initial Burnup Rate (kW/kg)	231
Average Total Flux Level at Core Midplane (n/cm ² -sec)	2.01×10^{15}
Maximum Total Flux Level at Core Midplane (n/cm ² -sec)	2.19×10^{15}
Minimum Total Flux Level at Core Midplane (n/cm ² -sec)	1.83×10^{15}
Radial Flux Gradient at Core Midplane (n/cm ² -sec/in.)	0.228×10^{15}
Flux Greater than 0.111 MeV (%)	82
Time at Full Power to Achieve 100,000 MWd/Te at center of Lead Pin (days)	470
At. % Burnup at 100,000 MWd/Te (%)	14.3
Total Fluence in Lead Rod at a Burnup of 100,000 MWd/Te (n/cm ²)	8.9×10^{22}
Fuel Inventory (gm per pin)	
Pu-239	16.67
Pu-240	1.47
Pu-241	0
U-235	49.91
U-238	4.94
Total (gm)	72.99
Fuel Pellet Density (gm/cc)	10.34
Fuel Pellet Smeared Density (gm/cc)	9.89
O/M	1.98
Cladding Material	316 SS annealed

*Based on 62.5-MW reactor power level

Table 8-3
DESIGN LOADS

1. Pin and Channel Bowing	5. Axial Gradients
•Thermal	•Thermal
•Irradiation	•Irradiation
2. Relative Pin-To-Grid Swelling	6. Axial Pressure Gradients
•Thermal	
•Irradiation	
3. Pin Vibration	7. Axial Pin Growth
4. Grid Expansion	•Thermal
•Thermal	•Irradiation
•Irradiation	

Table 8-4
GRID SPACER DESIGN LOADS AND STRESSES

	<u>Longitudinal Beam Bending</u>	<u>Axial Beam Bending</u>	<u>Beam Compression</u>	<u>Tab Shear Stress</u>
Pin and Channel Bowing				
Thermal	P			
Irradiation	P			
Pin-to-Grid Swelling				
Thermal	S			
Irradiation	S			
Pin Vibration	C			
Grid Expansion				
Thermal			S	
Irradiation			S	
Axial Gradients				
Thermal		S		
Irradiation		S		
Pressure		P		P
Axial Pin Growth				
Thermal		S		S
Irradiation		S		S

NOTE:

P = Primary Stress
S = Secondary Stress
C = Cyclic

Table 8-5
F9E FUEL ROD BENDING FOR ANNEALED AND COLD-WORKED TYPE-316 STAINLESS STEEL
(6-IN' GRID SPACING)

316 Stainless Steel

TIME = 0 hour

Grid Load (lb)

Rod Bow (mil)

Maximum Stress (1000 psi)

-0.012	0.25		0.14		-0.29		-0.25	0.14	-0.76	0.79
		-0.2		0.04		0.17				
										-6.6

GE Annealed, Swelling, and Thermal Creep

TIME = 10,200 hours

Grid Load (lb)

Rod Bow (mil)

Maximum Stress (1000 psi)

0.047	1.9		-2.2		-0.54		0.44	0.29	-0.73	0.79
		-1.1		2.1		0.6				
			18.7							

GE Annealed, Swelling, Irradiation, and Thermal Creep

TIME = 10,200 hours

Grid Load (lb)

Rod Bow (mil)

Maximum Stress (1000 psi)

0.025	+0.81		-0.85		-0.35		0.32	0.05	-0.72	0.70
		-1.2		1.8		0.54				
			8.0							

PNL Annealed, Swelling, Irradiation, and Thermal Creep

TIME = 10,200 hours

Grid Load (lb)

Rod Bow (mil)

Maximum Stress (1000 psi)

0.042	0.56		-1.3		0.70		0.53	-0.45	-0.69	0.64
		-1.2		2.0		-1.8				
			6.0							

PNL-Cold Worked, Swelling, Irradiation, and Thermal Creep

TIME = 10,200 hours

Grid Load (lb)

Rod Bow (mil)

Maximum Stress (1000 psi)

0.008	0.22		0.14		-0.47		-0.03	0.15	-0.61	0.59
		-0.9		0.4		0.4				
					5.7					

GEAP-10028-35

and grid spacers in the F9E grid spacer fuel assembly experiment planned for testing in EBR-II. A meeting was held with EBR-II and ANL personnel to review preliminary F9E design drawings and methods of assembly and disassembly of the test assembly. A master plan for the development and testing of grid spacers was prepared and submitted to RDT.

8.3 STEADY-STATE IRRADIATION EVALUATION

This subtask is aimed at assessing the information developed from steady-state and defected fuel irradiation tests in order to support the design and operation of LMFBR fuel assemblies. The data requirements are being assessed on the basis of their impact on core design, safety, and fuel cycle economics. A draft of the final report describing recommended steady-state and defected fuel tests was completed during this period and transmitted to RDT for review.

8.4 FUEL FAILURE EFFECTS

The objective of this subtask is to make a comprehensive assessment of the information on the effects of fuel element failure which is needed to design LMFBR fuel assemblies.

The concept of fault trees is being applied to the study of fuel failure effects. By exercising the fault trees, several mechanisms were identified as being the most important ones through which failure propagation could occur. During the current reporting period several of these mechanisms have been studied. These are described in the list below. The results of these investigations are reported in the following discussions:

1. Assessment of a crack failure progressing to a large-area failure.
2. Assessment of a large-area failure progressing to a burst failure.
3. Assessment of failed-fuel-pin debris and geometry.
4. Trapping of debris within a fuel assembly.
5. Thermal effects of trapped debris.
6. Thermal effects of fuel pin swelling.
7. Operating life of an overheated fuel pin.
8. Structural analysis of channels and pins.
9. Molten fuel/cladding-sodium thermal interaction.

8.4.1 Progression of a Crack Failure to a Large Area Failure

The progression of a hairline crack or pinhole failure in a pin to a large-area failure in the same pin is being considered. Leakage of sodium through stainless steel tubes at LASL³ suggests that sodium can leak through small

breaches of the cladding at operating temperature and sodium has been observed in nearly all known pin failures. For the purposes of the fault tree analysis, it was assumed that any cladding failure that allows release of fission gas will also admit sodium to the fuel pin. A "crack" failure has thus been redefined to allow coolant ingress but the sodium is unable to wash fuel or solid fission products out into the coolant stream as in a large-area failure.

By this definition, another mechanism of progression of a "crack" failure to a "large area" failure was added to the fault tree analysis; that of fuel-sodium reaction product swelling.

8.4.2 Progression of a "Large Area" Failure to a Burst Failure

Internal pressure in a fuel pin with a large-area failure may provide the driving force necessary to expel fuel into the coolant stream. This expulsion defines a "burst" failure and can lead to more serious consequences than the original large-area failure.

A model of a fuel pin was used in analyzing the potential pressures that could be contained in the fuel. The model consists of a central void, a plastic region of fuel that transmits pressures hydrostatically and an outer, brittle region that must provide the restraint to internal pressures when cladding restraint is lost through failure. The thickness of the brittle region increases as the power produced by the pin decreases. Structural analyses were performed on the brittle region in order to determine if it can restrain large pressures.

In order for the analyses to be meaningful, a strength of the fuel must be established. Out-of-pile data suggest that the strength in compression of UO_2 is in the range of 100,000 psi and strength in tension may be around 5000 psi. Confidence in these values is very low because the physical condition of fuel in-pile is much different from the unirradiated material.

From this study it appears that the internal pressures which the fuel and cladding may sustain could be high enough to indicate that fracture and ejection of some of the solid fuel adjacent to the failure might be credible following a period of steady-state reactor operation.

8.4.3 Assessment of Failed Fuel Pin Debris and Geometry

A brief study was made of photographs taken of irradiated fuel pins, both failed and sound. The purpose was to estimate the size distribution of fuel particles and cladding debris that could be expelled into the coolant in the event of a burst failure. The dimensions of individually identifiable fuel particles were measured. The maximum

dimension observed corresponded to the fuel radius, or about 0.1 inch. The average maximum dimension across a particle was about 0.03 inch.

Estimates of fuel loss were made for failed pins F2F, F4F, and F2O (encapsulated pins that suffered severe failures in EBR-II). All estimates of fuel loss were below 15% at any particular axial location. The fuel column appeared to have integrity even when cladding restraint was lost. Thus, as a first estimate of the debris from an LMFBR fuel pin failure, it is assumed that 15% of the fuel from the failure zone will be lost and the particles will have maximum dimensions of up to 0.1 inch.

8.4.4 Trapping of Debris Within a Fuel Assembly

An effort has been made to identify the areas within a fuel assembly where solid particles might be trapped. This was done as a function of the fuel assembly design parameters, namely the pin diameter, pitch-to-diameter ratio, spacer type, and the pin-to-channel wall spacing. In addition, an attempt was made to estimate whether fuel particles which enter the coolant streams will be carried along with the flow or will fall downward due to gravitational forces.

The sites available for trapping debris are largely dependent on the type of fuel pin spacer employed. In the present effort wire-wrap and grid-type spacers were studied. It was limited to one example of each type. An examination of the coolant flow areas available with each spacer type reveals what the minimum particle size has to be for it to be trapped at a given site. Smaller particles have the potential of passing through the site without being trapped.

It appears that grid spacers will trap a smaller size of particle than will wire-wrap spacers. These small trapping sites, however, do not have the potential for developing into more extensive blockages which are due to axial pile-up of additional particles. Trapping sites where axial pile-up can occur are smaller for wire-wrap fuel assemblies than for grid-spacer assemblies. Wire-wrap spacers are more restrictive than the grids in this sense.

The above assessment of the characteristics of debris which may originate from failed fuel pins showed that the maximum particle size would be on the order of 0.1 inch, and the average particle size would be closer to 0.03 inch. Thus, most of these fuel particles are small enough to pass through most of the trapping sites present in a fuel assembly. Only the larger debris would be trapped.

Trapping may occur below a failure due to the likelihood that the large particles will fall rather than be

carried along with the coolant flow. Such trapping is more likely to occur in the interior of the bundle than at the channel wall.

8.4.5 Thermal Effects of Trapped Fuel Debris

Fuel debris which becomes trapped between fuel pins or between the fuel pins and the channel wall will undoubtedly cause some local overheating. The overheating may lead to failures in the fuel pins affected, and it may even cause damage to the channel wall. To study this problem an analysis was made of a fuel pin adjacent to a channel wall between which fuel debris was assumed to have collected. The analysis of the overheating effects was carried out using the THTD computer program. The basic geometry and nodal structure assumed are shown in Figure 8-1. All outside surfaces are assumed to be adiabatic. In modeling the system characteristics such as coolant temperatures, coolant flow rate, and pin geometry and power were taken from the highest-power fuel assembly of GE 1000 MWe LMFBR Follow-On Study design as reported in GEAP-5678.⁴ The particle was assumed to become trapped at the core axial midplane. This is the location of peak power generation and it should result in the largest temperature perturbations.

The work sought to explore the effects of several parameters on the resulting channel wall and cladding temperature. These parameters included the pin-to-wall spacing, channel wall thickness, channel-to-channel gap, and the extent of the blockage. The heat transfer coefficient between the trapped fuel particle and the solid surfaces in contact with it was also varied.

