**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM**

Thirty-Sixth Quarterly Report

August 1970—October 1970

Approved:

J. H. Field, Principal Project Engineer
Fast Ceramic Reactor Development Program

Approved:

E. L. Zebroski, Manager
Development Engineering

Prepared for the
United States Atomic Energy Commission
Under Contract No AT(04-3) 189
Project Agreement No 10

LEGAL NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

*Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia
Price: \$3.00 per copy*

BREEDER REACTOR DEVELOPMENT OPERATION • GENERAL ELECTRIC COMPANY
SUNNYVALE, CALIFORNIA 94086

GENERAL ELECTRIC

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

leg

DISCLAIMER

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

DISCLAIMER

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

LEGAL NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

1

2

3

4

5

6

7

8

9

TABLE OF CONTENTS

	Page
1. INTRODUCTION	1
2. SUMMARY	2
2.1 Task B—Defected Fuel Testing	2
2.2 Task C-1—Safety Experimentation (Transient Overpower)	2
2.3 Task C-2—Safety Experimentation (Loss of Flow)	2
2.4 Task E—Fuel Materials Properties and Performance Evaluation	2
2.5 Task F—Fast Flux Tests and Support	8
2.6 Task G-6—Core Design Analysis	8
2.7 Task G-10—Kinetic Analysis	8
2.8 Task L—Cladding Development	14
2.9 Task N—Core Physics Analysis	14
2.10 Task P—Fuel Element Design Analysis	14
3. TASK B—DEFECTIVE FUEL BEHAVIOR	16
3.1 General	16
3.2 Defected Fuel Testing	16
3.3 Fission Product Distribution	22
4. TASK C-1—SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)	25
4.1 General	25
4.2 High Burnup Fuel Experiments	25
4.3 Defected or Sodium-Bonded Fuel Experiments	26
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	45
6. TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION	46
6.1 General	46
6.2 Material Property Studies	46
6.3 Center Temperature Limits and Fuel Behavior	46
6.4 Fission Product—Fuel-Cladding Reactions	48
6.5 Microprobe Analysis of Irradiated Fuel	48
6.6 Fuel Sodium Reactions	57
7. TASK F—FAST FLUX TESTS AND SUPPORT	62
7.1 General	62
7.2 Medium Power, Long Term Irradiations	62
7.3 Low Power Swelling Irradiations	89
7.4 High Power Intermediate Burnup Irradiations	89
7.5 Elevated Cladding Temperature Tests	93
7.6 Unencapsulated Fuel Pin Irradiations	93
7.7 Capsule F12B, In-Pile Axial Expansion and Fission Gas Release of Mixed Oxide Fuel	101
7.8 Capsules F12P and F12Q, Effect of Radial Restraint on Fuel Swelling	101
7.9 Characterization of Capsule Fuel Pin Hardware	101
7.10 Task F Quality Assurance	101

TABLE OF CONTENTS (Continued)

	Page
7.11 Neutron Radiography	102
7.12 Automation of Profilometry	102
7.13 Correlation of Fast Flux Fuel Pin Data	102
8. TASK G-6—CORE DESIGN ANALYSIS	103
8.1 General	103
8.2 Fuel Support System	103
8.3 Steady State and Defected Fuel Irradiation Testing Needs	120
8.4 Fuel Element Failure, Coolant Boiling and Operating Safety Limits	124
9. TASK G-10—KINETIC ANALYSIS	125
9.1 General	125
9.2 Model Development	125
10. TASK L—CLADDING DEVELOPMENT	129
10.1 General	129
10.2 Materials Irradiation Performance	129
10.3 Clad Alloy Swelling	133
10.4 Clad Alloy Mechanical Properties	137
10.5 Materials Management	139
11. TASK N—CORE PHYSICS ANALYSIS	142
11.1 General	142
11.2 Compilation and Distribution of Evaluated Data	142
11.3 Supplementary Analytical Interpretation of Integral Data	144
11.4 Improvements of Existing Methods for Generating Group Constants	144
11.5 Improvement of Fuel Cycle and Management Schemes	144
11.6 Computers, Computer Language and Code Center	144
11.7 Sensitivity Studies	145
12. TASK P—FUEL ELEMENT DESIGN ANALYSIS	147
12.1 General	147
12.2 Performance and Failure Mode Studies	147
12.3 Integral Performance and Failure Modeling	147
APPENDIX A — PRIOR REPORTS	154
CONTRIBUTORS	161
DISTRIBUTION	163

LIST OF ILLUSTRATIONS

Figure	Title	Page
3-1	B9C Technical Decision Tree	17
3-2	B9C Sodium Removal Fixture	19
3-3	Appearance of Weld Area of Flanged Defect Nub Electron Beam Welded to a Simulated Clad Tube	21
3-4	Centrifugal Gas De-Entrainer	21
3-5	Rate of Gas Entrainment versus Water Flowrate	22
3-6	Recommended Static Fill Level for Various Water Flowrates	23
3-7	Pressure Drop Through the De-Entrainer	24
4-1	Task C Series IV Inner Capsule Layout, Computer-Modeled Geometry	26
4-2	Task C Series IV Fuel Specimen C4X Epithermal Power Distribution	27
4-3a	Capsule C4X – First Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)	28
4-3b	Capsule C4X – First Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)	28
4-4a	Capsule C4X – Second Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)	29
4-4b	Capsule C4X – Second Calibration Transient Results, Sodium Temperature Histories (Calculated vs Measured)	29
4-5a	Capsule C4X – First Test Transient Results, Sodium Temperature Histories to Equilibrium (Calculated vs. Measured)	30
4-5b	Capsule C4X – First Test Transient Results, Expanded Sodium Temperature Histories (Calculated vs. Measured)	30
4-6a	Capsule C4X – First Test Transient Results, Sodium Temperature Histories to Equilibrium (Calculated vs. Measured)	31
4-6b	Capsule C4X – First Test Transient Results, Expanded Sodium Temperature Histories (Calculated vs. Measured)	31
4-7	First Test Transient Results Calculated Radial Fuel and Cladding Temperature Profiles (Capsule C4X)	32
4-8	Specimen C6B-1 (Sodium-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec	33
4-9	Specimen C6B-2 (Sodium-Filled Powder) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec	34
4-10	Specimen C6B-3 (Gas-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec	35
4-11	Specimen C6A-1 (Sodium-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid Plane, TT1312-485 mw-sec	36
4-12	Specimen C6A-2 (Sodium-Filled Powder) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1312-485 mw-sec	37
4-13	Specimen C6A-3 (Gas-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1312-485 mw-sec	38
4-14	Power Normalization and Distribution, No Filter vs. N ₁ Filter	40

LIST OF ILLUSTRATIONS (Continued)

Figure	Title	Page
4-15	Power Normalization and Distribution, B ₄ C Flux Filter	41
5-1	Rupture Disc Test Fixture	43
5-2	Computed Void Growth and Clad Temperature for Capsule C10X	44
5-3	Computed Void Growth and Clad Temperature for Capsule C10X	44
5-4	Computed Void Growth and Clad Temperature for Capsule C10X	45
6-1	Steady State Creep of (U-20 wt% Pu)O _{1.96} with 4 and 25 Micron Grain Sizes	47
6-2	The Migration of Cesium and Iodine Added to (U-25 wt% Pu)O _{2-x} in a Thermal Gradient	49
6-3	The Migration of Cesium and Tellurium Added to (U-25 wt% Pu) O _{2-x} in a Thermal Gradient	50
6-4	Radial Plutonium Concentration Across Specimen EIJ3 (U-25 wt% Pu)O _{1.94}	51
6-5	Radial Plutonium Concentration Across Specimen EIJ4 (U-25 wt% Pu)O _{1.97}	52
6-6	Radial Plutonium Concentration Across Specimen EIJ8 (U-25 wt% Pu)O _{2.00}	53
6-7	Plutonium Concentrations Over Positive Specimen Current Images of Sample EIJ-3 with an O/M = 1.94	54
6-8	Positive Specimen Current Image and X-Ray Images Showing Plutonium Concentrations in Porous Area of EIJ-3 (1.94 O/M)	55
6-9	X-Ray Images of Crack Observed 500X in the Incoloy 800 Cladding of F2C	56
6-10	X-Ray Images in Comparison to the Photomicrograph of a Metallic Phase Observed in the Fuel of F2C	58
6-11	X-Ray Images in Comparison to the Photomicrograph Showing a "Fuel" Phase that has Formed in Contact with the Clad in Sample F2C	59
6-12	Comparison of Calculated to Measured Oxygen Levels in EIJ-3, EIJ-4, and EIJ-8. Measured Values are Relative	60
7-1	Axial Temperature Profiles — Capsule F2G	64
7-2	Axial Fuel Temperature Profile — Fuel Pin F2G	65
7-3	Neutron Radiographs of Group F2 Fuel Pins Irradiated to ~105,000 MWd/Te (11.75 at. %)	66
7-4	Comparison of Fuel Pin Diameter and Fuel Diameter Determined by Neutron Radiography	67
7-5	Gross and Single Channel Gamma Scans of Pin F2D	69
7-6	Single Channel Gamma Scan of Pin F2D for Cs-134	70
7-7	Gross and Single Channel Gamma Scans of Fuel Pin F2G	71
7-8	Single Channel Gamma Scans of Pin F2G for Cs-137 and Zr-95	72
7-9	Typical Metallic Fission Product Observed in Center Void of Pin F2V	73
7-10	Typical, Non-Metallic, High Gamma Activity Material Deposited on Walls of Center Void on Pin F2V	74
7-11	β-γ Active Fission Product Plug in Center Void of Fuel Pin F2G	75
7-12	Comparison of EBR-II Measurements and Dial Gauge and Profilometer Measurements on F2D, G, V, and X Capsule Tubes	76
7-13	Diameter Changes of Capsule Tubes Irradiated in EBR-II to Different Fluences	79
7-14	Fuel Pins F2G, F2V and F2X after Removal from Capsule Tubes	80
7-15	Typical Appearance of Fueled Region of F2G, -V, and -X Cladding After Alcohol Bath	81

LIST OF ILLUSTRATIONS (Continued)

Figure	Title	Page
7-16	Spiral Profile Traces of Fuel Pins F2G, -V, and -X	82
7-17	Profilometer Traces of Type 347 Clad Fuel Pin F2G Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters	83
7-18	Profilometer Traces of Type 316 Clad Fuel Pin F2V Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters	84
7-19	Profilometer Traces of Type 316 Clad Powder Fuel Pin F2X	85
7-20	Mixed Oxide Fuel Pins Irradiated to 13.2 at.% Burnup (118,000 MWd/Te) in EBR-II	86
7-21	Photographs of Failed Region of Pin F2D	88
7-22	Comparison of Diameter Measurements from Pin F2D with Profilometer Trace of Pin F2G	89
7-23	Dial Gauge Measurement Results for Pins F2G, -V, and -X	90
7-24	Correlation Between Fuel Pin Diameter and Burnup and Fuel Smeared Density	91
7-25	Diametral Change in Stainless Steel Clad Fuel Pins as a Result of Fast Neutron Irradiation in EBR-II	92
7-26	Gross Gamma Activity Profile of Group F3A Pins #4 and #8	94
7-27	Profilometer Traces of Type 304L Clad Group F3A Fuel Pin #8 Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters	95
7-28	Profilometer Traces of Type 316 Clad Group F3A Fuel Pin #4 Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters	96
7-29	Schematic Representation of Effects of Wire Wrap on Pin Ovality	97
7-30	Rate of Fission Gas Release from Group F3A Pins #1, #2, and #7	99
7-31	Transverse Section of Group F3A Pin #7 Taken from Peak Flux Region (Fuel-Clad Interface is also shown)	100
8-1	G.E. 1000 MWe Plant Conceptual Test Assembly	105
8-2	Test Assembly Fuel Pin	106
8-3	Wave Beam Spacer	107
8-4	Wave Beam Spacer Locking Tab Design	108
8-5	CRASIB I Model	112
8-6	Fuel Pin Vibration Predictions	114
8-7	Grid Beam Structural Model Used for Conceptual Design	115
8-8	Wave-Beam Grid Design Spring Constant	116
8-9	Wave-Beam Grid Design Stress/Unit Load	117
8-10	Wave-Beam Grid Design Stress/Unit Deflection	118
8-11	Grid Spacer — Deflection vs. Height to Thickness Ratio with Irradiation Creep as a Parameter	119
8-12	Pin-to-Grid Growth	121
8-13	Pin-to-Grid Swelling Reaction Grid Height (0.35 in.)	122
8-14	Contributions to Axial Gradient	123
9-1	FREADM-2 Calculation Links	126
9-2	Void Pattern as a Function of Time	127
9-3	Computed Voiding Rate Due to Fission Gas Injection	128

LIST OF ILLUSTRATIONS (Continued)