8.4.5.1 Channel Wall Overheating

The extent of local overheating of the channel wall due to trapped fuel debris is indicated in Table 8-6. The table gives the peak local absolute temperature increase due to the blockage as a function of the major parameters included in the study. The results of the analysis indicate that the particle size (extent of blockage) and the channel-to-channel gap spacing have the strongest influence on the channel temperature. Parameters such as channel wall thickness, pin-to-wall spacing, and contact coefficient have a rather small effect. Over the explored range of these latter parameters the temperature increase varied by less than 50°F. Note that the ΔT refers to the temperature difference at a given spot resulting from the presence of the blockage. It is the temperature at that spot with the blockage minus the temperature at the same spot without the blockage.

In varying the extent of the blockage the debris cross section was changed, rather than the axial length. The

Figure 8-1. Computer Model Describing Local Flow Blockage

Table 8-6
CHANNEL WALL TEMPERATURE INCREASE DUE TO TRAPPED FUEL DEBRIS

Case No.	W (Btu/h-ft ² · °F)	X (in.)	Y (in.)	Z (in.)	Blockage Extent	Highest ΔT (°F)	Peak Channel Temperature (°F)
1	1000	0.05	0.070	0.25	Moderate	90	1040
2	10000	0.05	0.070	0.25	Moderate	130	1100
3	10000	0.03	0.070	0.25	Moderate	110	1080
4	10000	0.03	0.140	0.25	Moderate	140	1110
5	10000	0.03	0.070	0.00	Moderate	245	1220
6	10000	0.050	0.070	0.25	Severe	420	1370

Note:

- W = Contact coefficient
X = Pin-to-channel spacing
Y = Channel wall thickness
Z = Channel-to-channel spacing

computations showed that the channel ΔT is nearly fully established in a short axial length. Extending the blockage axially raised the peak temperatures, but only marginally above the initial ΔT incurred as soon as the blockage starts. The cross-sectional areas of the fuel debris assumed in the analysis are quoted below, along with the equivalent particle diameters if they are assumed to be cylindrical in shape. The values apply to a pin-to-wall spacing of 0.050 in.

Blockage	Cross-Sectional Area (in. ²)	Equivalent Particle Diameter (in.)
Nominal	0.0064	0.064
Severe	0.0099	0.079

With the severe blockage the portion of the channel wall in intimate contact with the fuel particle was more than twice that assumed for the nominal-blockage case. This greater coverage results in a channel wall ΔT which is nearly three times that occurring from the nominal case.

The cases investigated wherein the channel-to-channel gap distance was varied indicated that heat transfer through the channel to the gap coolant plays a significant role in minimizing the channel wall temperature increase. When the gap spacing was assumed to be negligible the peak channel wall absolute ΔT was about 250°F. The very pessimistic assumption was made in this case that the outer

wall surface was adiabatic. Actually some heat would be transferred to the adjacent channel. The predicted ΔT sets an upper bound for this case under the given assumptions. Increasing the gap size from 0 to 0.25 inch reduced the peak ΔT to about 100°F.

8.4.5.2 Fuel Pin Cladding Overheating

The overheating effects on a fuel pin due to trapped fuel debris are much more severe than those experienced by the channel wall. This is to be expected since heat removal from the pin is seriously restricted wherever the cladding is in contact with the debris.

Table 8-7 shows the magnitude of the cladding hot spot ΔT's caused by fuel debris. In general, the values are a factor of three higher than those experienced by the channel wall. The peak cladding temperatures at the hot spots are also tabulated.

For the cases of moderate type blockages a 10-degree arc along the cladding outer surface was assumed to be in contact with the fuel debris. The more severe blockage extended this arc to 40 degrees.

The present analysis has indicated that a relatively small amount of trapped debris can cause serious overheating of the cladding. This is the case when as little as a 10-degree segment of the cladding circumference is insulated from the coolant by the debris. Larger blockages lead to much more severe overheating. Fuel pin failures will likely propagate in this manner as long as the debris released from a failed pin can be trapped and brought into contact with the cladding of an intact pin.

The results of the present study concerning cladding temperatures are consistent with the results of a similar analysis as reported in GEAP-5618.⁵

8.4.6 Thermal Effects of Fuel Pin Swelling

A possible mechanism for fuel failure propagation is local coolant flow redirection due to swelling of a failed fuel pin. It is postulated that chemical reaction between the fuel and the coolant can cause fuel swelling. As the fuel pin swells in the region of a cladding failure it restricts flow in the surrounding coolant passages. This causes an asymmetrical overheating of the adjacent pins. The adjacent pins will tend to bow toward the swollen pin, further reducing the coolant flow area between them and producing stresses in their cladding due to restraint of the bowing by the fuel pin spacers. These stresses, together with the local overheating could cause failure of the adjacent pin.

The effect of the temperatures and stresses in the adjacent pins has been studied to identify the principal areas of uncertainty.

In analyzing this problem, only two major areas of uncertainty were found. One uncertainty is in predicting the amount of interchannel flow mixing within a fuel bundle. Although some data are available, more precision is required. The allowable axial extent of fuel swelling in the failed pin is strongly affected by varying the assumed amount of mixing within the range of existing data. This parameter can be characterized well by ex-reactor tests.

The second major uncertainty is in evaluating the damage to fuel pin cladding due to excessive forces at the spacer contact points. The extent to which the cladding can relieve these forces through either thermal- or irradiation-induced creep remains to be evaluated. However, more data are required on the extent of irradiation-induced creep and on the total strain to failure of cladding material operating in a fast neutron flux. These two properties can best be determined by in-reactor tests.

The following are the conclusions of this study:

1. The deflection of a fuel pin is greatest when the temperature gradient is imposed over a single free span between two spacer contact points.
2. The largest bending stress occurs when the temperature gradient is imposed over three or more spans.
3. For spiral wire leads of less than 12 inches and for grid spacer axial spacings of less than 4 inches, the allowable temperature gradient across a fuel pin is more likely to be limited by the strains due to the spacer contact forces than to the magnitude of either pin bowing or of the bending stresses.
4. If the swollen region is short (less than 3 inches) the ΔT across the adjacent pin does not produce either excessive stresses or excessive bowing, for a uniform swelling up to 35% $\Delta D/D$.
5. If the swollen region is long (greater than 6 inches) then the acceptable amount of swelling is very sensitive to the amount of interchannel mixing which can be assured.

Table 8-7
LOCAL FUEL CLAD TEMPERATURE INCREASE DUE TO TRAPPED FUEL DEBRIS

Case No.	W (Btu/h-ft ² -°F)	X (in.)	Y (in.)	Z (in.)	Radial Blockage Extent (deg)	Highest ΔT (°F)	Peak Cladding Temperature (°F)
1	1000	0.050	0.070	0.25	10	380	1500
2	10000	0.05	0.070	0.25	10	420	1530
3	10000	0.03	0.070	0.25	10	425	1535
4	10000	0.03	0.140	0.25	10	460	1560
5	10000	0.03	0.070	0.0	10	460	1560
6	10000	0.05	0.070	0.25	40	1325	2450

Note:

W = Contact Coefficient

X = Pin-to-channel spacing

Y = Channel wall thickness

Z = Channel-to-channel spacing

8.4.7 Operating Life of an Overheated Fuel Pin

The operating life of an overheated fuel pin has been studied. An increase in fuel pin temperature is assumed to be caused by some flow restriction due to the swelling of an adjacent failed pin or the trapping of debris from fuel pin failures. The higher temperature causes an increase in the corrosion rate due to flowing sodium. It also causes an attack of the cladding inner surface by fuel and/or fission products. The mechanical properties of the cladding also change rapidly with temperature. These are the only effects of overtemperature considered in this study.

The temperature range considered was from 1200 to 1800°F at the inside surface of the cladding. Cladding thicknesses of 0.010 and 0.015 inch were considered. Sealed and vented pins were studied. Sealed pins were assumed to have a fission gas pressure of either 500 or 1000 psi at end-of-life (residence time in core = 2 years). The vented pins were assumed to have constant pressure of 50, 100, and 300 psi after an initial 1000 hours to reach equilibrium. The overheating of the fuel pin was assumed to start at either the beginning-of-life, middle-of-life, or near the end-of-life.

Three different cladding failure criteria were used. For the conservative (lower) limit on pin lifetime, the cladding is assumed to fail when the stress in the cladding is sufficient to cause creep rupture in the elapsed time from the initiation of the overtemperature situation (overtemperature is assumed to start at time = zero). This conservatively assumes that the stress in the cladding has existed over the entire duration of the temperature transient instead of increasing monotonically from zero, as it actually does. The optimistic (upper) limit is determined by the smaller of two different criteria. The tensile strength criterion assumes that failure occurs when the stress in the cladding equals the tensile strength of the cladding at the temperature under consideration. The second optimistic criterion is simply twice the conservative value of time-to-failure. This assumes that the cladding is stress-free up to the conservative failure time, T , and then the stress is applied. Since the stress causes rupture in a time period, T , failure occurs at time $2T$ measured from the initiation of overheating.

A computer solution was used to predict the time-to-failure under the various conditions. The stress in the cladding is calculated at each time-step and compared to the desired failure criterion. The stress increases continuously due to the effects of pressure buildup and/or cladding thinning due to corrosion and attack. The time-to-failure is printed out for each temperature.

The analysis proved to be highly sensitive to the severity of the attack on the cladding inner surface by fuel and/or fission products. It was also sensitive to the choice of failure criteria, especially in the lower-temperature cases. The results of this study were sensitive to the corrosion rate due to flowing sodium only for the pins that were overheated late in life.

There were sixty cases run under varying conditions. Vented pins were generally longer-lived than sealed pins, especially late in life when the sealed pins had developed large pressures. The results for some example cases are given in Table 8-8. The numbers in the table are times-to-failure for the cases at each temperature; the right column under each temperature was obtained by using the two optimistic failure criteria and the left column results from the pessimistic criterion. The difference between the two represents the uncertainty due to the choice of failure criteria. Blanks indicate that failure does not occur during the 2-year residence.

8.4.8 Structural Analysis

The development of tools for the structural analysis of core components with loadings induced by accident has continued during the quarter. Solutions have been obtained for shock loadings external to the hexagonal fuel channel, as well as the fuel pin, and these solutions were programmed for computer solution.

To recapitulate briefly, in the analysis of core components for accidents in a liquid-metal-cooled fast breeder reactor there have been postulated dynamic loading from a shock pulse originating from vapor ejection of the coolant followed by re-entry of the liquid coolant. The shock pulse may originate from the thermal expansion incident to the violent heating of the liquid, to the vapor explosion, to the liquid hammer following the abrupt halt of the liquid column, or to some combination of these factors.