Figure	Title	Page
10-1	Diameter Increases in L-2 Tubular Specimens Due to Clad Alloy Swelling (L2A-3) and Pressurization (L2G-3)	130
10-2	Post Irradiation Appearance of Type 347 Tensile Specimens from L2E3	131
10-3	Apparent Emissivity of SS Tubes Plotted as a Function of Temperature of Inner Tube	132
10-4	Actual and Apparent Emissivity at Different Surface Temperatures	134
10-5	Pin F2H 316 Clad	135
10-6	Pin F2T 347 Clad	136
10-7	Stress-Rupture Properties of 5G430G Tubing: Type 316H, Heat No. M2783, Test Temperature: 1300°F	141
12-1	Status of BEHAVE	148
12-2	Status of STATUS—Local Deformation and Lifetime Analysis Code	150
12-3	Clad Material Properties Data Input to STATUS	151
12-4	STATUS Methods of Separating Components of Strain	152
12-5	Strain Damage Accumulation Diagram	153

LIST OF TABLES

Table	Title	Page
2-1	Task B — Summary of Current Mixed Oxide Capsule Irradiations	3
2-2	Task C — Current Experimental Test Series	5
2-3	Task E — Summary of Current Mixed Oxide Capsule Irradiations	9
2-4	Summary of Task F Fast Flux Testing Program in EBR-II	11
2-5	Summary of Task F Thermal Flux Irradiations	13
2-6	Summary of Task L Materials Irradiated in EBR-II	15
3-1	Fabrication Processes and Inspection Methods Required by Joint Classification System	20
6-1	Contents of Fuel Pins Comprising Thermal Gradient Test of Fuel — Fission Product-Cladding Reactions	57
7-1	Mixed Oxide Fuel Pre-Irradiation Information	63
7-2	Summary of Group F2 Capsule Diameter Measurements	77
7-3	Summary of F2 Fuel Pin Length Changes	78
7-4	Summary of F2 Fuel Length Changes	78
7-5	Summary of Group F2 Fuel Pin Diameter Measurements	78
8-1	Summary Description of Fuel Assembly Geometry	104
8-2	Test Assembly Operating Characteristics in FFTF Core	110
8-3	Load and Deflection Summary for a Corner Rod vs. Creep Magnitude	111
10-1	Clad Swelling Data From Pins F2H and F2T	133
10-2	Creep Test Conditions — Batch No. 2	138
10-3	Interim Creep Test Results — Batch No. 2	139
11-1	Preliminary Results of Benchmark Critical Evaluations	143
11-2	Comparison of Radial Distributions of Cross Sections and Flux	145

7

8

9

10

11

12

13

14

15

HIGHLIGHTS

- Side-by-side fast flux testing of three mixed oxide fuel pins to peak burnups of 117,000 MWd/Te indicates a direct relationship between diameter increase and original fuel smeared density.
Post-irradiation examination of three mixed oxide pins irradiated in EBR-II with smeared densities of 83.9, 86.6 and 92.9% indicated diameter increases of 2.2, 3.9 and 5.6% respectively. However, claddings of the pins also were different, being made of 316, 316, and 347 respectively.
- Clad density measurements on fuel pins irradiated in a fast flux to 65,000 MWd/Te indicate that one-third of the diameter change is due to clad swelling for these conditions.
Post-irradiation density measurements of both 316 and 347 clad fuel pins irradiated to a fast fluence of 3.7×10^{22} ($E > 0.1$ MeV) exhibited a 0.5% $\Delta D/D$ change caused by clad swelling. The remaining 0.7% $\Delta D/D$ is attributed to fuel swelling and/or mechanical fuel clad interactions.
- A comprehensive conceptual design study was completed for a 169-pin wave beam grid spacer assembly.
Design assumptions were defined by test conditions to simulate a 1000 MWe fuel assembly in an FFTF open loop. This study provides a basis for detailed analysis of a grid structure than can be tested in EBR-II.
- Fission gas puncture experiments from fuel pins irradiated in a fast flux to peak burnups of 52,000 MWd/Te suggest that the peak rate of gas flow from a failed fuel pin is a function of defect location.
Irradiated mixed oxide fuel pins were sampled for fission gas in the fission gas plenum and at the bottom of the fuel column. Measured gas release rates from these pins indicate that the irradiated fuel offers significant resistance to gas flow from the plenum to the defect location.
- Results of the first test calculations of integral benchmark problems have identified the need for further improvement in the Evaluated Nuclear Data File/B.
Version II of ENDF/B resulted from a one-year cooperative evaluation effort, coordinated by the National Neutron Cross Section Center. Initial comparisons give computed K_{eff} values as much as 4 percent low for fast assemblies, indicating a need for further revisions.
- A new computational model has been completed, for the calculation of single channel sodium voiding accidents.
Improvements over previous modeling include the ability to simulate experimentally observed phenomena such as axial condensation, multiple bubble formation, and evaporation of the liquid film retained on the fuel cladding.
- Thermal conductivity measurements on mixed oxide fuel indicated no density related conductivity change in the density range investigated.
Mixed oxide samples in the density range of 84 to 95% indicated no decrease in conductivity with decreasing density. This result is contrary to prior results on UO_2 ; the difference is attributed to differences in the size, shape and distribution of porosity.

7

8

9

10

11

12

13

14

15

16

17

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 selected portions of a reliable, high performance fast reactors having nuclear characteristics which provide stable and safe operation.
- Demonstration of low fuel cycle cost potential for such reactors, primarily through achieving high

burnup of ceramic fuels operating at high specific power.

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

This is the thirty-sixth 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 TESTING

A formal review of the engineering design, quality control, and quality assurance activities associated with forced convection capsules was conducted. Results of the inquiry indicate reliability of fabrication and operation of capsule B9D appears attainable with certain changes in design and procedures. As part of the upgraded quality assurance effort for capsule B9D, all joints found in the present design have been reviewed and classified as to consequence of their failure. Four classes of joints were established. A suitable quality assurance/quality control plan is being generated for each class of joint to increase its reliability under irradiation.

Examination of capsule B9C indicated the leak in the fuel failure detection system was caused by the failure of one of the brazed joints on the purge line. Due to the failure of the joint, the capsule sodium purity is now uncertain. An examination is presently in progress to determine the reparability of the capsule.

Due to the observed joint failures in capsule B9C, all fabrication work on capsule B9D was stopped.

Development work related to capsule B9B continued. A projection welding technique for attaching the defect nubs to the cladding is presently being evaluated. Fifteen defect nubs were subjected to 100 thermal cycles (300-1550°F) without leaking. In addition, tests were completed to determine if significant strengthening of the projection welds occurred after exposure to maximum capsule operating temperatures for an extended period of time. No strengthening of the welds was observed for a heat treatment of 1175°F for 30 days. A program to evaluate electron beam welding of the defect nubs to the cladding was also initiated. Testing of a smaller model of the centrifugal gas de-entrainer was initiated. The de-entrainer design was modified to handle flowrates up to 15 gpm. The maximum sodium flowrate in capsule B9B during operation will be 12 gpm. The smaller model showed the same efficiency in rate of gas de-entrainment as the previous larger model. The static sodium fill level for optimum dynamic operation of the capsule for different flowrates was also determined.

Work continued on the hazard analysis for in-core insertion of capsule B10A.

Sectioning and leaching of the inner surfaces of all remaining Miniloop assemblies was completed. Analysis of the leach samples is in progress.

The status of Task B, capsule irradiations, is shown in Table 2-1.

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

Capsule C4X was subjected to a test transient expected to produce cladding failure. The test resulted in ~62% of

the fuel reaching melt temperature, and a maximum cladding temperature of ~1800°F. Based upon instrumentation read-outs and post-transient TREAT neutron radiographs, the cladding did not fail.

Assembly of the C4G and C4H inner capsules has been completed, including remote closure welding. Difficulties in maintaining flow in the remote sodium fill system resulted in apparent heat damage to the upper instrumentation connector on the C4G inner capsule. Replacement of the connector is underway.

Thermal analysis of calibration transients performed on capsule C4X verify the reduced specimen power factor below that predicted by physics calculations. A review of the calculations indicate that the computational treatment of the ephithermal flux must be modified due to the hardened spectrum.

Destructive examination of all specimens from capsules C6A and C6B has been completed, and the final report initiated. Evidence of fuel melting was found in all specimens, with the gas-bonded pellet fuel melting up to ~60% by volume. Profilometry of the C6A specimens indicated up to 2% cladding deformation for the sodium-filled powder fuel specimen. Both pelleted fuel specimens (sodium and gas filled) showed negligible deformation.

A study has been initiated on the possibility of using the Series 6 multi-capsule design for TREAT irradiation of highly-enriched EBR-II fuel specimens.

The status of the current inpile test program for Task C-1 is shown in Table 2-2.

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

New heaters (fuel simulators) are being ordered for a second loss-of-flow test in the Mockup 1A facility. This second test is scheduled for operation in early 1971. Fabrication of C10X continued with scheduled reactor insertion in November. A proof test of the burst diaphragm and an analysis of bubble growth behavior expected in this experiment have been completed. Final C10A capsule specifications have been issued and work continued on updating the THTE program for capsule design. The work plan and the Mark II loop evaluation were completed.

2.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

Compressive creep test data on mixed oxides with small grain size (~4 microns) indicated a reciprocal grain size squared relationship with creep rate ($\dot{\epsilon} \propto 1/(G.S.)^2$). The measured creep rate was linear with stress, ($\dot{\epsilon} \propto \sigma$) over the entire range of applied loads, 1000 - 8000 psi., in contrast with a power dependency ($\dot{\epsilon} \propto \sigma^{4.5}$) observed in larger grain size samples (25 microns) at the higher stresses.

Table 2-1

TASK B — SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

		Design Parameters											
Capsule Designation	Purpose and Major Parameters	Fuel						Cladding Material Stainless Steel Type	Operating Conditions				Status
		Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in.-diam)	Density (Smear) (% TD)	Composition			Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)	
						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	Examination Completed
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	Examination Completed
φ 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 Completed
		0.204	2.25	0.017*	84	25	1.97		GETR			57,000	

GEAP-10028-36

Table 2-1 (Continued)

		Design Parameters							Operating Conditions				
Capsule Designation	Purpose and Major Parameters	Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in.-diam)	Fuel 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 GETR	971	23.4	8,600	
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

GEAP-10028-36

Table 2-2
TASK C – CURRENT EXPERIMENTAL TEST SERIES

		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			Calib- ration	Test	Before Trans.	After Trans.		
				Pellet	Smeared	%UO ₂	%PuO ₂	%U-235	O/M			Yes	No						
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	C	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	C	P	P	

GEAP-10028-36

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	C	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	C	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	C	C	C	P	

Table 2-2 (Continued)

Design Parameters												TREAT							Destruc- tive Exam	Final Report
Capsule	Test Objective	Diameter (inch)	Fuel Length (inch)	Fuel Density (% TD)		Fuel Composition				Cladding Material	Fabri- cation	Pre-Irra- diated		Transient Irradiation		Nondestructive Exam†				
				Pellet	Smeared	%UO ₂	%PuO ₂	%U-235	O/M			Yes	No	Calib- ration	Test	Before Trans.	After Trans.			
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 loaded 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.

Thermal conductivity measurements on mixed oxide samples have been completed. No significant ($\pm 10\%$) effects of the fuel density over the range of 84 to 95% of theoretical density have been detected at temperature above 1000°C . This result is contrary to prior results on UO_2 ; the difference is attributed to differences in the size, shape and distribution of porosity. Mixed oxide samples with plutonium concentration of 10, 20 and 30% do not have significantly different thermal conductivities.

Six stainless steel clad mixed oxide fuel pin samples were heat treated for 100 hours in an out-of-pile thermal gradient (500 - 1250°C) furnace to study fission product migration, combination and reactions with cladding. Cs_2O , CsI and Te (spiked with radioactive Cs , I , and Te) were loaded singly or together at the hot ends of the six pins. Gamma activity measurements on the annealed specimens revealed that cesium concentrated at the cool end of the capsule when initially present as Cs_2O . When combined with iodine, both the Cs and I concentrated at the cool end. In the fuel pin containing Cs_2O and tellurium, however, the cesium appears to have combined with tellurium and both migrated no further than three inches from the cold end which corresponds to a temperature of about 800 - 900°C .

X-ray lattice parameters were determined on a sample of fuel-sodium reaction product extracted from failed specimen F20. The unit cell size was slightly greater than the original value. No second phases were detected.

Microprobe analysis of plutonium gradients near the center void in a second set of EIJ fuel pins (EIJ-3, 1.94 O/M; EIJ-4, 1.97 O/M; EIJ-8, 2.00 O/M) was completed. The data confirmed the results obtained on the first set of EIJ samples which indicated that the location of the molten boundary in mixed oxide fuels may not be at the currently accepted microstructural feature.

Initial results from the microprobe analysis of fuel pin F2C indicate that a grain boundary crack in the Incoloy 800 clad contains molybdenum.

Oxygen gradients as a function of fuel radius were determined directly using the microprobe on samples from EIJ-3, EIJ-4 and EIJ-8 fuel pins.

The current status of the Task E Irradiation Test program is given in Table 2-3.

2.5 TASK F—FAST FLUX TESTS AND SUPPORT

Results obtained from the post irradiation examination of the $>105,000$ MWd/Te Group F2 encapsulated fuel pins (F2D, F2G, F2V, and F2X) obtained to date indicate:

- Peak diameter increases of the capsule tubes ranged from 1.8 to 2.7%.
- Periodic diameter variations of 1 to 1.5 mils were observed over most of the fueled lengths of the

capsules and corresponded to the location of the spacer wire on the fuel pins.

- Maximum diameter increases for pins F2G, -V, and -X were, respectively, 5.6% (14.1 mils), 3.9% (9.7 mils), and 2.2% (5.6 mils).
- Fuel smeared density appears to have a major effect on fuel pin diametral change at high burnup.
- Measurements of the density of clad samples from pins F2H and F2T after irradiation to $\sim 60,000$ MWd/Te indicate that greater than 50% of the diameter change of these pins resulted from fuel-clad interaction.

The post irradiation examination of the unencapsulated Group F3A fuel pins was initiated. Significant results obtained so far indicate:

- The 304L clad pins in general had larger diameter changes than the 316 clad pins. At least part of the difference in diameter can be contributed to differences in the degree of ovality of the fuel pins.
- Diametral changes of the fuel pins were effected by either the wire wrap on the pins or interaction between the wires on individual pins and the sub-assembly hex causing the pins to become oval in cross-section.
- Fission gas punctures in the plenum region and at the bottom of the fuel pins indicate that the insulator pellets and/or fuel column significantly decrease the rate of gas release from a fuel pin.

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 conceptual design of a 169-rod grid spaced fuel assembly for test in an open loop of FFTF was completed. Selected design considerations are discussed.

The Steady State and Defected Fuel Irradiation Testing Needs report is undergoing final review prior to its release.

The fuel element failure/coolant boiling/operating safety limits assessment of information required to design fuel assemblies was completed and is under review.

2.7 TASK G-10—KINETIC ANALYSIS

A version of the VENUS code was obtained and is operational. Steps are underway to convert FREADM to a compatible computer system to facilitate coupling (to VENUS).

FREADM modifications have been made to permit simulation of loop or tank type reactor concepts.

Limited sensitivity studies were performed using the new voiding model. Multiple bubble formation and collapse were calculated as a function of superheat and evaporation/condensation rates.

Table 2-3

TASK E—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

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								
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 reiew
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	Examination complete
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	Examination complete
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	Examination complete
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	Examination complete
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	Examination complete
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	Examination complete
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	Examination complete
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	Examination complete

GEAP-10028-36

Table 2-3 (Continued)

Capsule	Description	Design Parameters							Operating Conditions				Status
		Diameter (inch)	Length (inch)	Fuel Density (% TD)		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temp (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)	
				Pellet	Smeared	PuO ₂ (%)	O/M						
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	Examination complete
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	Awaiting irradiation
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	

Table 2-4
SUMMARY OF TASK F FAST FLUX TESTING PROGRAM IN EBR-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) (inv 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 steam 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 accel- erated 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 _p	At %	MWd/T _p	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_p 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		Description	Design Parameters							Operating Conditions					Status
			Fuel				Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/tt)	Burnup (MWd/Te)				
New No.	Former No.		Diameter (inch)	Fuel Length (inch)	Density (%TD)							Composition			
					Pellet	Smearcd	PuO ₂ (%)	O/M							
F12A	(F5F)	Linear fuel movement versus burnup	0.211	10.0	92.0	87.1	25	1.98	Type-316 stainless steel	GF1R	850	17	44,000	REPORT IN PROCESS	
F12B	—	Axial fuel movement and fission gas release versus burnup	0.211	10.0	92.0	87.1	25	1.97	Type-316 stainless steel	GETR	1200	20	100,000*	IN FABRICATION	
F12C	(F5C-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	(F5D-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	(F5E-1)	Powder, pellet, and annular fuel comparison at high burnup	0.215 (Annular)	16.0	95.0	83.1	25	1.96	Type-316 stainless steel	GFTR	1160	19.5	130,000		
F12P	(F2P)	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	GF1R	1160	15	~60,000	Irradiation	
F12Q	(F2Q)	Effect of radial restraint on fuel swelling (10 mil tubing)	0.229	2.0	92.0	91.0	25	1.98	Type-316 stainless	GF1R	1160	15	~60,000		

*Target

2.8 TASK L—CLADDING DEVELOPMENT

Density measurements on F2T (347 clad) and F2H (316 clad) fuel pin cladding indicate that less than half of the observed $\Delta D/D$ for these pins can be attributed to clad alloy swelling.

Post irradiation creep tests (1300°F) are in progress for Type 304 irradiated to 3.8×10^{22} n/cm² total fluence. Results to date show that irradiation reduces the creep rate substantially at higher stresses (12,000 psi). This hardening is less effective at lower stresses (6,000 psi).

The three L-6 capsules in S/A X018B were discharged from EBR-II after accumulating a total fluence of 8.3×10^{22} n/cm². The four L-15 and MT-4 capsules were reconstituted in S/A-X080 for further irradiation in row 6.

The four L-2 capsules are presently being examined. Dimensional measurements indicate that both gross mechanical induced swelling and alloy swelling occurred in L-2E (type 347) and L2G (type 304) tubular specimens whereas only clad alloy swelling occurred in L-2A (Incoloy-800) and L-2C (type 316L).

Bench tests on a thermal mock-up of the L-16 capsule design were completed. The results indicate an effective surface emissivity of 0.27 to 0.34 for the L16 design.

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

2.9 TASK N—CORE PHYSICS ANALYSIS

Computations of benchmark critical experiments using ENDF/B, Version II, data give slightly lower eigenvalues than Version I data. This new version is the result of a cooperative evaluation, coordinated by the National Neutron Cross Section Center. The eigenvalues for the plutonium fueled assemblies are about 3 percent lower than the measured values, placing a low confidence in the use of ENDF/B data for current design efforts.

Investigation of the self-shielding of U-238 cross sections from 500 to 700 KeV due to the fine structure measured by Kopsch,⁽²⁾ et al, indicates that the effect is minimal.

Cell heterogeneity computations for the inner and outer core regions of ZPPR-2 using S_n transport theory were completed.

The use of REAX⁽³⁾ for calculation of heterogeneity effects of resonance absorption was tested and found to give acceptable accuracy.

The fuel reload option has been improved and the extrapolation technique for determination of end-of-cycle reactivity was tested and confirmed in the fuel cycle code FUMBLE.

Efforts to convert the standard interface sections of coding to the GE-635 have identified specific problems in compilation due to differences in coding practices. Participation in the activities of the *de facto* Committee on Computer Code Coordination is directed at setting standards for LMFBR physics code development.

2.10 TASK P—FUEL ELEMENT DESIGN ANALYSIS

The first version of the STATUS local deformation and lifetime code is near operating status. The subroutines which accumulate the several types of local strain produced by the differential thermal expansion and the differential clad swelling caused by local temperature variations have been debugged. A sample problem using dummy data and lifetime criteria is being run.

The advanced model of fuel-clad mechanical interaction (code SS2) has been debugged, checked, and is ready for inclusion in BEHAVE. Work on the model of interaction during power changes is underway.

Subroutines have been added to the BEHAVE fuel element performance code to fill out the model of steady-state behavior. The integrated effect of startup with fresh fuel, initial fuel restructuring, and the steady state phenomena of clad swelling, loss of effective clad thickness, clad creep due to gas pressure, and clad strain due to fuel swelling (code SS1) can now be analyzed. In this form, BEHAVE is being used under the AEC-PDP program to study fuel behavior in the demonstration plant.

Table 2-6

SUMMARY OF TASK L MATERIALS IRRADIATED IN EBR-II

Series	Row-S/A	Capsules	Capsule Materials	Number of Specimens		Total Fluence			Completion Date**
				Biaxial	Tensile	Target, 10 ²³ nvt	Sept. 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	4 - 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
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*	2.0	0.83	0.69	Sept., 1970
Irradiation in Progress									
L-15	3-X039 X080	4	All five alloys	None	288	0.30	0.09	0.07	1971
MT-4‡	4-XA08 X080	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.

3. TASK B—DEFECTIVE FUEL BEHAVIOR

3.1 GENERAL

The purpose of Task B is to determine analytically and experimentally the manner and consequences of the propagation of fuel defects, the behavior and performance of the fuel 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 such release and on designs to limit the consequences of such release. 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) non-vented designs.

The technologies of defected, vented, and sodium-bonded fuels are closely related. Although directed primarily at studying the behavior of defected fuel, Task B also has 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 terminate this aspect of the work.

3.2 DEFECTED FUEL TESTING

Capsules B9C and B9D are designed to investigate the performance of defected mixed-oxide fuel at prototype LMFBR conditions in flowing sodium.

3.2.1 Forced Convection Capsule B9C

A complete examination of capsule B9C was initiated to determine the source of the leak detected in the fuel failure detection system. Capsule examination showed the source of the leak to be the brazed joint where the failure detection system purge line connects (capsule side) to a series of solenoid valves located in a junction box approximately 7 feet above the top of the capsule. Failure of the brazed joint is attributed to dissolution of an undetected flux pocket in the braze material which permitted leakage through the joint. Water was admitted to the lower portion of the capsule flex hose and junction box containing the braze joint when a seal weld at the junction box failed. The entering water short circuited the main capsule heater leads which burned through the wall of the heater well during the initial insertion of the capsule into the reactor. These failures were previously described in the 35th Quarterly Report.

The failure of the brazed joint constituted a breach of the capsule primary containment. Gross contamination of the capsule sodium by the air in the flex hose may have occurred through this leak prior to its discovery.

In the course of the diagnostic examination, the static pressure in the capsule binary gas system fell from 50 to 15 psig in a period of three weeks. Subsequent examination confirmed that the leak was in the binary gas exhaust line penetration through the secondary containment bulkhead. Detailed examination of this joint indicated several cracks in the weld area. Failure of this joint is thought to be due to excessive stresses induced by repeated flexing of the exhaust line during examination of the capsule.

Because of the observed series of weld and braze failures in the capsule, an extensive examination of the capsule was initiated to determine the feasibility of repairing it. Continuation of this experiment requires its safe operation and the attainment of meaningful test objectives. Safe operation of the capsule requires double containment of the mixed-oxide fuel and sodium along with necessary instrumentation to properly control the experiment.

At the present time, the ability to safely operate the capsule and attain the test objectives is unknown. The physical condition of the capsule after the extensive handling must be ascertained. In addition, the primary test parameter of sodium purity and the condition of the fuel pins have been jeopardized. The sodium purity is now unknown due to the breach in primary containment which allowed oxygen into the capsule. The fuel pins after the extensive handling may have defected prematurely.

The decision tree shown in Figure 3-1 defines an examination which can be conducted to identify the condition of the capsule and determine where repair is required and if it is possible. There are four distinct areas of examination. These are 1) the peripheral systems where all defects have been located to-date, 2) the main capsule body, 3) the fuel pins, and 4) the capsule instrumentation.

The examination of the peripheral systems has been completed. The defects found are described earlier in this report. In addition to the defects, it was determined that the failure detection system lines were plugged. All of these conditions are repairable with varying degrees of effort.

Following a decision to continue with the repair effort, the examination of the main capsule body would be initiated. In order to complete this portion of the examination, the capsule sodium must be removed and analyzed for oxygen and alpha-emitting material. The oxygen analysis in conjunction with the metallographic examination of a representative interior weld would identify the extent or lack of corrosion in the interior of the capsule. The analysis for alpha-emitting material would be an indicator of fuel pin integrity. The presence of

Figure 3-1. B9C Technical Decision Tree

plutonium in the sodium would preclude the possibility of reconnecting the capsule to the cleanup and fill system to recharge the capsule with pure sodium. Finally the integrity of the main capsule body would be determined by helium leak checks of the primary and secondary containments.

The third area of examination would further assess the integrity of the fuel pins. X-rays and/or neutrographs taken of the test section would determine if the defect nubs are in their original position. The shots would be compared with x-rays taken previously at the completion of capsule fabrication.

The fourth area of examination would check capsule instrumentation. At the present time, all thermocouples are operating and have been operating since the initial sodium filling operation. The flowmeter has operated twice previously. However, there is no flow in the capsule at the present time.

A procedure for sodium removal from the capsule has been established. As shown in Figure 3-2, a drain tank is attached to the bottom cold trap of the capsule. The drain tank is evacuated and back filled with argon. An electric arc is remotely actuated to burn a hole through the remaining portion of the fill tube. The argon is bled from the drain tank allowing the sodium to flow from the capsule. The entire fixture will be heated and insulated.

Two sample tubes are attached to the bottom of the drain tank. The sample tubes are attached such that they fill sequentially. They will then contain sodium from the lower and upper portions of the capsule. These sample tubes will be pinched off and the sodium analyzed for plutonium and oxygen. The capsule will be maintained at room temperature and sealed after the sodium removal process until definitive action is prescribed depending on the results of the analysis.

The electric arc assembly being used with the sodium drain tank was initially designed and tested as a possible clad defecting mechanism and reported in the 34th Quarterly Report.