It is not proposed that evaluation of the magnitude, velocity, extent, or attenuation of the pulse be discussed now. Acoustic theory would indicate that a wave traveling in a liquid medium may be transmitted, reflected, and absorbed by an immersed barrier as a function of the geometry, pulse wave length, and the characteristics impedances of the liquid and barrier. In addition, kinetics would indicate that impulse from the reflected pressure pulse and absorbed energy would be a function of the wave length of the pulse and the natural frequency of the obstacle for the mode excited by the pulse. The manner and magnitude of the energy deposited on, and absorbed by, the barrier has been discussed in the previous quarterly

Table 8-8
TIMES-TO-FAILURE FOR OVERHEATED FUEL PINS
(IN HOURS)

Fission Gas Accommodation	EOL Pressure	Cladding Thickness (mils)	Start of Overheating	Pessimistic	Optimistic	Pessimistic	Optimistic	Pessimistic	Optimistic	Pessimistic	Optimistic
				1200°F		1400°F		1600°F		1800°F	
Sealed	500	10	BOL	11,500	—	2,600	5,200	180	190	60	60
Sealed	1000	10	BOL	6,300	12,600	1,500	3,000	175	185	60	60
Sealed	500	15	BOL	—	—	6,080	12,160	470	580	100	100
Sealed	500	15	EOL	—	—	46	100	0.5	1	<0.1	<0.1
Vented	50	10	BOL	—	—	6,880	13,760	180	190	58	60
Vented	300	10	BOL	—	—	450	880	130	130	50	55
Vented	50	15	BOL	—	—	—	—	410	520	100	100
Vented	50	15	EOL	—	—	—	—	215	300	50	68

Note: Dash indicates failure does not occur in the normal fuel pin lifetime.
EOL cases are assumed to have 1000 hours remaining in their normal lifetime

report. It is hoped that a description of the structural analysis utilized for such loading of the barrier will suffice.

The shock pulse is a dynamic loading of high pressure and brief duration that results in an impulse load on the barrier that may be categorized by the energy deposited. This energy must be absorbed by the internal energy of the deformed barrier and the problem is one of relating the internal energy and accompanying deformation to the energy deposited. If the deformation is consistent with the loading, then the internal energy will also be consistent, regardless of the manner of imposing the deformation. It is proposed that a static loading resulting in a similar deformation mode can be employed to relate deformation, energy absorbed, stresses, and strains for the barrier.

Analyses employing this simulation have been performed for hexagonal fuel channels with pulses originating internal and external to the channel. In addition, analysis has been performed for fuel rods supported at intervals by spacers. The method of analysis hinges upon the choice of static loading that will simulate the deformation pattern imposed by an inertial loading resulting from the shock pulse. An exact simulation is difficult as the true loading is a function of the resistance to deformation of each point of the structure. A judicious choice frequently can be made that is conservative, i.e., that results in a less efficient energy absorption than the exact

loading. Following the choice of load, the structure is analyzed for successively increasing loads and deformations extending well into the plastic range and approaching the factor limiting the ultimate usefulness of the structure. Equating the energy deposited by the pulse to the internal energy indicates the appropriate load equivalence and corresponding stress, deformation, and strain.

For the channel loaded from the inside, the loading has been chosen as uniform for the cross section and a computer solution utilized for the successive analyses required. This loading will result in more bending and less membrane deformation in the hexagonal cross section than would the inertial loading so that it is a less efficient (and therefore conservative) energy absorber. The channel is successively analyzed in the elastic range for bending deformation of the walls, for elastic-plastic bending, and finally for plastic membrane resistance with adjacent panels furnishing the abutment reactions necessary. Figures 8-2 and 8-3 illustrate the resultant relations between energy, maximum deformation, simulated load and strain for the internally loaded channel.

The externally loaded channel has one flat loaded uniformly and is supported at the corners of the opposite flat. In this case, the deformation of the entire cross section contributes to energy absorption from the one loaded flat and the ability to resist load by membrane action is absent.

Figure 8-2. Channel Response to Internal Pressure Loads

Figure 8-3. Channel Internal Pressure Load — Deflection vs Thickness

Figure 8-4 illustrates, for one set of parameters, resultant relations between energy, maximum deformation, simulated load and strain for this loading condition.

The fuel rod is loaded as a beam with fixed supports (approximates continuous beam over multiple supports). Reactions to support membrane forces are not available so beam action is the only load resistance available. Figure 8-5 illustrates, for one set of parameters, the resultant relations of energy, maximum deformation, simulated load and strain.

8.4.9 Fuel/Cladding-Sodium Thermal Interactions

Work on the analytical modeling of fuel and cladding thermal interactions with sodium included the following activities:

1. A spherical geometry version of the GE thermodynamic model^{6,7} was used in a parameter study of work energy release of a single particle (fuel or cladding) immersed in sodium coolant.
2. The analytical model described by ANL⁸ was studied in detail. The model equations were derived in order to achieve an understanding of the modeling assumptions and limitations.
3. Further studies were made with the slab geometry wave propagation model.⁷

The spherical geometry thermodynamic analysis extends the slab geometry study of reference 6 in two ways:

1. The immersed-particle concept is, in certain situations, more realistic than the infinite-surface-contact implied in the slab model. This is particularly true when the initiating event is an overpower transient, leading to expulsion of molten cladding and fuel into the channel.
2. The highly conservative thermodynamic assumptions of the slab model are not implied in the results presented for the spherical model. (These assumptions were discussed in Reference 7.)

Results of the parametric studies are given in subsection 8.4.9.2.

The work available from liquid expansion appears to be very low, based on the more realistic thermodynamic assumptions. The efficiency of energy conversion from (potential) thermal to mechanical form is on the order of 10^{-5} .

The thermodynamic model applies to a single particle. When the coolant/fuel ratio is sufficiently large, then interactions between particles might be ignored, and the single-particle results would be descriptive of the process. However, the mass ratio (coolant/fuel) must be fairly large (i.e., 1.0 to 10.0 depending on particle size and

initial conditions) to justify ignoring interactions. Since the normal ratio is of the order 0.1, it must be concluded that the single-particle analysis has very limited application.

The ANL modeling approach is based on bulk heating of a specified coolant volume due to dispersed fuel particles. This appears to be a logical treatment of situations in which such dispersion can occur, and deals with particle interaction in pressure generation in a simple, but believable manner. Further analysis utilizing the ANL modeling concept, is planned during the next quarter.

The wave propagation studies were made to evaluate numerically the "small amplitude" assumptions of the model equations.⁷ On the basis of these studies, it appears that the assumptions are reasonable and would not lead to significant inaccuracy in the results.

8.4.9.2 Spherical Geometry Thermodynamic Model

This describes analytical results obtained from the spherical-geometry version of the thermodynamic model originally described in reference 6. The present work extends the originally published work in two ways:

1. The analysis is carried out in spherical geometry rather than semi-infinite slab geometry. This is a more realistic concept when we attempt to relate the model to an actual situation, and it enables us to normalize the result to unit fuel particle mass or unit fuel particle surface area. Also, we need to understand the dependency of mechanical work potential on fuel particle size in order to relate damage potential to size distribution data.
2. The second consideration is the degree of conservatism inherent in the thermodynamic model. In Reference 7 there was a discussion of the conservatism of the Reference 6 model. It was pointed out that a considerably less conservative upper limit can be derived and also that a more realistic thermodynamic model, based on constant volume heating, leads to an even lower estimate which includes only the work of isentropic expansion to ambient pressure.

8.5 COOLANT BOILING AND OPERATING SAFETY LIMITS

The object of this task is to make a comprehensive assessment of the information on coolant boiling and operating safety limits needed to design LMFBR fuel assemblies.

Two areas studied this quarter were sodium hammer and sodium vapor bubble collapse.

Figure 8-4. Channel Response to Externally Loaded Flat

Figure 8-5. Fuel Pin Beam Response to External Loads

A key question identified in the fault-tree analysis of failure propagation was the energy content of a shockwave produced by sodium hammer and vapor bubble collapse.⁷ Of chief concern are the mechanical effects as they affect rapid failure propagation.

If failed fuel or system debris causes a local blockage sufficient to cause boiling, the collapse of vapor bubbles in surrounding subcooled sodium generates pressure pulses capable of damaging adjacent fuel pins and channels. The most direct analogy is the cavitation phenomenon.

If the above blockage degenerates to the point of boiling in the entire channel and a slug flow chugging regime ensues, sodium hammer may occur on vapor condensation and slug re-entry into the fuel bundles. The energy potential is sufficiently large to cause concern for the integrity of the adjacent bundles.

8.5.1 Sodium Hammer

The literature on sodium hammer offers no predictions as to the probability of damage occurring in sodium-cooled reactors. The probability of sodium hammer occurring appears to be very low. A simplified, very conservative calculation of sodium hammer damage potential to a channel has been completed.

The maximum channel strain imposed due to sodium hammer at 30 ft/sec should lie between 2.5 and 5.5%. With the large uncertainties in the ductility of irradiated steel, particularly during an overtemperature transient and at high strain rates, it must be concluded that strains in this range might induce channel failures with the attendant risks of failure propagation to the fuel pins. Future analysis should consider the simultaneous response of the surrounding channels and their tube bundles to sodium hammer in a fuel assembly.

Dynamic solutions, particularly in the radial direction, are necessary, not only to determine the magnitude of pressures due to sodium hammer and fuel cladding-sodium thermal interaction but to simultaneously determine the response of adjacent channels and fuel pins.

The model investigated⁹ for calculating the sodium hammer shock wave velocity and pressure assumes an instantaneous elastic response of the structure to the imposed pressure. The dynamic description of the radial accommodation to sodium hammer pressures is not included. Provided that the duration of the pressure pulse is much longer than the response time of the radial accommodation, the dynamic response should have negligible effect on the resulting pressure.

The response of a hexagonal channel to sodium hammer pressures is extremely nonlinear, but the results of the channel deformation analysis (subsection 8.4.8) permit evaluation of the channel response. The channel volume change due to an imposed internal pressure permits

calculation of the impact velocity, propagation velocity, and pressure relationship.

Figure 8-6 shows the pressure versus impact velocity for the cases studied, including circular pipe cases for comparisons.

The channel deformation shows a plastic region where continued deformation occurs at constant pressure. This causes the "flat" pressure versus impact velocity response of Figure 8-6.

Within the tube bundle the fuel pins reduce the flow area and the resultant sodium hammer. In addition, the frictional resistance to flow prohibits a high impact velocity.

As a first estimate to include the influence of the surroundings on the channel response, it was assumed that two equal channel walls must deflect to accommodate a sodium hammer pulse. The deformation is thus characteristic of one-half the imposed pressure. This gives a much stiffer containment of the sodium hammer, with higher pressures but less strain than the single-channel-wall response. First results of the deformation of a channel due to an external load indicate a much more flexible or soft response than to an internal load. For the conservative sodium hammer velocities up to 30 ft/sec, the cross-hatched area in Figure 8-6 indicates the range of probable conservative response.

8.5.2 Vapor Bubble Collapse

Vapor bubble collapse can cause shock waves capable of damaging fuel pin. This is a condition of a spherical bubble collapse within a tube bundle. A local boiling situation generates vapor, which condenses and collapses on the edge of the two-phase plume. This collapse on or near fuel rods either behind or adjacent to the blockage would add another mechanism contributing to pin failure propagation.

No data or methods appear available to predict the efficiency of energy deposition onto the fuel pins or channels. Because the wave length of a bubble collapse pressure wave is much longer¹⁰ than the pin diameter, damage potential is probably significant only if collapse occurs in proximity to the fuel pin, analogous to cavitation.

Since the cavitation damage rate is less than 1 mil/hour for the coldest sodium in the reactor exit ($\approx 1200^\circ\text{F}$),¹⁰ cavitation damage is significant only if the local boiling condition is stable (doesn't degrade to envelop the entire fuel bundle) and the blockage is undetected. The cavitation correlation also indicates that at high subcooling the behavior of the cladding as a thin-wall member under repeated pressure pulses should be considered, particularly if slight overheating has created a fuel-cladding gap and thus reduced the degree of fuel support for the cladding.