Tests were initiated to determine the feasibility of burning a hole through a tube filled with sodium. Tests were performed on tubes filled with copper and lead. The tests with copper indicated approximately 12 volts and 200 amperes were required to burn a hole through the tube, while approximately 10 volts and 60 amperes were required for the lead. The thermal and electrical conductivities of copper and lead bracket those of sodium. Therefore, the electric arc assembly should be capable of burning a hole through a tube filled with sodium. Mockup tests using sodium will be conducted to establish final voltage and current parameters.

3.2.2 Forced Convection Capsule B9D

Due to the various weld and braze failures observed in the capsule B9C, all fabrication of the B9D experiment has been stopped and the capsule assemblies and parts placed in a safe interim storage container. Because of the similarity between the two experiments, information obtained from the B9C examination will be used to aid in the redesign and upgrading of the reliability of B9D before its fabrication is resumed.

A formal review of the engineering design, quality control, and quality assurance activities associated with forced convection capsules was conducted by a design audit team composed of General Electric personnel not associated with past FCC designs. Results of their findings indicate that fabrication and operation of capsule B9D with a high degree of reliability appears attainable with certain changes in design and procedures. This General Electric Audit has been followed by an AEC audit; the results and conclusions are not yet available.

As part of the upgraded quality assurance effort being placed on capsule B9D, all joints found in the present design have again been reviewed and classified as to:

1. consequence of the failure
2. effect on nuclear safety
3. consequence on experimental objectives
4. further operational capability of the experiment

Using the above criteria, all welds and brazes in the capsule have been grouped under four general classifications based on a single failure analysis. They are as follows:

- Class A-1: Joints whose failure requires capsule removal from reactor with no repair possibility.
- Class A-2: Joints whose failure requires capsule removal from reactor with repair possibility.
- Class A-3: Joints whose failure permits reduced power operation and/or repair in place with subsequent full power operation.
- Class B: Joints whose failure does not affect capsule or reactor operation.

All joints found in the final B9D capsule design will fall into one of the above classes. Each class of joints has specified fabrication processes and inspection methods which each joint must meet. These requirements are given in Table 3-1. A similar classification system for the critical components in the B9D capsule is being developed.

A redesign of the capsule is in process. All heaters have been removed and the space gained allows other components to be arranged in a manner which makes fabrication easier and more reliable. Weld designs are being standardized and consolidated from the thirteen types found in the B9C design to six in B9D.

Additional stress analysis of critical joints and components is being conducted for both steady-state and transient conditions. Results of these calculations may require additional redesign of the capsule.

Figure 3-2. B9C Sodium Removal Fixture

Table 3-1
FABRICATION PROCESSES AND INSPECTION METHODS
REQUIRED BY JOINT CLASSIFICATION SYSTEM

		JOINT CLASSIFICATION			
		<u>A-1</u>	<u>A-2</u>	<u>A-3</u>	<u>B</u>
I.	Made to qualified and documented fabrication procedures	X	X	X	X
II.	Made by operators qualified to make that specific joint	X	X	X	
III.	Visual Inspection*	X	X	X	X
VI.	Radiography capable of providing a volumetric examination* and/or the preparation of pre- and post-welds for metallographic inspection	X			
V.	Radiography for weld penetration measurement	X	X	X	
VI.	Dye Penetrant Inspection*	X	X	X	X
VII.	Helium Leak Checks*	X	X	X	X

*Acceptance-rejection criteria will be used to interpret inspection results

Welding and brazing procedures are receiving extra documentation, along with the procedures for qualifying the various B9D joint designs and the personnel performing the work during fabrication. Each of the above items is a result of the emphasis being placed on quality assurance and quality control in the B9D capsule. Each is part of an overall quality assurance plan which will increase the reliability of B9D and future forced convection capsules.

3.2.3 Forced Convection Capsule B9B

Capsule B9B will be used to investigate mixed-oxide fuel which fails after reaching a burnup of approximately 25,000 MWd/Te.

Development work related to capsule B9B continued. Testing was continued to determine the feasibility of using a projection welding technique for attaching the fuel pin defect nubs to the cladding for capsule B9B. A batch of fifteen defect samples were fabricated with identical parameters of 6 lbs. forging force and 43 watt-sec energy absorption. All specimens passed a test of 100 thermal cycles (300-1550°F) without leaking. Two specimens were burst tested to failure. The failure pressures were 6600 and 8000 psi, respectively. Tests were also performed to determine if weld bond strengthening occurs after exposure to maximum capsule operating temperatures for extended periods of time. No strengthening of the welds was observed for a heat treatment of 1175°F for 30 days. Concurrent

with the above work, a program was initiated to evaluate electron beam (EB) welding of the defect nubs to the cladding. Tests of electron beam welded specimens showed burst strengths similar to those of B9C which were TIG welded. Figure 3-3 shows the appearance of an EB welded nub. It was also determined that EB welded nubs will require a post-weld ballizing but that this can be done without compromising the weld integrity. Determination of which of the two above methods will be used to attach the B9B defect nubs will be made upon completion of the two test programs.

Testing was continued on the (stationary) centrifugal gas de-entrainer for capsule B9B described in the 35th Quarterly Report. A second model of the centrifugal type gas de-entrainer was fabricated. The model was sized from the previous model to simulate that required for insertion in a forced convection capsule. The de-entrainer was also modified to handle flowrates of up to 15 gpm. The maximum sodium flowrate during capsule operation will be 12 gpm. Prior to the modification, excessive turbulence on the vanes of the de-entrainer was observed at the highest capsule flowrates. Additional slots and vanes were added to the de-entrainer to reduce the velocity. Figure 3-4 shows the appearance of the gas de-entrainer after modification. The overall height is 3 inches. It has eight vanes (~0.333" radius of curvature) and eight flow slots with a total flow area of 0.437 in². A schematic showing the arrangement of

Figure 3-3. Appearance of Weld Area of Flanged Defect Nub Electron Beam Welded to a Simulated Clad Tube

Figure 3-4. Centrifugal Gas De-Entrainer

Figure 3-5. Rate of Gas Entrainment versus Water Flowrate

the test equipment and a description of the test procedures were given in the 35th Quarterly Report.

Tests were performed to determine the efficiency of the smaller model. The tests showed the efficiency of the two models was similar (>90%). However, due to slightly increased turbulence of the vortex surface, a slightly larger residual re-entrainment is measured. Figure 3-5 shows the rate of re-entrainment as a function of the coolant flow rate. A thermal-hydraulic analysis of the effects of the entrained gas on an operating FCC was made and confirmed that it has an insignificant effect.

The performance of the de-entrainer is related to the level of the water surrounding it. Figure 3-6 shows the recommended minimum static fill level for a capsule for optimum dynamic operation as a function of the coolant flowrate. The levels shown were used to obtain the performance information shown in Figure 3-5. If a level higher than the recommended level is used, there is no degradation of performance. However, if the level is reduced below the recommended level, there is a deterioration of performance. The lower limit for the water level is labeled the "break-away level" in Figure 3-6. At this level, there is a dramatic break in performance such that re-entrainment rates become very high (1-10 cc/sec or greater). The

pressure drop through the de-entrainer was also measured for various flowrates and is given in Figure 3-7.

3.2.4 Forced Convection Capsule B10A

Capsule B10A will be used to study the propagation of pin failure to adjacent pins. Preparation of the hazard analysis required to irradiate capsule B10A in-core was continued. Work completed to-date on establishing detailed capsule performance criteria included completion of the analysis of the steady state operation of the capsule. Work during the report period was directed primarily toward analysis of transient flow or off-normal operation of the capsule. Analysis is continuing for a number of specific reactor accident conditions.

3.3 FISSION PRODUCT DISTRIBUTION

3.3.1 Miniloop

Preparation of equipment to permit sectioning of the thicker walled sections of the loop was completed during the report period. Final sectioning and leaching of the inner surfaces of the remaining sections were then completed and analysis of the leach samples is presently in progress.

Figure 3-6. Recommended Static Fill Level for Various Water Flowrates

Figure 3-7. Pressure Drop Through the De-Entrainer

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 burn-up), and fuel form (powder and pelleted fuel). Tests are continuing 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

Final remote assembly operations have been completed for both the C4G and C4H inner capsule assemblies. These capsules are designed to determine the effect of burnup on failure threshold. The operations entailed remote insertion of the fuel specimens into their inner capsules and remote welding of the bottom closure. A successful remote sodium fill operation was completed on a dummy inner capsule, and the actual fill operation was initiated on the C4G inner capsule. Difficulties in maintaining sodium flow throughout the fill system resulted in an extended period of furnace heating of the inner capsule. As a result, there was apparent heat damage to the upper instrumentation connector (external to the containment barriers), and the connector is being replaced. Following the replacement, the inner capsule will be neutrographed to determine if an acceptable sodium level was attained. The sodium in the remote sodium fill system is being filtered and cleaned and the filling procedures modified to avoid the conditions that led to overheating of the connector.

4.2.2 TREAT Irradiation

The two calibration transients performed on the prototype capsule C4X (TT1323 and TT1324) were successfully correlated by thermal analysis using the THTD computer code and a complete symmetry mockup of the Series 4 inner capsule. The computer model geometry simulates a 30° section of the inner capsule as shown in Figure 4-1.

The results of the thermal analysis indicate that the actual C4X specimen power factor measured by capsule calorimetry is only 60% of the power factor predicted by physics calculations (0.925×10^{-4} versus predicted $1.54 \times 10^{-4} \frac{\text{watt/cc fuel}}{\text{TREAT watt}}$). Correlations obtained on previous experiments⁽¹⁾ indicate that the difficulty stems

from either the high enrichment, or the hardened spectrum peculiar to this capsule series.

A similar over-prediction of power factor was noted by other experimenters during a similar test involving highly enriched preirradiated fuel (PNL-1-2 irradiated to 8,000 MWd/Te in EBR-II)⁽²⁾.

Review of the physics calculational techniques is underway in an attempt to understand and rectify the differences between calculated and measured power factors. Using the REAX code⁽³⁾, initial studies indicate that the epithermal flux profile (energies $0.625 \leq E \leq 55.6$ ev) is not flat (as was assumed in the calculations for previous capsules), but has significant radial dependence as illustrated in Figure 4-2. This spatial self shielding above thermal neutron energies accounts for nearly half of the discrepancy between the predicted and actual C4X power factors.

The calculated versus measured sodium temperature histories for the two C4X calibration transients are presented in Figures 4-3 and 4-4. The calculated maximum fuel temperatures are $\sim 2300^\circ\text{F}$ and $\sim 2860^\circ\text{F}$ for TT1323 (96 MW-sec) and TT1324 (153 MW-sec) respectively. As evidenced in Figures 4-3 and 4-4, there are significant differential responses in both transients, between the 20 mil and 40 mil thermocouples located 180° apart, and adjacent to the cladding. In addition, during the larger transient, local temperature perturbations appear in all three thermocouples in the inner sodium region (Figure 4-1). These perturbations may be a result of either temperature-density induced convective sodium currents, or fuel pin oscillatory movement resulting from the severe transient heating, or both.

Based on the measured power factor, a large test transient was performed. The primary objective of the test transient was to induce C4X cladding failure as a final checkout of the TREAT capsule design. This test would indicate the reactions of both the capsule flowmeter and pressure transducer to fuel pin failure, and provide background for correlation of previous and upcoming experiments.

The test transient resulted in $\sim 62\%$ (calculated) of the fuel volume reaching melt temperature and peak cladding temperatures near 1800°F . These conditions were expected to have produced cladding failure, but examination of capsule data and the post-transient neutron radiographs, indicate that the specimen did not fail. The TREAT fast neutron hodoscope⁽⁴⁾ was used in an attempt to detect fuel motion during the transient, but data reduction will take some time.

It is planned to subject the C4X capsule to an additional transient to cause failure. Prior to this second irradiation, additional neutron radiographs will be taken with the

Figure 4-1. Task C Series IV Inner Capsule Layout, Computer-Modeled Geometry

inner capsule removed from the heavy-walled pressure vessel. This may permit viewing of greater detail of interior fuel movement as partial confirmation of the calculated molten fuel volume.

The comparison of calculated and measured sodium temperature histories for the completed test transient (TT1329), show good correlation (Figures 4-5 and 4-6). The resulting radial fuel and cladding temperatures are presented in Figure 4-7. The temperature perturbations noted earlier in the calibration transient TT1324, appear in magnified proportions in both sets of the three inner sodium region thermocouples (Figures 4-5b and 4-6b). Again there appears to be some phasing of the perturbations which may be due to oscillatory fuel pin movement.

4.3 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

This test series 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 refer-

ence gas-filled pellet pin. The three pins in each assembly were separately encapsulated and were arranged side-by-side in a triangular array (Figure 4-10, 31st Quarterly Report, GEAP-10028-31).

Destructive examination of both capsule assemblies has been completed. Initial results were given in the 35th Quarterly Report, GEAP-10028-35.

Profilometer traces of the C6A fuel pins indicated clad distortion of ≈ 2.5 to ≈ 4 mils for the sodium-filled powder pin. Both pellet pins had less than 1 mil deformation.

Figures 4-8 through 4-13 show transverse sections from the six fuel specimens (all locations correspond to the thermocouple locations for which data were shown in the 35th Quarterly Report, GEAP-10028-35). The effect of the sodium is clearly evident in the fuel microstructure of the sodium-filled specimens. The increased heat transfer afforded by the sodium resulted in significantly lower fuel temperatures and melt volumes. A summary of the fuel melt diameters (estimated from fuel microstructure) is shown below. Although some melting occurred in the powder fuel samples, the molten fuel area could not be adequately defined.

Figure 4-2. Task C Series IV Fuel Specimen C4X Epithermal Power Distribution

Figure 4-3a. Capsule C4X - First Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-3b. Capsule C4X - First Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-4a. Capsule C4X - Second Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-4b. Capsule C4X - Second Calibration Transient Results, Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-5a. Capsule C4X - First Test Transient Results, Sodium Temperature Histories to Equilibrium (Calculated vs. Measured)

Figure 4-5b. Capsule C4X - First Test Transient Results, Expanded Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-6a. Capsule C4X - First Test Transient Results, Sodium Temperature Histories to Equilibrium (Calculated vs. Measured)

Figure 4-6b. Capsule C4X - First Test Transient Results, Expanded Sodium Temperature Histories (Calculated vs. Measured)

Figure 4-7. First Test Transient Results Calculated Radial Fuel and Cladding Temperature Profiles (Capsule C4X)

20 X Polished

A505-02

20 X Etched

A505-17

Figure 4-8. Specimen C6B-1 (Sodium-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec

20 X Polished

A509-03

20 X Etched

A509-12

Figure 4-9. Specimen C6B-2 (Sodium-Filled Powder) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec

20 X Polished

A515-03

20 X Etched

A515-11

Figure 4-10. Specimen C6B-3 (Gas-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1308-442 mw-sec

20 X Polished

A540-02

20 X Etched

A540-12

Figure 4-11. Specimen C6A-1 (Sodium-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1312-485 mw-sec

20 X Polished

A547-12

20 X Etched

A547-11

Figure 4-12. Specimen C6A-2 (Sodium-Filled Powder) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1312-485 mw-sec

20 X Polished

A535-02

20 X Etched

A535-10

Figure 4-13. Specimen C6A-3 (Gas-Filled Pellet) Transverse Sample Showing Fuel Microstructure at the Fuel Column Mid-Plane, TT1312-485 mw-sec

<u>Specimen</u>	<u>Melt Diameter</u>	<u>Melt Volume</u>
C6B-1 (Sodium, Pellet)	~ 90 mils	~17%
C6B-3 (Gas, Pellet)	~150 mils	~48%
C6A-1 (Sodium, Pellet)	~120 mils	~30%
C6A-3 (Gas, Pellet)	~170 mils	~60%

A preliminary design study to determine the feasibility of using multipin TREAT capsules which can accommodate specimens previously irradiated in EBR-II or GETR is being conducted. The Series 6 capsule as presently designed is

acceptable for transient irradiation of low or slightly enriched specimens, but fully enriched specimens will require thermal flux filters.

The use of boron and nickel as possible filter material was considered. Computed power distributions and power normalization factors for boron and nickel filters are shown in Figures 4-14 and 4-15. In this design, the filter material is placed outside the capsule region containing the three encapsulated test pins to avoid the local perturbations which would occur if a filter were placed inside each capsule.

REFERENCES

1. Thomas, G. R. and Field, J. H., "Transient Overpower Irradiation of Axially-Restrained, Zero-Burnup Fast Reactor Fuel Specimens, "September 1969, (GEAP-13562).
2. Cabell, C. P., "FFTF Monthly Informal Technical Progress Report, June 1970," July 1970, (BNWL-1300-6).
3. Prostik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod — The REAX Code, "April 1966, (GEAP-4973).
4. ANL, "Reactor Physics Division Annual Report, July 1, 1968 to June 30, 1969," January 1970, (ANL-7610).

Figure 4-14. Power Normalization and Distribution, No Filter vs. Ni Filter

Figure 4-15. Power Normalization and Distribution, B_4C Flux Filter

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 potential for 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 under-cooling 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 reentry, and subsequent contact with hot or molten fuel. The purpose of this task is to begin a program of such tests, utilizing the Forced Convection Capsule and/or the Mark II Loop as the basic vehicle.

5.2 MOCKUP 1A

Mockup 1A is an out-of-pile loop to investigate loss of flow. It is a concentric loop with three 24 inch long electric heaters which simulate fuel rods.

X-rays of the Mockup 1A heaters (fuel simulators) revealed that the failure of the two heaters which were operating during the loss-of-flow test resulted from melting of the resistance element. The molten material was dispersed within the heater but did not result in a short to the clad material. This type of failure is consistent with that experienced in the heater development program of ORNL. The premature failure of the third heater was apparently the result of separation of the resistance element from the input electrode resulting, perhaps, from an inferior weld at this point.

Based upon this and the encouraging results obtained in the ORNL heater tests, heaters of the ORNL type are being ordered from the manufacturer. A schedule and estimate is being prepared for the assembly, sodium filling and operation of a second loss-of-flow test in the Mockup 1A facility. This second test is scheduled for operation in early 1971.

5.3 C10X

Capsule C10X is the vehicle for a study of the interaction of sodium and molten fuel designed to establish the pressures and energies associated with the reaction.

Fabrication of the C10X capsule is continuing, sodium filling of the capsule has been completed and subsequent assembly steps are in progress. A design audit has been conducted in response to the difficulties encountered in the Task B capsule B9C. As a result of this audit, several weld and braze modifications were suggested and are being implemented. Some schedule slippage has occurred and C10X operation is now planned for GETR cycle 122 (November 1970).

The proof test of the burst diaphragm operation has been completed. This test utilized water and a transparent mockup of the C10X capsule (Figure 5-1) to simulate the actual capsule geometry. The results were recorded with a high speed motion picture camera. The test was successful in demonstrating that the burst diaphragm ruptured as required and that complete expulsion of the NaK would require approximately 0.25 seconds.

A computer simulation of the transient behavior of capsule C10X following sodium boiling has been executed. This study was initiated to assess the influence of parameters such as plenum temperature, initial liquid film thickness and superheat, on clad temperature, system pressure and the general time dependent system response. It was found that the main parameter governing the dynamics of the bubble growth and collapse is the plenum temperature and heat capacity. If the plenum temperature is low, the condensation rate is high and the bubbles collapse very rapidly, conversely, for a high plenum temperature bubble collapse is restricted or prevented.

A parametric study has shown that, in the range of interest, the influence of the initial superheat and evaporation coefficient is relatively minor. The thickness of the liquid film, on the other hand, has a much greater influence on the observed results. The literature shows some scatter in the reported values of film thicknesses, Kottowski⁽¹⁾ measured 0.003-0.010 inches while Peppler⁽²⁾ found values between 0.005 and 0.025 inches. For the purpose of this investigation a value of 0.010 inch was assumed.

A specific (computed) case of bubble growth and collapse for an initial film thickness of 0.010 inch, a superheat of $\sim 30^\circ\text{F}$ and a plenum temperature of 1200°F is shown in Figure 5-2. This should be compared with Figures 5-3 and 5-4, for plenum temperatures of 480°F and 1600°F respectively. These results confirm the need for the initial design requirement that the plenum be heated to approximately 1600°F prior to initiation of the transient.

5.4 C10A/C10B

Capsules C10A/C10B are designed for the study of a total loss of coolant flow.

The final capsule specifications for capsule C10A have been issued and the final conceptual design based upon these specifications is nearing completion.

Figure 5-1. Rupture Disc Test Fixture

Figure 5-2. Computed Void Growth and Clad Temperature for Capsule C10X

Figure 5-3. Computed Void Growth and Clad Temperature for Capsule C10X

Figure 5-4. Computed Void Growth and Clad Temperature for Capsule C10X

Work continued on the updating and checkout of the THTE program. The isothermal phase change option has been incorporated and successfully checked out. The remote terminal summary has been extended to include all execution time error comments. Work has been initiated on the incorporation of the binary gas and thermal expansion options from the THTD program to facilitate the analysis of the forced convection capsule. This will involve the

inclusion of seven new subroutines and modifications to approximately twenty-five existing routines.

5.5 PLANNING AND DESIGN

The thermal-hydraulic study of the ANL Mark II loop and the task C-2 work plan are in the final stages of review. The Mark II loop study will appear as an appendix to this work plan.

REFERENCES

1. H. Kottowski, "Über die Ausbildung einer Restschicht beim Sieden von Alkalimetallen in Kanälen", ATKE 15, 1970.
2. W. Peppler, "Natrium-Siedestosssexperimente in einem grösseren Kreislauf unter Verwendung induktiv beheizter Teststrecken", KFK 964, paper IX, April 1969.

6. TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

6.1 GENERAL

The purpose of Task E 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 experimental program to measure compressive creep of mixed plutonium-uranium oxides has been completed. Experimental results on the effect of stoichiometry, density, composition, temperature and stress have been previously reported.^(1,2) Studies on small grain size samples (4 microns) of composition, (U-20 wt% Pu)O_{1.96} show that the initial grain size has a significant effect on the creep rate.

Results of these studies with compressive stresses varying from 1000 to 8000 psi and temperature from 1250 to 1400°C are shown in Figure 6-1. These data indicate a linear creep rate stress dependence ($\dot{\epsilon} \propto \sigma$) with an activation energy of 105 ± 10 Kcal/mole. The small grain size seems to enhance the linear stress dependency and no power law relationship was found at stresses up to 8000 psi. This is quite different from the situation for large grains where the power law $\dot{\epsilon} \propto \sigma^{4.5}$ is well established. A comparison of the small (~4 micron) grain size samples with other data on mixed oxides shows that the strain rate is inversely proportional to the square of the grain size $\dot{\epsilon} \propto \frac{1}{(G.S.)^2}$. An identical grain size effect was previously noted in stoichiometric UO₂ materials.⁽³⁾

6.2.2 Thermal Conductivity Studies

Thermal conductivity measurements on mixed plutonium-uranium oxide samples have been completed. Experimental variables included stoichiometry, plutonium content, density, physically mixed and coprecipitated powders. Results on the effect of stoichiometry and the powder preparation technique (coprecipitated vs. physically mixed) have been presented previously⁽²⁾. The effect of density on mixed oxide thermal conductivity above 1000°C was found to be small and within the experimental error of the test technique (~±10%). These data were obtained on

mixed oxide fuel pellets of 84, 92 and 95% of theoretical density. This insensitivity to density differs from the earlier work by Asamoto et al.⁽⁴⁾ on stoichiometric UO₂. This difference may be due to the size, shape and distribution of the porosity. In UO₂ samples the density variations were primarily differences in the number of pores (pore size did not vary widely), while in the mixed oxide samples, the density variations were predominantly due to pore size variations.

Measurements on mixed oxide samples with plutonium concentrations of 10, 30 and 100 weight percent indicate virtually no variation in conductivity. These data were obtained in tests carried out in a dry He-6% H₂ atmosphere that results in constant oxygen-to-plutonium ratios at each of the test temperatures. The O/Pu ratios varied from approximately 1.82 at 800°C to 1.65 at 2400°C. A UO_{2.00} pellet was included as a control in each test and consistently gave thermal conductivity values greater than the mixed oxide or pure plutonia samples.

A slight increase in the thermal conductivity of the mixed oxide at high temperatures (2000°C) was observed in most of the tests. This behavior may be due either to the introduction of radiative effects on the heat transfer or additional, unaccounted for, heat generation in the mixed oxide pellet samples. At high temperatures, the electrical conductivity of UO_{2.00} or (Pu, U)O_{2.00} may be sufficiently large that Joule heating may be induced in the pellets themselves as well as in the central tungsten heater rod. Ainscough⁽⁵⁾ has shown that such power generation in the specimens will lead to an artificially high value for the measured thermal conductivity. Further analysis of this effect is underway.

6.3 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

6.3.1 Capsule E2R (Rate of Center Hole Formation at Startup)

The E2R capsule fabrication and assembly has been completed. The irradiation will be initiated upon completion of a neutrographic standard that will enable small changes in pellet dimensions or center void size to be measured during the series of planned incremental irradiations.

The neutrography standard will contain a stack of mixed oxide pellets individually characterized for dimensions, isotopic content, density and plutonium concentrations to duplicate as nearly as possible the neutron attenuation characteristics of the E2R fuel pin. This will maximize the precision of the measurements taken from neutrographs and will complement earlier work⁽²⁾ in determining the optimum neutron radiograph parameters.