Figure 8-6. Impact Velocity, V , (μ /sec)

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9. TASK G-10—KINETIC ANALYSIS

9.1 GENERAL

The purpose of Task G-10 is to contribute to the understanding of fast reactor safety, to develop and use analytical tools to aid in fast reactor safety evaluation, and to provide guidance for effective planning and coordination of experimental programs. The results of these studies provide direction for further development of analytical methods and form a basis for establishing current design and safety criteria which define the range of reactor characteristics and test parameters upon which the experimental programs must focus.

9.2 MODEL DEVELOPMENT

9.2.1 FREADM Code

The FREADM coupled kinetics and multichannel heat transfer calculation was coupled to a revised version of the MARS two-dimensional modified Bethe-Tait calculation. The coupled code will permit calculation of design basis accidents for fast reactor containment from the point of initiation through core disassembly. Transient temperatures, reactivity versus time, and reactor kinetics information are passed through the interface to provide transient starting conditions for the MARS calculation. The version of MARS used by Argonne National Laboratory in the SAS1A code was modified for the FREADM calculation. Linking of the major calculational segments to permit overlaps of computer fast memory reduced the size of the MARS module from ~ 44K to a total of ~ 28K for the coupled FREADM-MARS calculation.

Checkout of the interface routines has been completed. Checkout of the coupled calculation is continuing.

A MARS calculation can be initiated from a FREADM case using the following tests:

1. Elapsed problem time greater than a user-specified trigger level.
2. The reactor period is positive and less than a user-specified trigger level.
3. The reactivity is greater than a user-specified trigger level and a fuel temperature exceeds a user-specified trigger level.
4. A fuel temperature is greater than a user-specified trigger level.
5. Elapsed time from fuel failure exceeds a user-specified trigger level (input for each bundle type).

Wrapup and restart options have been added to FREADM which permit the user to initiate a problem from a precomputed transient or steady-state condition. Checkout of the coupled FREADM-MARS calculation is being done using this option.

The VENUS code package was requested from Argonne National Laboratory. Coupling of VENUS to FREADM will be investigated when the code package is received. The major innovation in VENUS is the addition of continuity considerations to the set of coupled hydrodynamics and reactor kinetics equations solved. This permits the computation of local pressures by the use of an

equation of state, which is explicitly dependent on the material density.

9.2.2 Thermal-Hydraulics

The improved core heat transfer and voiding computational model has reached its operational state as an independent model. The program is able to simulate, dynamically and thermally, loss-of-coolant-flow and flow-blockage accidents in reactor channels. The significant extensions to the previous voiding computational model are the capability to:

1. Continue heat transfer computations for both the fluid and the reactor fuel after voiding has been initiated.
2. Account for the evaporation rate from the liquid film retained on the fuel cladding surface as the liquid recedes.
3. Account for the condensation rate to cooler components of the reactor core when the computed pressure within a bubble indicates a saturation temperature above the adjacent reactor core material temperature.
4. Account for several subsequent bubbles as well as back-flow, slug-ejection, and coolant re-entry.

Core structural integrity is assumed throughout the computation.

A series of flow-blockage cases is under study, to examine the influence of various parameters on the voiding process. The parameters being considered are;

1. The amount of flow blockage.
2. The initial superheat.
3. The initial thickness of the liquid film retained on the fuel cladding.
4. The evaporation coefficient.

9.2.3 Fuel Melting Slumping and Freezing

Analysis of the flow of molten fuel and fission gas from a burst fuel pin under two-phase compressible-flow conditions is continuing. A Fortran IV program which determines the time taken to evacuate the molten fuel and the effects of the molten fuel on the coolant is being debugged. Heat transfer is assumed to take place between the fuel-gas mixture and the coolant, at a mean conductivity lying between the fuel and fission gas. To this end a fuel-sodium interaction model (previously evolved for

an FFTF study) is being converted to Fortran IV and modified to account for fuel motion.

Figures 9-3, 9-4, 9-5 and 9-6 of the previous quarterly report¹ were found to contain some errors. The following figures (9-1 through 9-4) represent corrected versions of the corresponding curves.

9.2.4 Parametric Analysis

A computer program has been written for the GE time share system to examine the sodium voiding rate due to fission gas release from a ruptured fuel pin. This analysis considers the behavior of a system after a large rupture in a fuel pin causing a coolant flow reversal. This treatment does not include problems with smaller breaks resulting in a downstream two-phase flow and no coolant reversal.

The model assumes one-dimensional slug flow and considers a single fission gas bubble. The assumptions used in constructing the model were selected to yield the upper limit for the voiding rate due to fission gas release. It is assumed that all fuel pins in the bundle being considered rupture simultaneously and at the same elevation. The fission gas is assumed to expand isothermally, which yields the upper limit for the driving pressure in the gas plenum. The pressure drop for the gas as it passes through the blanket material and end effects at the top of the coolant channel are neglected to yield the upper limit for the gas flow rate into the coolant channel. The model takes into account the effects of inertia and single-phase pressure drop for the reactor coolant and the static head in the coolant channel.

In the present study the fission gas temperature and the diameter and position of the break in the fuel pins are varied as parameters. The output includes the time, the location of the top and bottom of the gas bubble, the inlet and outlet coolant velocities, and the pressures in the plenum and in the void.

The results of a typical calculation are shown on Figure 9-5. A representative 350 MWe reactor fuel bundle is analysed under the following conditions:

Plenum temperature	1560°F
Location of rupture	top of the active fuel
Rupture diameter	0.06 inch
Pin/coolant volume ratio	1.98

¹ PA-10, 34th Quarterly Report

Figure 9-1. Fuel Failure Model Results

Figure 9-2. Fuel Failure Model Results

Figure 9-3. Fuel Failure Model Results

Figure 9-4. Fuel Failure Model Results

Figure 9.5. Computed Voiding Rate Due to Fission Gas Injection

10. TASK L—CLADDING DEVELOPMENT

10.1 GENERAL

The technical and economic feasibility of ceramic-fueled fast reactors depends on the attainment of high performance of core structural members and the fuel system. It depends especially on the survival of the cladding at temperatures and specific powers of practical interest. The present status of testing and analysis of potential core structural and cladding materials, with regard to alloys and environmental variables, indicates good prospects that adequate initial performance of structural members and cladding for fast reactor fuel elements can be obtained by using carefully specified materials within the range of compositions of commercial processing methods. This lack of test experience to support the rational selection of preferred material specifications is regarded as a major remaining obstacle to the attainment of the reliable operation of an LMFBF core.

The general objectives of the core materials development program for the next several years are to contribute to the attainment of the desired levels of fast reactor fuel element performance which, in turn, are required by the desired fuel-cycle economics. The objectives of specific activities follow:

- Determine the performance characteristics—including mechanical properties, fracture characteristics, radiation damage, and structural stability—of the most likely cladding material candidates and core structural materials under test conditions simulating the LMFBF environment.
- Investigate the rate of alloy swelling and the environmental and metallurgical factors which influence and control it in LMFBF primary alloys and those which are potential candidates.
- Determine the minimum performance criteria and the quality control techniques to measure performance against these criteria, required to establish recommended specifications for nuclear quality materials for LMFBF systems.
- Perform systematic data analysis of the response of candidate alloys to test environments which simulate reactor applications, on which to base selection of a preferred alloy for near-term LMFBF applications.
- Interpret the in-depth behavior of core materials under simulated LMFBF conditions and recommend specific material applications and design limits for LMFBF use.

10.2 CLADDING ALLOY SWELLING

The objective of this program is to monitor cladding alloy materials and S/A hardware swelling in available

PA-10 irradiated specimens to determine the relationships between swelling and fluence, irradiation temperature, alloy microstructure, and stress history. The techniques used to measure cladding swelling are liquid immersion density, dimensional analysis, and transmission electron microscopy.

10.2.1 Transmission Electron Microscopy of MT-3

The MT-3 materials capsule irradiated in EBR-II to 19,280 MWd contained as a structural member, a Type-304 stainless strip. The strip was 0.200 inch wide by 0.010 inch thick and ran the entire length of the capsule. Its initial purpose was to position the high-nickel-alloy tensile specimens. The peak total neutron fluence of the strip was calculated to be 5.8×10^{22} n/cm².^{*} The temperature distribution in the MT-3 capsule has been reported previously.²

Specimens for transmission electron microscopy were taken from the strip at various intervals to determine the degree of void formation at high temperatures (580-680°C) for highly irradiated material.

To date examination of eight specimens has shown that no voids are formed in Type-304 when irradiated above 680°C. One sample irradiated at 655°C showed only one cluster of small voids (<150 Å) in the entire foil so that the void volume fraction is essentially zero at this temperature. Foils at 645°C exhibited a low number of extremely large voids (1000 to 2000 Å). Specimens at lower temperatures show a trend of decreasing void diameter and increasing number of voids. Table 10-1 summarizes the data on samples which have been examined to date.

As can be seen in the table, there is an apparent discrepancy between the E1 and P1 samples. It must be stressed that void formation is very sensitive to temperature in this temperature region so that a slight error in the estimate of temperature in either sample could have a strong effect. It would appear from the void size and distribution that the P1 sample was actually at a higher temperature than E1.

Figure 10-1 shows the typical microstructures observed for each temperature region. The relatively high density of dislocations visible in some pictures was considered to have been introduced by the sampling technique rather than by any pre-irradiation cold work. Some samples taken at a later date by a different process showed a much lower dislocation density.

* Fluence based on EBR-II physics calculation with 1/1.2 correction factor.

A. 460°C $4.3 \times 10^{22} \text{ n/cm}^2$

B. 603°C $4.2 \times 10^{22} \text{ n/cm}^2$

C. 644°C $4.8 \times 10^{22} \text{ n/cm}^2$

D. 656°C $4.5 \times 10^{22} \text{ n/cm}^2$

Figure 10-1. Typical Microstructures for each Temperature Region

Table 10-1
VOLUME CHANGE OBSERVED IN MT-3 SPECIMENS

<u>Sample</u>	<u>Temperature (°C)</u>	<u>Fluence X 10⁻²² E > 0.1 MeV</u>	<u>Average Void Diameter A°</u>	<u>Number cm³</u>	<u>% ΔV V</u>
F1	686	4.97	0	0	0
H1	682	4.88	0	0	0
J1	656	4.53	<140	N.D.*	0
E1	646	4.96	2080	2.2 X 10 ¹²	1.4
I1	644	4.73	1450	6.5 X 10 ¹¹	0.1
B3	633	4.62	1830	3.4 X 10 ¹²	1.5
K1	603	4.19	1865	3.1 X 10 ¹²	1.6
M1	581	3.56	1110	1.3 X 10 ¹³	1.4
M3	556	3.48	1000	2.1 X 10 ¹³	1.8
N1	504	3.14	660	2.5 X 10 ¹³	0.9
A1	462	4.36	455	4 X 10 ¹⁴	2.4
O4	408	2.24	120	1.0 X 10 ¹⁵	0.12

*Not determined (due to isolated occurrence of the voids).

Figure 10-1d shows the sample at 656°C. The four or five voids visible in the upper portion of the photograph were the only ones observed in the foil. The maximum diameter is less than 150 Å.

Average void diameters were plotted versus temperature and is compared with the work of other experimenters in Figure 10-2. The measurements from MT-3 seem quite consistent up to 650°C at which point the number, density, and void size rapidly decrease to zero. When the MT-3 data are compared with previous data, it appears there is some disagreement. One explanation may be that there is a lack of uniformity in defining irradiation conditions. Another consideration is that the fluence of MT-3 specimens was higher than any of the other specimens examined so that what may be appearing is an increase in average void size with increasing fluence.