Figure 6-1. Steady State Creep of (U-20 wt% Pu)O_{1.96} with 4 and 25 Micron Grain Sizes

6.4 FISSION PRODUCT — FUEL-CLADDING REACTIONS

6.4.1 Out-of-Pile Studies in a Thermal Gradient

Six type 316 stainless steel clad fuel pins have been heat treated in a thermal gradient furnace (500-1250°C) to study volatile fission product migration, combination, and reaction with cladding. The capsules contain either (U-25 wt% Pu)O_{2.00} or (U-25 wt% Pu)O_{1.97} mixed oxide fuel compositions. Cs₂O, Te and I as PdI₂ were loaded singly or together at the hot end of the six pins as shown in Table 6-1. The fission product additives were spiked with appropriate radioactive nuclides to measure migration by gamma activity measurements after the 100 hour tests. The initial Cs, I and Te concentrations were equivalent to fission yields that would be present after 2 to 8 at. % burnup. Test results from these experiments are shown in Figures 6-2 and 6-3. Figure 6-2, showing migration of cesium and iodine, indicates that both transport to the cool end of the capsule. Residual concentrations of the radioisotopes were apparently retained at the hot region where they were originally positioned within the capsule. Figure 6-3 shows the migration of cesium when tellurium is present. The behavior of the Cs and Te, with the deposition of both at a location about three inches from the cold end, implies that Cs and Te cohabit in the fuel or exist as cesium telluride or a tellurite compound that condenses or solidifies at some temperature in the vicinity of 800-900°C. Telluride and tellurite compounds of sodium and potassium (also of Column 1 in the Periodic Table) have been reported. The restricted migration of cesium in the presence of tellurium is in contrast to its movement in capsules without any additives, as shown in the same Figure, and to its movement in the presence of iodine, Figure 6-2. Additional gamma activity measurements will be performed on these samples after removal of the fuel. The capsules will also be examined metallographically for fission product reactions with the cladding.

6.5 MICROPROBE ANALYSIS OF IRRADIATED FUEL

6.5.1 Plutonium Analyses

Microprobe analysis of the second group of three pins from the EIJ test series was completed. These pins contained (U-25 wt% Pu)O_{2-x} fuel and were irradiated in a thermal flux at power levels sufficient to produce center melting. Fuel of three stoichiometries were analyzed: O/M of - 1.94 (EIJ-3), 1.97 (EIJ-4) and 2.00 (EIJ-8). The fabrication parameters and the results of the examination of the earlier group of samples were previously reported^(1,6). One total and one partial radius of each sample

were analyzed for Pu and U and the $\frac{\text{Pu}}{\text{U} + \text{Pu}}$ ratio calculated.

These results, along with the photo micrographs showing the microstructural regions analyzed are shown in Figures 6-4, 6-5 and 6-6.

The data verify the previously reported results in that no variation in plutonium was observed in these rapidly quenched samples at the location of microstructural features generally considered to be the molten fuel boundary*. No microstructural feature corresponded to the location of highest plutonium concentration which was within the high density fuel zone near the center void.

A gradual change in plutonium content was noted in the cooler regions of the fuel. The porous regions in the central void exhibited a variable plutonium content; between radius b and the central void, the Pu/U+Pu ratio decreased, then increased at a smaller radius. The largest change in Pu content occurred in the sample with 1.94 O/M fuel, while no definable change occurred in the stoichiometric sample. The pin with 1.97 O/M fuel exhibited a Pu gradient that was intermediate between that of the other two stoichiometries. Figure 6-7 shows two Pu rate-out traces over negative specimen current images. Pu and U content determinations were made at the points indicated on the traced lines and the Pu/(U+Pu) ratio calculated. From these analyses and from the metallographic observations it is apparent that some Pu segregation occurred during the 84 hour irradiation cycle. These conclusions were substantiated by Pu - Mg x-ray images of the area which show that the plutonium was concentrated in the porous regions of the molten zone. Figure 6-8 is a negative specimen current photo and an x-ray image of the area that shows the concentration of plutonium in the area normally defined as the molten boundary in the porous region of the fuel.

6.5.2 Initial Data from Microprobe Examination of Sample F2C

Fuel clad reaction zones in sample F2C are currently being analyzed using the shielded electron microprobe. Fuel pin F2C contains mixed oxide fuel with an initial O/M of 2.00 clad in Incoloy 800 and irradiated in EBR-II to a peak burnup of about 65,000 MWd/Te. A crack or intergranular separation was observed in the high temperature region of the inside surface of the clad similar to observations on other fuel pins examined at the same time (F2C, H, O, R and T).

The intergranular area was observed to contain only molybdenum and clad constituents. This is shown in Figure 6-9 along with the Te and Cs x-ray images. The Fe, Cr, Ni and Mn had uniform distributions across the fuel-clad reaction layer. A metallic phase composed of

* The start of fine porosity in the high density region near the center void, radius a in Figure 6-4, 6-5 and 6-6.

Figure 6-2. The Migration of Cesium and Iodine Added to $(U-25 \text{ wt\% Pu})O_{2-x}$ in a Thermal Gradient

Figure 6-3. The Migration of Cesium and Tellurium Added to $(U-25 \text{ wt\% Pu})O_{2-x}$ in a Thermal Gradient

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.

50X

Figure 6-4. Radial Plutonium Concentration Across Specimen EIJ3 (U-25 wt % Pu) $O_{1.94}$

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.

Figure 6-5. Radial Plutonium Concentration Across Specimen EIJ4 (U-25 wt % Pu)O_{1.97}

50X

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.

50X

Figure 6-6. Radial Plutonium Concentration Across Specimen EIJ8 (U-25 wt % Pu)O_{2.00}

(REGIONS SHOWN EXTEND FROM FUEL CENTER TO POROSITY LIMIT)

Figure 6-7. Plutonium Concentrations Over Positive Specimen Current Images of Sample E1J-3 with an O/M = 1.94

CENTER OF
FUEL PIN

LINE b IN
FIGURE 6-4

250 X

POSITIVE SPECIMEN CURRENT IMAGE

CENTER OF FUEL PIN

EDGE OF POROUS
REGION

LINE b IN FIGURE 6.4

250 X

Pu - M_{β} X-RAY IMAGE

(PLUTONIUM CONCENTRATIONS INCREASE WITH INCREASING INTENSITY)

Figure 6-8. Positive Specimen Current Image and X-Ray Images Showing Plutonium Concentrations in Porous Area of E1J-3 (1.94 O/M)

500 X

CRACK OBSERVED IN CLAD

500 X

$M_o - L_{\alpha}$ X-RAY IMAGE

500 X

$C_s - L_{\alpha}$ X-RAY IMAGE

500 X

$T_e - L_{\alpha}$ X-RAY IMAGE

(BRIGHTNESS IN AN X-RAY IMAGE IS A MEASURE OF CONCENTRATION)

Figure 6-9. X-Ray Images of Crack Observed 500X in the Incoloy 800 Cladding of F2C

Table 6-1
CONTENTS OF FUEL PINS COMPRISING THERMAL GRADIENT
TEST OF FUEL – FISSION PRODUCT-CLADDING REACTIONS

Pin	Fuel	Prime Additive		Secondary Additive	
		Species	Amount	Species	Amount
1.	(U-25 wt% Pu)O _{1.97}	Cs ₂ O	50 mg	I as PdI ₂	10 mg
2.	(U-25 wt% Pu)O _{1.97}	Cs ₂ O	50 mg	Te	40 mg
3.	(U-25 wt% Pu)O _{2.00}	Cs ₂ O	50 mg	I as PdI ₂	10 mg
4.	(U-25 wt% Pu)O _{2.00}	Cs ₂ O	50 mg	Te	44 mg
5.	(U-25 wt% Pu)O _{2.00}	Cs ₂ O	100 mg	-	-
6.	(U-25 wt% Pu)O _{2.00}	Cs ₂ O	200 mg	-	-

NOTE: 1. Fuel Pins are heat treated in a thermal gradient, 500°C–1245°C for 100 hours.

molybdenum was observed in one area parallel to the clad. The relative distributions of uranium and molybdenum are shown in Figure 6-10.

An area with the fuel in direct contact with the clad was also examined. This was shown to be primarily fuel with a few metallic particles. Figure 6-11 shows the x-ray images for iron, plutonium and uranium in this area. This structure is similar to those observed in previous examinations of fuel rods in the F2 series.^(7,8)

6.5.3 Oxygen Distribution in Irradiated Fuel

A group of three EIJ samples analyzed for plutonium content (section 6.4.2) was also analyzed for oxygen distribution using the shielded electron microprobe. Initial attempts to measure gradients indicate that the fuel stoichiometry near the clad was identical for each of the specimens examined. The oxygen signal for the stoichiometric pin was relatively constant as a function of radial position (except in the melt region) while the signal for the 1.94 O/M pin dropped abruptly near the columnar grain region. These data are plotted in Figure 6-12 along with the expected oxygen distribution as calculated by a method detailed elsewhere⁽⁹⁾. The measured points are relative intensities associated with the specimen current and were not normalized against the standard. The radial slopes of oxygen intensities measured within the different samples

appear to correspond well with the calculated slopes of oxygen distribution. The error band is composed of the radial spread of the 20 points taken, along with a one sigma deviation around the mean value for these 20 points. Standardization and the determination of the lower limits of detectability for oxygen analyses are being investigated.

Chips from samples of (U-25 wt% Pu)O_{1.93} and (U-25 wt% Pu)_{1.97} (U-25 wt% Pu)O_{2.00} were also analyzed for oxygen. The relative intensities agree with the intensities shown for the EIJ samples. Development of the techniques for direct oxygen analysis using the microprobe will continue during the next reporting period.

6.6 FUEL-SODIUM REACTIONS

Portions of the fuel-sodium reaction product in the fuel-cladding gap of failed pin F20 (U-20 wt% Pu)O_{2.03} at ~65,000 MWd/Te were analyzed by x-ray diffraction. Only mixed oxide fuel was identified and the compound showed an expanded mixed oxide fuel lattice, equivalent to an O/M of about 1.97. The starting unit cell size of 5.453 Å increased to 5.458 Å during irradiation and exposure to sodium. Virtually no sodium (<1%) was detected during subsequent examination by x-ray spectroscopy. This increase in fuel lattice parameter cannot account for the large volume changes associated with this reaction.

AREA OF METALLIC PHASE OBSERVED IN THE FUEL

U - M_{α} X-RAY IMAGE

$M_o - L_{\alpha}$ X-RAY IMAGE

(BRIGHTNESS IN AN X-RAY IMAGE IS A MEASURE OF CONCENTRATION)

Figure 6-10. X-Ray Images in Comparison to the Photomicrograph of a Metallic Phase Observed in the Fuel of F2C

AREA OF FUEL-CLAD CONTACT

$\text{Fe} - \text{K}_{\alpha_{1,2}}$ X-RAY IMAGE

$\text{Pu} - \text{M}_{\beta}$ X-RAY IMAGE

$\text{U} - \text{M}_{\alpha}$ X-RAY IMAGE

(BRIGHTNESS IN AN X-RAY IMAGE IS A MEASURE OF CONCENTRATION)

Figure 6-11. X-Ray Images in Comparison to the Photomicrograph Showing a "Fuel" Phase that has Formed in Contact with the Clad in Sample F2C

a) 2.00 STARTING O/M, EIJ-8

b) 1.97 STARTING O/M, EIJ-4

c) 1.94 STARTING O/M, EIJ-3

Figure 6-12. Comparison of Calculated to Measured Oxygen Levels in EIJ-3, EIJ-4, and EIJ-8. Measured Values are Relative

REFERENCES

1. Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, the Thirty Fourth Quarterly, May, 1970, (GEAP 10028-34).
2. Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty Fifth Quarterly, August, 1970, (GEAP 10028-35).
3. Bohaboy, P. E. et al, "Compressive Creep Characteristics of Stoichiometric Uranium Dioxide," May, 1969, (GEAP 10054).
4. Asamoto, R. R., Anselin, F. L., Conti, A. E., "The Effect of Density on the Thermal Conductivity of Uranium Dioxide", April, 1969, (GEAP 5493).
5. Ainscough, J. B., "The Effect of Electrical Conduction in UO_2 on the Measurements of its High Temperature Thermal Conductivity as determined by Radial Flow Techniques" *J. Nucl. Materials* 20, 184-192 (1966).
6. Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty Third Quarterly, February, 1970, (GEAP 10028-33).
7. Nelson, R. C., Baily, W. E., Duncan, R. N., Spalaris, C. N. and Zebroski, E. L. "Performance of Plutonium-Uranium Mixed-Oxide Fuel Pins Irradiated in a Fast Reactor (EBR-II) to 50,000 MWd/Te," August, 1969, (GEAP 13549).
8. Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty Ninth Quarterly, November 1968-January 1969, (GEAP 5753).
9. Aitken, E. A., Thermodynamic Analysis of Possible Chemical Interactions in the System UO_2 - PuO_2 Fuel-Sodium-Stainless Steel, July, 1968, (GEAP 5683).

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, 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 (under 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 MEDIUM POWER, LONG TERM IRRADIATIONS

7.2.1 Group F2 (Medium Power Scoping Irradiations)

Post irradiation examination of the four Group F2 encapsulated pins (F20, F2G, F2V, F2X) continued. These pins were irradiated to a calculated peak fuel burnup of 117,000 MWd/Te and a total fluence of $\sim 1.0 \times 10^{23}$ nvt*.

The power-time operating history of these fuel pins was presented in the 35th Quarterly Report (GEAP-10028-35). Detailed fabrication parameters for each pin are listed in Table 7-1. The axial temperature profiles for the F2G capsule and fuel pin are presented in Figures 7-1 and 7-2.

7.2.1.1 Post Irradiation Examination

Visual Examination

The capsules were found to be in excellent condition when removed from the subassembly, although shallow longitudinal scratches were observed in the capsule tubes. The scratches appeared to be made by the dimples in the subassembly shroud tubes.

Bow measurements made at EBR-II on capsules F2D and F2G indicated maximum bow was 0.010 to 0.011 inch.

Neutron Radiography

Neutron radiographs of the fuel pins after irradiation are shown in Figure 7-3. The radiographs clearly showed that pin F2D had failed. Neutron radiographs of the F2D pin at a burnup of approximately 60,000 MWd/Te showed that the pin was intact. The fuel in this pin was axially restrained.

The neutron radiographs also show a central void in each pin and that high cross section nuclides were present in the central voids of pins F2D, -G, and -V. Several of these regions were present in the grossly failed areas of pin F2D while only one or two could be seen in pins F2G and F2V.

Figure 7-4 shows profilometer traces of the diameter

*All burnup and fluence values were calculated based upon fission rates in the EBR-II Experimenters' Guide with corrections made for earlier core changes. The burnup values include corrections for fuel depletion.

Table 7-1
MIXED OXIDE FUEL PRE-IRRADIATION INFORMATION

Pin	Type	Form	Fuel Weight Grams	Fuel Length Inches	Composition % PuO ₂	Enrichment % U-235	Density		O/M	Diameter (Pellet) Inches	Diametral Gap Mils
							Theoretical %	Smeared %			
F2D	SP	CPT	92.60	14.286	19.40 - 21.00	93.00-94.10	94.93-97.77 96.44	94.31	1.998 - 2.001 2.000	.2163-.2184 .2178	1.8-4.1
F2G	SP	CPT	90.63	14.219	19.70 - 20.50	91.58-94.00	93.50-97.05 95.56	92.95	1.967 - 1.980 1.975	.2146-.2193 .2173	0.9-6.0
F2V	SP	CPT	84.53	14.227	20.00 - 20.00	93.00-93.00	88.35-90.69 89.69	86.66	1.984 - 1.989 1.988	.2158-.2174 .2164	2.8-4.5
F2X	VC	CPT	81.42	14.190	20.00 - 20.00	93.00-93.00	83.90	83.90	1.989 - 1.995 1.992	-	-

GROUP F2
CLADDING
PRE-IRRADIATION INFORMATION

Pin	Type	ID Inches	Wall Thickness Inches	OD Inches	Clad Cond.	Heat Number	Cert. Number	Fab. Tech	ASTM Grain Size	O/A Pin Length Inches
F2D	347	.2202-.2204 .2202	.0147-.0154 .0150	.2493 - .2497 .2495	ANNSB	67276	385133	WD	7	35.5470
F2G	347	.2202-.2206 .2203	.0141-.0154 .0150	.2436 - .2496 .2493	ANNSB	67276	385133	WD	7	35.5020
F2V	316	.2202-.2203 .2202	.0137-.0150 .0150	.2485 - .2485 .2485	ANNSB	804078	385135	WD	6	35.5350
F2X	316	.2202-.2202	-	.2485 - .2500	ANN	804078	385135	WD	6	35.2990

NOTE:

SP = Solid Pellet
VC = Powder Fuel

CPT = Coprecipitated
ANNSB = Annealed, Swaged and Ballized

WD = Welded and Drawn

Figure 7-1. Axial Temperature Profiles — Capsule F2G

Figure 7-2. Axial Fuel Temperature Profile – Fuel Pin F2G

Figure 7-3. Neutron Radiographs of Group F2 Fuel Pins Irradiated to $\sim 105,000$ MWd/Te (11.75 at. %)

Figure 7-4. Comparison of Fuel Pin Diameter and Fuel Diameter Determined by Neutron Radiography

of pins F2G, -V, and -X, and fuel pellet diameters measured at EBR-II using neutron radiograph enhancement techniques. The correspondence between the clad and fuel profiles is quite good. As a result of the tolerance on the measurement of the fuel diameter and uncertainties in the calculation of clad swelling the usefulness of this technique cannot be determined at present. Further analysis of these data and evaluation of the enhancement techniques will be made when metallographic examination of the fuel pin samples is completed along with measurements of cladding density.

Gamma Spectroscopy

Typical gamma scans are shown in Figures 7-5 to 7-8. The general shape of the gross gamma activity of pins F2D (failed pin), F2G, and F2X were similar to the chopped cosine flux profile of the EBR-II with the exception of the gamma energy peaks detected along the length of pins F2D and F2G. The gamma scan of pin F2V indicated that a high concentration of gamma emitting nuclides was located near the bottom of the pin. Single channel gamma scans of pins F2D, -G and -V were made to identify the nuclides producing the activity peaks.

Pin F2V

Single channel gamma scans of this pin were made for Ba-La 140, Cs-134, Cs-137, Nb-95, Ru-103, and Zr-95. The scans showed that the gamma activity at the bottom of the pin resulted from concentrations of Ba-La 140, Cs-134, and Nb-95, the Nb-95 being a major contributor to the activity.

Subsequent sectioning and metallographic examination of pin F2V revealed that several "ingots" of metallic fission products were present in the bottom one inch of the center void. These "ingots" were not visible in the neutron radiographs. A typical ingot is shown in Figure 7-9. These regions of enhanced gamma activity also contained non-metallic fission product material. A region of the fuel containing a concentration of this material is shown in Figure 7-10.

Pin F2G

Single channel gamma scans of this pin were made for Cs-134, Cs-137, Nb-95, and Zr-95. Each peak observed in the gross gamma scan was found to correspond to Nb-95 activity. No actual peaks were detected for the other isotopes.

Sectioning and metallographic examination of pin F2G revealed multiple "plugs" of fission products in the center void. These fission product concentrations apparently caused the gamma peaking seen in the gross gamma scan. The fission product "plugs" were not visible in the neutron

radiographs. Figure 7-11 shows a fission product plug located 7-1/2 to 8 inches above the fuel bottom, the peak flux region of the pin. This corresponds to the location of one of the gamma peaks in Figure 7-7. The $\beta - \gamma$ autoradiographs in Figure 7-11 illustrate the enhanced activity in the region containing the fission products.

Pin F2D

Single channel scans of pin F2D for Cs-134 are shown in Figures 7-5 and 7-6. A concentration of Cs-137 was detected near the bottom of the fuel column which did not correspond to peaks in the gross gamma scan. No single channel scan was made for Nb-95. A comparison of the peaks occurring in the gross gamma scan and the high cross-section regions observed in the central void of this pin in the neutron radiographs showed a direct correlation.

DIAMETER AND LENGTH MEASUREMENTS

Capsule Tubes

Capsule diameter measurements were made using a continuous recording profilometer and a dial gauge. Measurements were also made at EBR-II using the DR-25 gauge. Profilometer traces were made at 0°, 45°, and 135° orientations and with the capsules rotating. Dial gauge measurements were taken at the 0° and 90° orientations.

Shown in Figure 7-12 are comparisons of the capsule diameter measurements made at EBR-II and the dial gauge and profilometer traces. This comparison shows that the EBR-II diameter measurements are inconsistent when compared to the dial gauge and profilometer traces. Since all three measurements agree closely in the plenum region of the capsule these data suggest that some bias, possibly associated with the orientation of the capsules, exists between the measurements made at EBR-II and GE.

The profilometer measurements of the capsule tubes showed periodic diameter variations of 1 to 1.5 mils over most of the fueled length of the capsules. The positions of the variations corresponded to the location of the spacer wire wrapped around the fuel pins. Similar periodic variations in the diameter of the fuel pins were not observed.

Measurements on the capsule tubes after removal of the pins showed no change in diameter occurred as a result of removing the pins.

The maximum diameter increases measured on each capsule tube are listed in Table 7-2. The maximum diameter changes ranged from 1.8 to 2.7%.

Calculations were made to determine whether the periodic variations in capsule tube diameter resulted from interference between capsule tubes and the wire wrap on the fuel pins as a result of increases in diameter of the

*Similar variations in the diameter of the F2 capsules removed at ~60,000 MWd/Te burnup were not observed.

Figure 7-5. Gross and Single Channel Gamma Scans of Pin F2D

Figure 7-6. Single Channel Gamma Scan of Pin F2D for Cs-134

Figure 7-7. Gross and Single Channel Gamma Scans of Fuel Pin F2G

Figure 7-8. Single Channel Gamma Scans of Pin F2G for Cs-137 and Zr-95

Figure 7-9. Typical Metallic Fission Product Observed in Center Void of Pin F2V

Figure 7-10. Typical, Non-Metallic, High Gamma Activity Material Deposited on Walls of Center Void on Pin F2V

(A569-01)

MACROGRAPH

(A569-62)

MICROGRAPH

(A569-03)

β - γ AUTORADIOGRAPH

Figure 7-11. β - γ Active Fission Product Plug in Center Void of Fuel Pin F2G

Figure 7-12. Comparison of EBR-II Measurements and Dial Gauge and Profilometer Measurements on F2D, G, V, and X Capsule Tubes

Table 7-2
SUMMARY OF GROUP F2 CAPSULE DIAMETER MEASUREMENTS*

CAPSULE	ORIENTATION								AVERAGE	RANGE OF
	0°		45°		90°		135°		Max ΔD (%)	Max ΔD (%)
	Max ΔD		Max ΔD		Max ΔD		Max ΔD			
	Mils	%	Mils	%	Mils	%	Mils	%		
F2D	9.8	2.61	9.0	2.40	8.6	2.29	8.6	2.29	2.40	2.29-2.61
F2G	9.5	2.53	9.4	2.51	9.3	2.48	8.4	2.24	2.44	2.24-2.53
F2V	10.0	2.67	9.6	2.56	9.8	2.61	8.8	2.35	2.55	2.35-2.67
F2X	7.5	2.00	7.1	1.89	6.8	1.81	7.8	2.08	1.95	1.81-2.08

*Measurements determined from GE profilometer traces

fuel pins. The worst possible tolerances were assumed for the capsule I.D. and wire diameter. Measured end-of-life pin diameters were used for the analysis. Capsule tube and fuel pin diameters were calculated at operating temperatures at end-of-life. A possible interference fit was calculated for two of the unfailed pins, F2G and F2V, however the diameter variations occurred with equal intensity on all three capsule tubes.

A summary of capsule tube diameter changes obtained from experiments examined previously and the F2D, -G, -V and X capsules is plotted versus fluence in Figure 7.13. The clad ΔD for pin F2X is also plotted for comparison. As can be seen the F2X capsule and fuel pin exhibited nearly the same maximum ΔD. Since this pin contained powder fuel it is assumed that no straining of the clad occurred due to the fuel clad interactions and all of the ΔD measured was due to clad swelling.

The correlation between the F2X capsule and pin ΔD is interesting also because of the temperature difference between the two. At the peak power attained, the average clad temperature in the peak flux region was 960°F. Average capsule tube temperature in the peak flux region was 860°F.

Fuel Pins

The fuel pins were removed from their capsule tubes by cutting the tubes at each end and melting out the sodium bond.

A photograph of pins F2G, F2V and F2X after removal of the capsule tubes is shown in Figure 7.14. After orientation marks were made the spiral wires were removed and the pins were cleaned in an alcohol bath. Figure 7.15 shows the typical appearance of the clad in the fueled region. The sodium oxide residue on pin F2V was removed with acetic

acid. No deterioration of the clad was found and no evidence of the location of the spiral wire remained after the pins were cleaned.

Length measurements made on the fuel pins are summarized in Table 7.3. Fuel column length changes based on neutron radiographs are summarized in Table 7.4.

Diameter measurements were made on fuel pins F2V and F2X using precision profilometry at 0°, 45°, 90°, 135° and with the pins rotating. Measurements were also made at 0° and 90° with a dial gauge device.

Spiral traces of the O.D. of fuel pins F2G, V and X are shown in Figure 7.16. Continuous profilometer traces of pins F2G, V, and X are shown in Figures 7.17 to 7.19. For comparison purposes the pellet lineups for each pin are graphically represented in each figure. * The vertical bars represent the O.D. of each pellet in the fuel column. The number at the bottom of each bar is the density of the pellet. The horizontal line represents the as fabricated diameter.

The profilometer measurements indicate that the peak diametral changes of pins F2G (93% smeared density pellet fuel), F2V (87% smeared density pellet fuel), and F2X (84% smeared density powder fuel) were, respectively 5.64% (14.1 mils), 3.88% (9.7 mils), and 2.24% (5.6 mils). A summary of the peak diameter measurements for each pin is given in Table 7.5.