10.3 MATERIAL IRRADIATION PERFORMANCE

The objective of this subtask is to provide and follow capsules used as irradiation vehicles for irradiating cladding alloy specimens in EBR-II, and then to characterize the irradiation history of the capsules and prepare specimens for evaluation of mechanical and physical properties. A summary of this effort is shown in Table 2-6.

10.3.1 Series L-6, L-15, and MT-4—EBR-II Materials Capsules

The L-6 capsules (Mark-B7 type) are presently being irradiated in subassembly (S/A) X018B and have

accumulated a total fluence of 7.9×10^{22} n/cm² (with a target total fluence of 1×10^{23} n/cm² in X018B).*

The L-15 capsules (Mark-A19 type) previously irradiated in S/A X039 to 0.94×10^{22} n/cm² total fluence have been assigned a new S/A, X080, for further irradiation to a target total fluence of 5.0×10^{22} n/cm².

The MT-4 capsule (Mark-A19 type) previously irradiated in S/A XA08 to 6.9×10^{22} n/cm² total fluence has also been assigned to S/A X080 for further irradiation to a target total fluence of 1.5×10^{23} n/cm².

10.3.2 Series L-2 EBR-II Materials Capsules

The remaining four L-2 series capsules have been discharged from EBR-II and shipped to the VNC for destructive post-irradiation examination. The capsules, L2A (Inconel-800), L2C (Type-316L), L2E (Type-347), and L2G (Type-304), are identical in design to the L-2' described earlier.¹ They have accumulated 12,640 MWd in S/A X605 plus 11,706 MWd in S/A X050 which resulted in a total fluence of 9.05×10^{22} n/cm², making them the highest fluence material in this program. These capsules are scheduled for post-irradiation examination during the next quarter.

* Unless otherwise noted, all reported fluences are based on EBR-II physics calculations.

Figure 10-2. Temperature Dependence of Void Diameter

10.3.3 Series L-16 Materials Capsules

10.3.3.1 L-16A Capsules

The design of these capsules has been completed and fabrication initiated on 19 capsules. These capsules will utilize three basic types of temperature monitors: (1) bimetal thermometer, (2) low melting alloys, and (3) SiC.

Two bimetallic thermometers of the design shown in Figure 10-3 were tested. Calibration showed that maximum temperatures could be determined with these thermometers to within $\pm 1\%$ (not including irradiation effects which might alter the expansion characteristics and/or specimen dimensions). Gap measurements between the tubing and follower were made microscopically and with feeler gages. The feeler gages were notched on one end to allow them to "collar" around the rod to prevent cocking of the follower. Values using both techniques agreed within ± 0.0002 inch. Since the feeler-gage technique is more readily adapted for in-cell use, only values from this method are reported.

A preliminary run was performed on one thermometer by taking it directly to 1300°F (the thermometers were heated in a furnace to within $\pm 2^\circ\text{F}$ accuracy and uniformity). After approximately 15 minutes at temperature, the thermometer was removed, cooled, and gap measurements made. The follower was then reset (pushed back flush against the primary tubing), and the

thermometer reheated to 1300°F. Measurements were again performed and compared with the initial value. Table 10-2 illustrates the excellent agreement attained (designated runs 1 and 2).

The follower was again reset to a zero gap and the thermometer plus a second thermometer were heated together in incremental steps. The thermometers were checked at temperatures of 914, 990, 1196, and 1355°F. The follower on the first thermometer was reset to a zero gap after measurements at each of the above temperatures. The follower on the second thermometer remained in position after each of the incremental measurements. As indicated in Table 10-2 (runs 3, 4, 5, and 6), the results from the two thermometers were essentially the same. Figure 10-4 shows the apparently linear relationship between temperature and expansion gap.

10.3.3.2 L-16B Capsules

Fabrication of these capsules is complete and the 19 L-16B capsules were sent to EBR-II, in early June. The design of these capsules is shown in Figure 10-5. The capsule identification and operation characteristics are shown in Table 10-3. The individual capsules are broken down into eight general regions designated A, B, C, D₁, D₂, D₃, D₄, and E. The specimens for each region are listed in Table 10-3. Regions D₁ through D₄ are the primary specimens in these capsules and will be irradiated at the

Table 10-2
BIMETALLIC THERMOMETER GAP MEASUREMENTS

Run No.	Thermometer	Measured* Temperature (°F)	Gap (in.)	Comments
1	1	1305	0.0387	Start gap at zero
2	1	1305	0.0387	Reset gap to zero
3	1	914	0.0247	Reset gap to zero
3	2	914	0.0250	Start gap at zero
4	1	990	0.0272	Reset gap to zero
4	2	990	0.0277	Accumulated
5	1	1196	0.0347	Reset gap to zero
5	2	1196	0.0347	Accumulated
6	1	1355	0.0407	Reset gap to zero
6	2	1355	0.0407	Accumulated

*Measured using chromel-alumel thermocouple

Figure 10-3. Bimetal Thermometer of the Type to be Used in the L-16A Capsules

Table 10-3
SPECIMEN IDENTIFICATION FOR THE L16B CAPSULES

**Specimen Identification^(a) for Various
Regions^(b) in Capsules**

Capsule Identity	A	B^(c)	C	D₁	D₂	D₃	D₄	E
LP2A	6AA	SiC	4AK	4AA	41A	42A	43A	6AI
LP2B	62A	Alloy	42I	63A	4CB	6CA	65A	62I
LP2C	4AB	Alloy	43F	6AA	61A	62AA	63A	63F
LP2D	42A	SiC	43G	4AB	42X	62BB	6AB	63G
LP2E	6AB	Alloy	4AM	4AC	41B	42Y	43B	6AJ
LP2F	62B	SiC	42A	63I	4CC	6CB	65B	62JJ
LP2G	4AA	Alloy	43H	6AC	61B	62CC	63B	63H
LP2H	42B	SiC	43I	4AD	42W	62DD	6AD	63I
LP2I	6AC	SiC	4AS	4AE	41C	42V	43C	6AK
LP2J	62C	Alloy	42J	63H	4CG	6CG	65C	62KK
LP2K	4AC	SiC	43J	6AN	61K	62F	63C	63J
LP2L(d)	42C	SiC	43K	7AI	1AI	0AI	45I	63K
LP2M	63A	Alloy	4AL	4AF	41J	42K	43D	6AS
LP2N	43A	Alloy	41A	63D	4CF	6CF	65D	61G
LP2O	4CA	SiC	42B	6AF	61D	62E	63D	62NN
LP2P	6AD	Alloy	4AR	4AG	41E	42R	43E	6AM
LP2Q	62D	SiC	42D	63F	4CE	6CE	65E	62MM
LP2R	4AD	Alloy	43L	6AL	61F	62GG	63E	63L
LP2S	42D	Alloy	43M	4AH	42L	62LL	6AH	63M

(a) Specimens are identified as follows:

First character designates alloy

6 = Type-316H; 4 = Type-304H; 1 = Type-321; 7 = Type-347; 0 = Incoloy-800

Second character designates tube condition

A = annealed; 1 = 10% cold work; 2 = 20% cold work; 3 = 30% cold work;

5 = 50% cold work; C = carbide agglomerated; G = Garofalo treatment.

Third character designates specimen letter A, B, C, etc.

(b) Regions are those shown in Figure 10-3.

(c) Region B will contain the temperature sentinels listed in this column.

Alloy refers to: (1) Ag Sb, (2) Al Cu, and (3) Al Si alloys.

(d) Specimen 7AI tubing is Type-347, Heat No. 99973; Specimen 1AI tubing is Type-321, Heat No. 310 3; Specimen 0AI tubing is Incoloy-800, Heat No. 4167.

Figure 10-4. Calibration Curve for Bimetal Thermometer

temperatures shown in Table 10-4. These temperatures are based on detailed two-dimensional calculations using the TIGERV heat transfer code. Calculated temperature profiles are shown in Figure 10-6 for a row 4 position at 62.5 MW. The local hot spots shown between the specimens in regions D₁ to D₄ are due to excess material required to join the individual tubular specimens.

10.3.3.3 L-16C Capsules

The design of these capsules has been initiated. Preliminary heat transfer analysis has shown that to vary the irradiation temperature over the desired temperature range for a given material will require two capsules, one operating in the high temperature range 1000 to 1300°F and the other in the lower temperature range of 800 to 970°F.

10.4 CLADDING ALLOY MECHANICAL PROPERTIES

The objective of this subtask is to evaluate the post-irradiation mechanical properties of available cladding alloy specimens, including pre-cut tensile specimens and tubular specimens from Task L and Task F fuel cladding and capsule tubing. Various uniaxial and biaxial techniques are being used to perform the required mechanical evaluation.

10.4.1 Biaxial Creep Program

The biaxial creep facility described earlier³ was designed and fabricated to provide facilities to creep test

irradiated tubing materials at elevated temperatures to gain creep rate data on highly-irradiated tubular specimens. The initial control specimens have accumulated 3000 hours at 1300°F and tests have also begun on the first lot of irradiated specimens which have accumulated over 2000 hours.

Present indications based on the most recent measurements are as follows:

1. Specimens undergoing testing at low stress levels (6000 psi) and a temperature of 1300°F are responding in a manner comparable to unirradiated material at this stress level and temperature; the values for total strain fall in the range only slightly below the values predicted for unirradiated material under these conditions.
2. Specimens stressed to higher values (8000 psi), also tested at 1300°F, have total strain values approximately one-half the values predicted for unirradiated material, or in other words, under these particular test conditions, irradiated specimens creep less than unirradiated specimens. The tests are continuing. A total of 27 tests on material irradiated at higher fluence levels ($> 3 \times 10^{22}$ n/cm²) are scheduled to begin in August. These specimens have been fabricated from F2 and L2 capsule material removed from EBR-II subassemblies XG05 and XG06 after exposures of 12,640 and 9317 MWd, respectively, in EBR-II. The response of these

Table 10-4
IDENTIFICATION AND OPERATION CHARACTERISTICS FOR L16B CAPSULES

<u>Capsule Identity</u>	<u>Drawing Identity(d)</u>	<u>Operating Temperature</u>		<u>Pin Location(a) in Subassembly</u>	<u>Type of Flux Wire</u>
		<u>°C</u>	<u>°F</u>		
LP2A(b)	286E240-G4	450	842	15(0.97)	Cu
LP2B(b)	286E240-G4	450	842	16(0.96)	Ti
LP2C(b)	286E240-G4	450	842	18(0.97)	Ni
LP2D(c)	286E240-G4	450	842	19(0.95)	Fe
LP2E(b)	286E240-G3	525	977	11(0.98)	Fe
LP2F(b)	286E240-G3	525	977	12(0.97)	Al-Co
LP2G(b)	286E240-G3	525	977	14(0.99)	Al-Co
LP2H(c)	286E240-G3	525	977	17(0.98)	Ti
LP2I(b)	286E240-G2	600	1112	3(1.01)	Fe
LP2J(b)	286E240-G2	600	1112	6(1.01)	Ti
LP2K(b)	286E240-G2	600	1112	7(0.99)	Cu
LP2L(c)	286E240-G2	600	1112	9(1.01)	Cu
LP2M(c)	286E240-G2	600	1112	10(1.00)	Al-Co
LP2N(c)	286E240-G2	600	1112	13(1.00)	Ni
LP2O(c)	286E240-G2	600	1112	5(1.02)	Al-Co
LP2P(b)	286E240-G1	675	1250	1(1.05)	Fe
LP2Q(b)	286E240-G1	675	1250	2(1.03)	Ni
LP2R(b)	286E240-G1	675	1250	4(1.04)	Ti
LP2S(c)	286E240-G1	675	1250	8(1.03)	Cu

(a) Based on core position 4E2, number in parentheses is pin fission rate normalized to S/A average.

(b) Capsules (A,B,C), (E,F,G), (I,J,K), and (P,Q,R) are four sets of capsules designed for nondestructive examination. These capsules will continue irradiation to maximum fluences of 3×10^{23} n/cm², or until saturation of cladding swelling is established.

(c) Capsules D,H,L,M,N,O, and S are to be destructively examined and replaced by new capsules in later subassembly loading.

(d) See Figure 10-4 for G-1 to G-4 location.

specimens to the selected test conditions will determine the test parameters for an equivalent number of high fluence ($> 7 \times 10^{22}$ n/cm²) F2 and L2 specimens to be prepared from material recently removed from EBR-II X050.

10.5 MATERIALS MANAGEMENT

The objective of this subtask is to provide the best quality of tubing to experimenters on all tasks of the PA-10 programs. Also included is the characterization of this tubing.

10.5.1 Characterization of Specialty Tubing

Evaluation of the long-term mechanical properties behavior of the various lots of cold-worked tubing procured under purchase order 5G430G is continuing. The material consists of Type-304H and Type-316H with 0, 10, 20, 30, and 50% cold work, the figures representing the area reduction impressed during the final cold draw.

The test program has thus far been restricted to stress-rupture testing at 1300°F, with initial hoop stresses

selected to cause failure in 100, 1000, and 4000 hours. Data collection during the current report period, however, has been slight, the bulk of the remaining specimens being those pressurized for failure in 4000 hours.

Nevertheless, results on the small number of specimens that have failed during this time period continue to complement the earlier data. The trends shown by these experiments indicate that, although light to moderate cold working may serve to enhance short-term stress-rupture strength, maximum strength after long exposures, i.e., times in excess of 4000 hours, can be realized only with annealed material. The stress-rupture strength of severely cold worked material—in these tests, tubing cold reduced more than 30 percent—is inferior at all times to originally annealed material. Rupture ductility of the latter material, however, is far superior to that of the less severely cold worked tubing, which suggests that the heavily cold worked material undergoes rapid recovery at this temperature. Metallographic examination of selected Type-304H

Figure 10-5. Capsule Assembly (L16B)

Figure 10-6. L16B Specimen Temperature Profiles EBR-II Row 4 Position 62.5 MW

specimens after failure has revealed evidence of recrystallization in the originally 50% cold worked tubing.

Other microstructural modifications in the Type-304H material include sigma phase formation in all four lots of tubing and severe grain boundary cavitation in the two lesser cold worked structures.

To further evaluate the creep behavior of these materials, 1300°F constant stress creep-rupture tests have also been initiated. The pre-pressurized test technique being used was described earlier³ and is similar to that being used in the biaxial tests on irradiated capsule tubes described in subsection 10.4.1.

The initial stress levels chosen for the two alloys, 304H and 316H, are 9000 psi and 13,000 psi, respectively. The specimens with the highest percentages of cold work (30 and 50%) have, in the case of both alloys, shown an accelerated creep rate and abbreviated rupture life by comparison with annealed control specimens (Figure 10-7).

The specimens with lower percentages of cold work (< 30%) exhibited less creep than the annealed material during the earlier stages of testing—the 10%-cold-worked specimens having the lowest creep rate of the series. However, results after 1200 hours testing indicate an increase in strain rate for all the cold-worked material in contrast to the annealed material which is continuing to exhibit a constant strain rate. Because further data are required before any definite conclusions are drawn, these tests are continuing.

Creep specimens fabricated from cold worked Types-304 and -316 material are scheduled for further tests at a lower stress level to confirm and corroborate present findings. These specimens are currently being pressurized. In addition, specimens of annealed, fine-grained material, annealed-and-carbide-agglomerated material, and material stabilized after varying degrees of cold work, are also being pressurized to compare the creep behavior of these microstructures to those presently under test.

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11. TASK N—CORE PHYSICS ANALYSIS

11.1 GENERAL

The purpose of Task N is to contribute to the development of improved physics computational tools for the LMFBR program. To accomplish this work, analytical methods and computer codes are specified and written, and evaluations are performed on the basic nuclear data. Progressively improved methods and data are used in analytical studies to determine their effects on LMFBR neutronics, design, fuel costs, and the reactivity aspects of reactor safety. The results of these studies, together with results from associated reactor design and development activities, provide direction for further development of physics analytical methods and form a basis for establishing current design and safety criteria.

11.2 COOPERATIVE EVALUATION

The topical report which describes the evaluation of Pu-239 cross sections in the unresolved resonance range and the generation of statistical parameters for ENDF/B, Version II¹, has been distributed.

11.3 COMPILATION AND DISTRIBUTION OF EVALUATED DATA

Twenty-nine-group and sixty-group generalized cross section files for rapid cross section adjusting have been prepared for LMFBR materials from the data on ENDF/B, Version II, as distributed by Brookhaven National Laboratory on tapes 991, 992, and 997. All of the materials are category 1, as specified by the Cross Section Evaluation Working Group, except for Fe, Cr, Ni, Zr, and Pu-241, which are in the process of being modified. The unresolved resonance parameters for Pu-239 are based on the latest ENDF/B procedures.¹ A lethargy increment of 0.5 was used for each group of the 29-group file, whereas the lethargy increments in the 60-group file varied from 0.1 in the important lethargy ranges to 0.5 in the less important lethargy ranges, corresponding to the GE standard group structure. These two data files are being used in an evaluation of the rapid cross section adjusting approach, particularly in a study of the effects of group structure on the accuracy of the approach (see subsection 11.5). The

Figure 10-7. Biaxial Creep Curves for Specialty Tubing

29-group set will also be used for standard benchmark critical calculations as part of the data-testing activities of the Cross Section Evaluation Working Group. These benchmark calculations will be performed when category 1 data are available for structural materials for ENDF/B, Version II.

Data for U-238, Pu-239, and Pu-240 which is in the ENDF/B format and basically in agreement with values of the GE cross section file, BAS04, were also used to prepare a generalized 60-group file, GMUG1. This file, as well as that from ENDF/B, Version II, will be used during the preliminary phase of the analysis of ZPPR critical experiments (see subsection 11.4). Neutronics calculations for ZPR-3 Assembly 48 were carried out to compare the GMUG1 and BAS04 cross sections. The principal difference found between the two sets was generally larger elastic removal cross sections in GMUG1, leading to a softer spectrum and a lower k_{eff} value by 1%. These results are presently being analyzed to determine the effect of the rapid cross section adjustment procedure on the GMUG1 elastic removal values.

11.4 SUPPLEMENTARY ANALYTICAL INTERPRETATION OF INTEGRAL DATA

Analysis of ZPPR Assembly 2,² the first demonstration plant benchmark critical experiment, has been started. This analysis is aimed at using integral

measurements to evaluate the accuracy of physics computational methods presently in use for LMFBR design, and will be performed in a series of steps. The first step is the determination of uncertainty ranges in basic physics quantities measured in ZPPR, such as fissile loading and reaction rate ratios, by comparison of detailed calculations with experiment. The second step is to qualitatively determine how these uncertainties affect power reactor design. The third step is an extrapolation of the uncertainties in the basic quantities to uncertainties in design quantities. This third step is the most difficult to perform and subject to arbitrary considerations. For the most part, the extrapolations will be from static conditions through time varying (fuel cycle) conditions. Data adjustment will be required in some cases in order to propagate the uncertainties in basic data.

An example of this procedure is shown in Table 11-1, which lists the steps required to estimate the accuracy of the design quantity Initial Enrichments (plutonium enrichments in clean fuel). The principal design condition for determining initial enrichments is that the reactor must be critical during all times of operation, consistent with an average burnup target for the fuel, a limit to the power peaking and available control worth and control rod motion. The measured quantities from ZPPR which will first be evaluated will be fuel loading, control worth, and

Table 11-1
EXAMPLE FOR DETERMINING ACCURACY RANGES OF DESIGN ENRICHMENTS

Design Quantity:	Pu enrichments of clean fuel
Principal Condition:	$k_{\text{eff}} \geq 1.0$, consistent with burnup, power peaking, and control requirements
Basic Physics Quantities with Uncertainties determined from ZPPR measurements:	Initial enrichment, control worth, fission and capture ratios in fuel isotopes
Data Adjustments for Uncertainty Ranges:	Design k_{eff} , σ_f , σ_c for U and Pu isotopes, volume fraction of control material
Uncertainty Ranges of Pu Enrichments from Fuel Cycle Calculations.	

the reaction ratios in fuel isotopes. Following this, adjustments in the design k_{eff} , fission and capture cross sections for uranium and plutonium isotopes, and the volume fraction of control material will have to be made to conform to the uncertainties in the basic quantities. Finally, fuel cycle calculations, based on these adjustments, will be performed to determine uncertainties in initial plutonium enrichments. Consideration will be given to the effects of proof-test criticals and general knowledge of integral results in arriving at the final uncertainty ranges.

A program of analysis is also being formulated which is directed at applying basic measurements made with SEFOR³ to general methods development. Comparisons of measurements and computations performed under the SEFOR research and development program indicate some significant differences which could be attributed to computational errors associated with the neutron spectrum and nickel reflectors. While such differences do not seriously affect the SEFOR experimental program and measurement of the Doppler coefficient, they do provide information useful to general computational methods development.

Several characteristics of SEFOR, such as the axial fuel gap, tend to complicate physics analyses of the measurements. However, the rod structure and composition of SEFOR provide a closer approximation to power reactors than is possible with existing critical facilities. This analysis will be concentrated on material worth computations, many of which are 15 to 20% higher than the measurements; fission rate traverses, which are not predicted well in the corners of the core; and the sensitivity of computed results to cross section uncertainties.

11.5 IMPROVEMENTS OF EXISTING METHODS FOR GENERATING GROUP CONSTANTS

Additional modifications to the ENDRUN-2 code to conform to recent format changes in Version II of ENDF/B are being checked out. The principal change in the ENDF/B data format is the renumbering of the File 3 inelastic excited states from 5-14 and 51-80, to 51-90. The continuum is also renumbered from 15 to 91.

The user's manual for the ENDRUN-1 code⁴ has been distributed.

Additional checkout problems with the TDOWN-1 code have been completed and shakedown of the code is in progress through the processing of nuclear data for specific applications. Incorrect results were found for successive calculations in which the Bell rational approximation for resonance heterogeneity effects was used. This error has been corrected. The initial checkout of TDOWN-2, in which

zero- and one-dimensional flux solutions and iterations on the elastic removal cross section may be performed, is under way.

Computations are being carried out to study the sensitivity of the rapid cross section adjusting approach to the number and structure of the groups used. The generalized multi-group files generated with the ENDRUN code in 29 and 60 groups using ENDF/B, Version II, data, described in subsection 11.3, were used in this study. The TDOWN code was used to generate effective group constants for the problems considered, and care was taken to ensure that the processing of the data for the two files was consistent except for the group structure.

The effect of group structure was investigated by comparing the results of one-dimensional neutronics calculations for ZPR-3 Assembly 48, using the two different files. These results are given in Table 11-2. The results indicate that the 29-group structure is generally adequate for the calculation of the effective multiplication constant, the core and reactor breeding ratio and the material balance calculations. However, there is a significant difference between the 29-group and the 60-group calculations of the Doppler coefficient. About one-half of this difference can be attributed to the softer spectrum which is calculated in 60 groups. The remaining difference, pending additional investigation and depending upon the emphasis which is placed on the Doppler calculation, may indicate a need for a fine group structure (a lethargy increment of less than 0.5) in the energy range where the Doppler effect is important.

The softer spectrum at low energies, as calculated by the 6-group file, is attributed to a generally larger degree of elastic down scattering in the energy range below 200 keV (i.e., in the energy range where the lethargy increments of the 60-group file varies from about one-fifth to one-half of the lethargy increments of the 29-group file). For example, the difference in the carbon down scattering matrix terms between 29 groups and 60 groups condensed to 29 groups is about 15% in this energy range. This difference may be primarily due to the noniterative approach to the elastic removal cross section correction in Version I of the TDOWN code. This possibility will be checked in greater detail with the iterative approach included in Version 2 of TDOWN.

11.6 IMPROVEMENT OF EXISTING TRANSPORT THEORY CODES

Conversion of the two-dimensional, diffusion/transport theory code, DOT2DB, to the standard

Table 11-2
RESULTS OF RAPID CROSS SECTION ADJUSTING CALCULATIONS USING
29- AND 60-GROUPS FOR ZPR-3, ASSEMBLY 48

<u>Parameter</u>	<u>60-Groups</u>	<u>29-Groups</u>
k _{eff}	0.9625	0.9647
Core Conversion Ratio	0.5717	0.5730
Core Breeding Ratio	0.5593	0.5606
Blanket Breeding Ratio	1.0862	1.1038
Total Breeding Ratio	1.6455	1.6644
Doppler Coefficient (-T dk/dT)	0.0072	0.0063
Mean Fission Energy (keV)	226	219
Fissions Below 5.5 keV (%)	12.42	11.44
<i>Central Core Material Balance (Normalized)</i>		
U-238 Fissions	2.040-5	2.037-5
Pu-239 Fissions	1.383-4	1.390-4
U-238 Captures	9.623-5	9.624-5
Pu-239 Captures	3.481-5	3.356-5
Pu-240 Captures	2.151-6	2.135-5
Cr Captures	1.005-6	9.582-7
Fe Captures	7.539-6	6.821-6
Ni Captures	1.362-6	1.364-6
Na Captures	5.701-7	5.416-7

interfaces specified by the *de facto* Committee on Computer Code Coordination was completed. Copies of the modified code, the DOT2DB user's manual,⁵ and a description of the modifications were sent, upon request, to D. Butler, of Argonne National Laboratory, D. Vondy, of Oak Ridge National Laboratory, and B. Carmichael, of Los Alamos Scientific Laboratory.

Copies of the other sections of coding which have been converted to the standard interfaces have been requested from the organizations involved. The modified versions of ANISN and DAC have been received and are being converted to the GE-635 computer. Problems encountered in converting these sections of coding and

eventually attempting to run them together in a linked problem are being documented for later consideration by the *de facto* Committee on Computer Code Coordination.

11.7 SENSITIVITY STUDIES

The study of the effects of uncertainties in nuclear data on fast reactor design and operation, completed earlier, was reported at the 2nd International Conference on Nuclear Data for Reactors, Helsinki, Finland.⁶ This Conference proved to be interesting and informative. However, in spite of a multitude of new differential data measurements reported, the amount of progress indicated toward attaining more precise data for reactor design was disappointing and less than expected.

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12. TASK P—FUEL ELEMENT DESIGN ANALYSIS

12.1 GENERAL

The objective of Task P is the development of an integral fuel performance and lifetime model. This modeling and analysis effort relates to other portions of the FCR Development Program in the following manner:

1. Guidance is provided for fuel and cladding irradiations.
2. Assistance is provided for the analysis and interpretation of experimental results from fuel and cladding irradiations.
3. A performance and failure model is being developed for use in parametric design studies of the sodium cooled breeder reactor.

12.2 EXPERIMENTAL DATA ANALYSIS

Task P is continuing to provide assistance for improvements in data collection as needed for analysis and interpretation of experimental results from fuel and cladding irradiations.

12.2.1 Microsurveying

During this reporting period the preliminary design and component specifications for a "microsurveying" instrument reported in the 34th Quarterly were finalized.

The unit will measure the local longitudinal strains of Task F fuel pins and Task L cladding specimens with the aid of scribed fiducial marks. It will make absolute

measurements over the entire scribed length of the fuel pin relative to a zero datum reference position and it will make more accurate measurements over gage lengths up to 2 inches.

For the absolute measurements, a surface profile sensor which detects the scribe marks is translated along the length of the specimen. The corresponding travel is indicated by an electronic digital travel wheel which is attached to the sensor support and in contact with a machined surface on the stationary bed. The accuracy of this system is better than ± 0.001 inch in 20 inches.

For the detailed measurements the same sensor is driven along the fuel pin by an electric motor drive system engaged to an electronic digital micrometer (2 inch travel). The output is recorded on an X-Y plotter. This system is accurate to ± 15 micro-inches, allowing more detailed measurements over small gage lengths.

12.2.2 Profilometry

The process of automating fuel pin profilometer analysis outlined in subsection 12.2.2 of the 33rd Quarterly Report was initiated. A magnetic tape recording of a profilometer trace was made and successfully reproduced. It is currently intended that the profilometry traces be recorded on magnetic tape, then interfaced with the computer through a multiplexing data acquisition system. This will allow more detailed sampling of profile traces than the present time-consuming manual methods. As a result of automating profilometry analysis all errors due to manual manipulation of data would be eliminated and a permanent record of analog information could be stored on magnetic tapes.

Implementation of this system will facilitate correlation of experimental results to present modeling. With this reporting, the development of profilometry data taking is being terminated as part of Task P and further development of this system will continue under Task F.

12.3 PERFORMANCE AND FAILURE MODE STUDIES

12.3.1 A Model for the Prediction of Local Fuel Pin Diametral Increase at First Start Up

In the previous Quarterly Report excellent agreement was reported between measured diametral increases of a group of irradiated mixed-oxide fuel pins and a model predicting initial fuel-cladding thermal expansion interference (START code) plus subsequent cladding swelling. In this reporting period, the model has been refined to calculate local diametral increase rather than the mean diameter change. On this basis local cladding strain

can be predicted if local cladding thickness and strength property values are known as a function of position. A simple way to treat the problem of local cladding strain is to assume a constant cladding wall thickness and a strain hardening law of the form

$$\epsilon = k\sigma^4 \quad (1)$$

where

ϵ = cladding strain,

σ = cladding stress,

k = constant.

Force equilibrium between the cladding hoop stress and the hydrostatic pressure in the 100% dense plastic core of fuel gives

$$\sigma t = P_H R_p \quad (2)$$

where

σ = hoop stress in cladding,

t = thickness of cladding,

P_H = hydrostatic pressure in plastic core of fuel,

R_p = radius of 100% dense plastic core.

The two equations may be solved for the local cladding hoop strain in terms of the mean cladding hoop strain, $\bar{\epsilon}$, and the mean value of R_p^4 , giving

$$\epsilon/\bar{\epsilon} = R_p^4/\bar{R}_p^4 \quad (3)$$

The value of R_p can be obtained from the START code. This refined version of the START code has been incorporated in the BEHAVE fuel pin performance computer code. The results of such calculation using the refined START code are shown in Figure 12-1 for pin F2H. Good agreement is found in this case between local diametral increase due to startup plus cladding swelling and the actual diameter change from pin profilometry. Departures from the predicted shape of the curve, noted in some pins, are believed to be caused by variations in cladding wall thickness and the effect of temperature on cladding yield strength.

12.3.2 Simplified Model of Fuel-Cladding Mechanical Interaction at Steady-State

This model is concerned with predicting the amount of cladding deformation which will occur under conditions in which the fuel swelling rate is greater than the cladding swelling rate. Under such conditions a stress is set up at the

Figure 12-1. Behave Model Predictions Versus Experimental Results

fuel-cladding interface tending to deform cladding outward and/or the fuel inward relative to their positions in the unstressed state.

Considerable simplification is afforded both mathematically and conceptually if it is realized that over most operating conditions, either the fuel or the cladding will constitute a clearly weaker link in the fuel-cladding structure. The fuel-cladding ensemble will therefore change its dimensions at a rate determined almost exclusively by the stronger of the two members. The relative creep strength of the fuel and cladding may be determined by calculating the radial stress at the fuel-cladding interface necessary to deform the fuel inward or the cladding outward at some rate relative to the rate at which they would change their dimensions in an unstressed state. That member requiring the lesser radial stress to produce a given displacement rate is the weaker of the two and, for the purposes of this model, will be considered to be infinitely weak. Results of such a set of calculations are shown in Figure 12-2; the basis of the model is discussed below.

The model of fuel creep strength employs the following assumptions:

1. The portion of fuel carrying the load is continuous (i.e., uncracked).
2. A constant isotropic fuel swelling rate prevails throughout the stressed portion of the fuel.
3. Porosity in the stressed region is unvarying with radius.
4. A zero-pressure boundary condition exists at the 1300°C isotherm.

The latter assumption is justified by the following explanation:

In the lower range of interfacial stress, the creep strength of the fuel is dominated by fission-induced creep. The creep strain rate is assumed to be of the form,

$$\dot{\epsilon} = A f \sigma, \quad (4)$$

where

$\dot{\epsilon}$ = creep strain rate,

f = fission rate,

σ = stress,

A = a proportionality constant.

Above a temperature of about 1300°C, the predominant creep mechanism becomes thermal creep. The creep strength in the thermal creep regime falls off exponentially with increasing temperature (stress is halved, roughly, with every mil of decreasing radius above the radius of the 1300°C isotherm). Thus the fuel above the

temperature associated with the transition from fission-induced creep to thermal creep, can be considered with little error, to be completely inviscid. So long as voidage exists in the region above the 1300°C isotherm for the fuel to swell into, the inviscid fuel in this region can be considered to exert no pressure on the fuel surrounding it. All high-linear-power mixed-oxide-fuel pins examined to date have been found to contain a central void, justifying this assumption up to reasonably high burnups.

The equations describing the above physical problem are mathematically identical to the equations describing stresses and displacements in an elastic thick-walled cylinder, permitting a simple closed form solution. The results of such a calculation are shown in the two shaded areas in Figure 12-2, for fuel at 16 kW/ft and at 6 kW/ft. The shaded area reflects the uncertainty in fission-induced creep rates of at least 1 decade. Values for fission-induced creep rates taken from Duncombe,¹ have been referred to as "hard" and "soft" fission-induced creep.

At 16 kW/ft, the radius of the transition temperature for fission-induced creep is well out toward the o.d. of the fuel pellet, and the load-bearing area is a thin ring of fuel. As linear power is reduced, the thickness of the load-bearing fuel is increased as the transition temperature isotherm retreats toward the center. Below some linear power, the center temperature is below 1300°C and the entire fuel body acts totally under the influence of fission-induced creep. The solid body is not incompressible, owing to the presence of porosity into which the material may deform but is much more deformation resistant than the thin fuel ring bearing the load at high linear power.

At high radial stresses, the fuel deformation mode changes from fission-induced creep to thermal creep and the displacement rate versus stress curve turns upward (see Figure 12-2). This occurs, however, at higher displacement rates than have been observed in experimental fuel pins under steady-state conditions and hence this thermal-creep-controlled regime need not be further discussed here.

The cladding displacement rate versus stress curve (the dashed lines in Figure 12-2) is determined by three deformation mechanisms:

1. Neutron-flux-induced creep at relatively low stress,
2. Thermal creep at intermediate stresses,
3. Yielding at high stresses.

The magnitude of flux-induced creep is uncertain at present, since the data that have been evaluated are for fairly low fluence irradiations. The other two quantities can be regarded as well characterized.

Figure 12-2. Radial Displacement Rate vs Radial Stress at Fuel-Clad Interface

With the fuel and cladding deformation rate curves plotted on the same graph, the relative strength of the two structural members can be compared easily. The linear swelling rate of low-temperature fuel is estimated to be on the order of 10^{-6} in./in./h. The radial stress to deform the 16 kW/ft fuel at this displacement rate is 1.5 to 2.5 decades below the stress to deform the cladding by an equivalent amount. Thus it appears that for high-linear-power fuel operating under the conditions denoted in Figure 12-2 the cladding is clearly stronger than the fuel, despite the considerable uncertainty in material properties.

In the case of the 6 kW/ft fuel and a displacement rate of 10^{-6} in./in./h, the fuel and cladding strengths seem to be evenly balanced. Here, better data on fission-induced fuel creep are required before a final judgment can be made as to which is stronger.

A computer code subroutine, "SS 1," consistent with the steady-state fuel-cladding interaction model which has just been described has been incorporated into the STEADY portion of the BEHAVE fuel pin performance computer code.

12.3.3 ADVANCED MODEL OF FUEL-CLADDING MECHANICAL INTERACTION AT STEADY STATE

A computer code for steady-state fuel pin stress analysis covering several phenomena not considered in the simplified model has been written and is in the debugging stage. The code "SS 2" performs calculations for an assumed axisymmetric fuel pin at times long enough for elastic strain to have relaxed out.

The new code makes the same physical assumptions as the previously described simplified model but, in addition, considers:

1. The effects of radial cracks (cracks in the r, θ plane) and transverse cracks (cracks in the r, z plane).
2. Axial flow of the inviscid fuel above the fission-creep to thermal-creep transition temperature in the case in which the inviscid fuel has become incompressible through densification.
3. Axial flow or fuel-cladding slippage of the viscous fuel below the fission-creep to thermal-creep transition temperature.
4. Porosity generation in the fuel due to healing of mismatched crack surfaces.

In order to permit solution of the physical problem, the fuel and cladding are divided into a number of rings with specific constant properties. The plane strain assumption is employed within the cladding and the viscous portion of the fuel individually but relative fuel-cladding slippage is allowed by means of a "slip element" interposed

between the fuel and cladding. By adjusting the properties of the slip element, fuel-cladding slippage and/or flow of the viscous fuel relative to the cladding can be simulated.

Either of two boundary conditions may be employed at the transition surface between fission and thermal creep. Where some porosity or a central void remains in the interior of the fuel, it is appropriate to input the gas pressure in the central void or interior porosity as a boundary condition. Where essentially all the porosity has been used up and the inviscid fuel has become incompressible, the rate of volume increase of the inviscid fuel is used as the boundary condition.

The top of the fuel may, at the option of the user, be considered to be locked to the cladding, an "axial restraint" condition, or it may be considered to be free to slide, an "unrestrained" condition.

Output from the code consists of rates of crack and porosity densification, stresses, and displacement rates throughout the fuel pin.

12.4 INTEGRAL PERFORMANCE AND FAILURE MODELING

12.4.1 Behave: Fuel Element Performance Code

Construction of the BEHAVE code to simulate the nominal performance of a fuel element throughout its operating history, including power changes and transients, is continuing. The first operational level was reached this quarter with completion of the integration of the models for startup with fresh fuel, initial fuel restructuring, cladding swelling, and fuel-cladding mechanical interaction. The first use of this code was in the prediction of the performance of a peak power pin in subassembly F9E.

The basic approach and the flexibility of the structure of BEHAVE were discussed in the 34th Quarterly. The progress toward filling out the structure is shown in Figure 12-3 where the subroutines which were used in this problem are designated. A fresh pin is brought to power (regime 1, calculated in START) where mechanical interference due to differential thermal expansion is considered to have the dominant effect on performance. If there is cladding strain, the strains at each axial node and the new pin dimensions are stored in an accumulator file.

Grain growth and central void formation occur during the first few days at power (regime 2, calculated in RESTRU). During the steady-state operation which follows (regime 4, calculated by STEADY), fuel swelling, cladding swelling, and fuel-cladding mechanical interaction, are important influences. The approach in this regime is to divide the time at power into sub-time intervals. Starting

Figure 12-3. Behave Analysis of an F9E Fuel Pin - Subroutines Used

with the state of the pin then in the accumulator file, each effect is calculated in turn, assuming it is independent of the other effects, and the new geometry, etc., are returned to the accumulator file.

Test subassembly F9E is currently being prepared under another task for irradiation in EBR-II. The BEHAVE code was used to predict the performance of one of the F9E peak power pins. The results show that cladding swelling will increase the diameter of the pin ~ 2 mils by 50,000 MWd/Te and ~ 4 mils by 75,000 MWd/Te. Neither cladding strain during startup, nor cladding strain due to fuel swelling up to 75,000 MWd/Te, is indicated.

The F9E subassembly will test performance at high linear power, where the strength of the fuel is low relative to the strength of the cladding. Fuel swelling is expected, therefore, to be accommodated by densification rather than cladding strain, until the available void is closed. At low power the fuel strength is high relative to the cladding as discussed in subsection 12.3.2, and cladding strain is possible. An example of this is shown in Figure 12-4 for a BEHAVE simulation of a pin which is presumed to have operated at 7.1 kW/ft. Except for the pin average linear power of 7.1 kW/ft, the pin is assumed to be operating under F9E conditions. At 50,000 MWd/Te, cladding strain due to fuel swelling is evident in the regions of the pin $\sim < 7$ kW/ft.

Construction of BEHAVE will continue through the next quarter. First, subroutines to account for cladding creep under gas pressure and loss of effective cladding thickness at the outside and inside diameter will be encoded for the STEADY regime simulation. Second, models describing the regimes of power changes during operation will be added. Third, analysis of experimental pin performance using BEHAVE will continue.

12.4.1 Status: Local Deformation and Life Code

As described in the 34th Quarterly Report the purpose of the STATUS code is to:

1. Calculate local cladding strain distributions as a function of reactor operation.
2. Determine residual pin life as a function of reactor operation by comparing strain levels with a failure criterion.

Completed during the reporting period was the portion of STATUS necessary to calculate local strain distributions within cladding caused by differential expansion from thermal gradients and differential cladding swelling. The thermal gradients considered were caused by:

1. Proximity of adjacent pins (circumferential),
2. Position in bundle (bowing),
3. Cladding heat transfer (radial).

STATUS first superimposes the bowing, circumferential, and radial temperature distributions depicted in Figure 12-5 of the 34th Quarterly Report on the average cladding temperature supplied by BEHAVE. Temperature levels are produced at 37 axial positions, and at 12 positions around the cladding on both inside and outside surfaces.

The temperature distributions are used to establish local values of material properties and in calculations of thermal strains and strains due to differential flux-induced swelling of the cladding. The present output of STATUS is the summed components of hoop and axial strain determined at each calculational node (axial circumferential, and radial) at each specified time during operation.

This phase of development will be completed in the near future with the addition of local stress calculations based on an elastic-perfectly plastic stress-strain model for estimating plastic flow and relaxation, and the addition of a failure criterion based on the previously described cumulative strain-to-failure criterion. Latter phases will involve incorporating model improvements, adding other loading mechanisms, and improving mechanical properties inputs.

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NOTE:
OPERATING CONDITIONS ASSUMED THE SAME AS TEST
SUBASSEMBLY F9E, EXCEPT PIN AVERAGE LINEAR
POWER = 7.1 KW/FT

Figure 12-4. Behave Analysis of Low Power Pin

1

2

3

4

5

6

APPENDIX A PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Number 1-100, from July 1959 through June 1970.

The quarterly reports under the title "Sodium-Cooled Reactors Fast Ceramic Reactor Development Program" follow:

GEAP-3888
First Quarterly Report
October–December 1961.

GEAP-3957
Second Quarterly Report
January–March 1962.

GEAP-3981
Third Quarterly Report
April–June 1962.

GEAP-4080
Fourth Quarterly Report
July–September 1962.

GEAP-4158
Fifth Quarterly Report
October–December 1962.

GEAP-4214
Sixth Quarterly Report
January–March 1963.

GEAP-4300
Seventh Quarterly Report
April–June 1963.

GEAP-4382
Eighth Quarterly Report
July–September 1963.

GEAP-4400
Ninth Quarterly Report
October–December 1963.

GEAP-4601
Tenth Quarterly Report
January–March 1964.

GEAP-4640
Eleventh Quarterly Report
April–June 1964.

GEAP-4723
Twelfth Quarterly Report
July–September 1964.

GEAP-4795
Thirteenth Quarterly Report
October 1964–January 1965.

GEAP-4860
Fourteenth Quarterly Report
February–April 1965.

GEAP-4931
Fifteenth Quarterly Report
May–July 1965.

GEAP-4982
Sixteenth Quarterly Report
August–October 1965.

GEAP-5098
Seventeenth Quarterly Report
November 1965–January 1966.

GEAP-5158
Eighteenth Quarterly Report
February–April 1966.

GEAP-5198
Nineteenth Quarterly Report
May–July 1966.

GEAP-5292
Twentieth Quarterly Report
August–October 1966.

GEAP-5433

Twenty-First Quarterly Report
November 1966—January 1967.

GEAP-5491

Twenty-Second Quarterly Report
February—April 1967.

GEAP-5522

Twenty-Third Quarterly Report
May—July 1967.

GEAP-5541

Twenty-Fourth Quarterly Report
August—October 1967.

GEAP-5584

Twenty-Fifth Quarterly Report
November 1967—January 1968.

GEAP-5631

Twenty-Sixth Quarterly Report
February—April 1968.

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May—July 1968.

GEAP-5700

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August—October 1968.

GEAP-5753

Twenty-Ninth Quarterly Report
November 1968—January 1969.

GEAP-10028

Thirtieth Quarterly Report
February—April 1969.

GEAP-10028-31

Thirty-First Quarterly Report
May—July 1969.

GEAP-10028-32

Thirty-Second Quarterly Report
August—October 1969.

GEAP-10028-33

Thirty-Third Quarterly Report
November 1969—January 1970.

GEAP-10028-34

Thirty-Fourth Quarterly Report
February 1970—April 1970

Supplementary Program Letters, Nos. 1-11 from
October 1962 through April 1964.

Reports from G. D. Collins and W. J. Ozeroff on
Assignment to CEA—France.

In addition, the following topical reports have been
issued:

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Fast Oxide Breeder - Reactor Physics,
Part I - Parametric Study of 3000 MWe Reactor Core;
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GEAP-3347

Fast Oxide Breeder - Stress Considerations
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K. M. Horst
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GEAP-3486

Fast Oxide Breeder Project -
Fuel Fabrication

Part I - Plutonium-Uranium Dioxide Preparation
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