Metallography

Sectioning of the pins has been completed and metallographic examination of sections from each pin initiated. Center voids were observed in the fuel in all pins. Shown in Figure 7.20 are transverse sections from pins F2G, V, and X. These sections were taken from the peak flux region of the pins. ** Fission products and fuel were observed

*No pellet lineup is shown for pin F2X since it contained powder fuel.

**The half moon shaped depression in the O.D. of the clad of the section from pin F2X is an orientation mark applied during post irradiation examination.

Table 7-3
SUMMARY OF F2 FUEL PIN LENGTH CHANGES

<u>Pin No.</u>	<u>F2G</u>	<u>F2V</u>	<u>F2X</u>
Pre-Irradiation Pin Length, in.	35.510	35.535	35.297
Post-Irradiation Pin Length, in.	35.805	35.793	35.668
ΔL , in.	0.295	0.258	0.371
ΔL , % of Pre-Irradiation *	2.07	1.81	2.61

*Assumes all of length change occurred over the fueled length of the pin.

Table 7-4
SUMMARY OF F2 FUEL LENGTH CHANGES

<u>Pin No.</u>	<u>F2G</u>	<u>F2V</u>	<u>F2X</u>
Pre-Irradiation Fuel Length, in. (measured)	14.219	14.227	14.190
Post-Irradiation Fuel Length, in. (from neutrograph)	14.518	14.583	14.474
ΔL , in.	0.299	0.356	0.284
$\Delta L/L$, % of Pre-Irradiation Length	2.10	2.50	2.00

Table 7-5
SUMMARY OF GROUP F2 FUEL PIN DIAMETER MEASUREMENTS

PIN	ORIENTATION								AVERAGE	RANGE OF
	0°		45°		90°		135°		Max $\Delta D/D$ (%)	Max $\Delta D/D$ (%)
	Max ΔD		Max ΔD		Max ΔD		Max ΔD			
	Mils	%	Mils	%	Mils	%	Mils	%		
F2D*	—	—	—	—	—	—	—	—		—
F2G	13.7	5.48	13.8	5.52	14.1	5.64	13.8	5.52	5.54	5.48—5.64
F2V	9.1	3.64	9.0	3.60	9.0	3.60	9.7	3.88	3.68	3.60—3.88
F2X	5.5	2.20	5.5	2.20	5.6	2.24	5.5	2.20	2.21	2.20—2.21

*Pin Failed

Figure 7-13. Diameter Changes of Capsule Tubes Irradiated in EBR-II to Different Fluences

Figure 7-14. Fuel Pins F2G, F2V and F2X after Removal from Capsule Tubes

Figure 7-15. Typical Appearance of Fueled Region of F2G, -V, and -X Cladding After Alcohol Bath

Figure 7-16. Spiral Profile Traces of Fuel Pins F2G, -V, and -X

Figure 7-17. Profilometer Traces of Type 347 Clad Fuel Pin F2G Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters

Figure 7-18. Profilometer Traces of Type 316 Clad Fuel Pin F2V Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters

Figure 7-19. Profilometer Traces of Type 316 Clad Powder Fuel Pin F2X

F2G (PELLET)

CLAD	347 ANNEALED
O/M	1.98
FUEL DENSITY	95.5% THEORETICAL
SMEARED DENSITY	92.9% THEORETICAL
PRE-IRR. DIA. GAP	0.9/6.0 MILS
LINEAR POWER*	16.0/13.5 kW/ft
% Δ L (FUEL PIN)	2.0
% Δ D (FUEL PIN)	5.9
BURNUP	12.7 at. %

F2V (PELLET)

CLAD	316 ANNEALED
O/M	1.99
FUEL DENSITY	89.8% THEORETICAL
SMEARED DENSITY	86.7% THEORETICAL
PRE-IRR. DIA. GAP	2.8/4.5 MILS
LINEAR POWER*	15.5/13.0 kW/ft
% Δ L (FUEL PIN)	1.8
% Δ D (FUEL PIN)	3.8
BURNUP	13.2 at. %

F2X (POWDER)

CLAD	316 ANNEALED
O/M	1.99
SMEARED DENSITY	83.9% THEORETICAL
LINEAR POWER*	15.2/11.5 kW/ft
% Δ L (FUEL PIN)	2.6
% Δ D (FUEL PIN)	2.2
BURNUP	11.0 at. %

*Start/End of Life Linear Power

Figure 7-20. Mixed Oxide Fuel Pins Irradiated to 13.2 at. % Burnup (118,000 MWd/Te) in EBR-II

in the center voids at random axial locations in each pin. A typical "bridge" is shown in Figure 7-11. The bridges appear to consist of more than one phase. One region has a dendritic structure (indicating that the material may have been molten), another phase appears to be amorphous. Identification of the constituents will be made by electron microprobe analysis. As discussed in the section on gamma spectroscopy these phases contain nuclides of high gamma activity

A summary of the observations made on the metallographic samples of pins F2G, -V and -X follows *

1. No cracks have been observed in the cladding of any pin
2. Very little chemical reaction was observed at the I.D. or O.D. of any pin. Typically less than 0.2 mil of attack was observed in the transverse sections from the peak flux region of each pin, however, isolated regions of slightly greater reaction depth (0.5 mil) were observed in a single longitudinal section from pin F2G. The peak I.D. clad temperature of these sections was calculated to be 1030°F
3. Fuel-clad gaps ranging from 2 to 3 mils were observed in the transverse sections from the peak flux region of each pin. Since significant fuel-clad gaps were not observed in similar fuel pins examined at lower burnups, these specimens are being repolished to determine if material in the fuel clad gap was possibly lost during specimen preparation
4. During sectioning of the pin F2G an orange brown colored substance was observed oozing from the fuel clad interface region of a section cut between 1 1/2 and 2 inches from the bottom of the fuel column. A sample was collected and analyzed by gamma spectroscopy. The major nuclide was Cs 137, (99.8%) but Cs-134 and Ru 103 were also present

The failed pin, F2D was removed from its capsule and sections are being prepared for metallographic and microprobe examination. In the upper portion of the fuel pin, melting and multiple cracking of the cladding was observed. The cause of the initial failure was not discovered during the examination. A portion of the failed region of the fuel pin approximately 8 to 12 inches from the bottom of the fuel column is shown in Figure 7 21

The diameter of the pin was measured at two radial positions 90° apart at one inch intervals in the unfailed areas. Figure 7 22 compares these values with the profilometer trace of pin F2G which had a very similar pre-

irradiation diameter and had similar fabrication and irradiation parameters.

Fission Gas

Fission gas punctures were made in the plenum regions of pins F2G and F2X. A volume of 168.7 cc of fission gas and helium were collected from pin F2X (83.9 % dense powder fuel) and 196.6 of fission gas and helium from pin F2G (95.5% dense fuel pellets). The majority of the fission gas from pin F2V was lost when the top end plug weld was inadvertently touched with a cutting wheel. Only 18 cc of fission gas and helium were collected from F2V. Analysis of the fission gas will be completed in the next reporting period.

The capsule of failed pin F2D contained 107 cc of Xe and Kr in the ratio of 5.95 to 1. This is only 49.5% of the amount calculated to have been formed disregarding any relocation of fuel caused by the failure. It would be expected that at least 60 to 70% of the fission gasses would have been released at the burnup attained

Discussion

Diameter profiles for the three pins as determined by dial gauge measurements are shown in Figure 7 23. The dial gauge data was plotted in this figure so a convenient vertical scale could be used. The ΔD values shown are from the more precise profilometer data. Plotted in Figure 7 23 for comparison is the peak ΔD measured on the NUMEC D-5 pin.

Figure 7 24 shows plots of ΔD versus peak burnup and smeared fuel density for pins F2G, V, and X along with data from previous GE pins irradiated in EBR II. The data obtained from the NUMEC D 5 pin is also shown for comparison. These plots show that significant increases in fuel pin diameter occurred between approximately 65,000 and 120,000 MWd/Te burnup and that fuel pin diameter change is a strong function of initial fuel density at high burnups. It is noted that between 65,000 and 120,000 MWd/Te, the diameter of pin F2G increased approximately 3.5% based on the ΔD values obtained from the pins examined at approximately 65,000 MWd/Te.

Diameter changes of all pins examined to date are plotted in Figure 7 25 along with peak ΔD values for the F2G, -V, and -X pins. This plot was used to predict the maximum diameter increases of the F2 pins previously. As can be seen the diameter increases of the F2 pins fall within the predicted band.

*The metallographic examination of samples from these pins has not been completed

**Approximately 40% of the total ΔD measured for the F2H and F2T pins examined at ~65,000 MWd/Te was due to swelling of the clad based on recent clad density measurements (see points plotted in Figure 7 24)

FAILED REGION OF PIN F2D APPROXIMATELY 8"
TO 12" ABOVE THE BOTTOM OF THE FUEL.

TRANSVERSE SECTION OF FAILED REGION OF
PIN F2D APPROXIMATELY 8" ABOVE THE
BOTTOM OF THE FUEL.

Figure 7-21. Photographs of Failed Region of Pin F2D

Figure 7-22. Comparison of Diameter Measurements from Pin F2D with Profilometer Trace of Pin F2G

The following preliminary conclusions can be drawn from the data obtained to date:

1. The diameter increase due to clad swelling in the peak flux region of pin F2X was no greater than 2.2%.
2. Initial fuel smeared density appears to have a major effect on the fuel pin diameter change at high burnup.
3. Fuel-clad fission product reaction was insignificant (<0.5 mil) at clad inside surface temperatures up to 1030°F)

7.3 LOW POWER SWELLING IRRADIATIONS

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

The interim examination of the Group F0 encapsulated fuel pins was completed at EBR-II, and the capsules reconstituted into new subassembly hardware. Capsules F0A and F0C were placed in subassembly XG03A and inserted in position 8D2 of the EBR II grid on September 14, 1970. Capsules F0B and F0D were placed in subassembly XG04A and inserted in position 8A7 of the EBR-II grid on September 15, 1970. Target burnup for the F0A and F0C capsules is 80,000 MWd/Te (8.95 at %). The F0B and F0D capsules are to be irradiated to 100,000 MWd/Te (11.18 at %). Average burnup on the capsules in XG03A (F0A and F0C) and XG04A (F0B and F0D) when reinserted in the EBR-II core were, respectively, 61,500 MWd/Te (6.88 at %) and 68,100 MWd/Te (7.62 at %). At the end of run 46A the average burnup on the peak pin in the XG03A and XG04A subassemblies was respectively, 62,500 MWd/Te (6.99 at %) and 69,200 MWd/Te (7.74 at %).

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

The nine Group F8A encapsulated fuel pins in EBR-II

subassembly X081 continued to be irradiated toward a goal exposure of 100,000 MWd/Te (11.25 at %). The average calculated burnup on the peak pin in the subassembly is 66,317 MWd/Te (7.47 at %).

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

The Group F8B capsules in EBR-II subassembly X036 were removed from the reactor core at the end of run 45A. The Group F8B capsules are to undergo an interim examination at the EBR-II and be reconstituted into a new subassembly for irradiation toward a target burnup of 50,000 MWd/Te (5.63 at %). The average burnup on the peak burnup pin in the subassembly was 42,000 MWd/Te (4.73 at %) when removed from the reactor core.

7.4 HIGH POWER INTERMEDIATE BURNUP IRRADIATIONS

7.4.1 Group F3B (Density and Stoichiometry Effects)

Six Group F3B encapsulated pins were returned to the GE Vallecitos Nuclear Center for destructive examination. EIH-21, -23, -26, -29, -30, and -33. Neutron radiographs indicate that pin EIH 33 failed. Examination of the six Group F3B encapsulated fuel pins was initiated.

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

Non-destructive examination of the seven Group F4 capsules removed from EBR-II at the end of run 44B was completed at the EBR-II site and the capsules prepared for shipment from EBR-II to the GE Vallecitos Nuclear Center. These pins have reached average calculated burnups of 73,100 MWd/Te (8.25 at %).

Figure 7-23. Dial Gauge Measurement Results for Pins F2G, -V, and -X

Figure 7-24. Correlation Between Fuel Pin Diameter and Burnup and Fuel Smeared Density

Figure 7-25. Diametral Change in Stainless Steel Clad Fuel Pins as a Result of Fast Neutron Irradiation in EBR-II

7.5 ELEVATED CLADDING TEMPERATURE TESTS

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

The X064 subassembly was removed from the EBR-II core at the end of run 44B. The average calculated burnup on the peak burnup pin in the subassembly was 50,900 MWd/Te (5.71 at.%). An interim examination was carried out on all nineteen of the capsules. Five capsules (F5A, -B, -F, -M, and -Q) were shipped to the GE Vallecitos Nuclear Center for examination. The fourteen remaining capsules will be reconstituted into new subassembly hardware for irradiation at a reactor power of 62.5 MW toward a goal burnup of 100,000 MWd/Te (11.25 at.%).

7.5.2 Group F10A, F10B, and F10C (Cladding Performance at Peak FCR Cladding Temperatures)

Preparation of the Group F10A and F10B data package draft was completed. Inspection of the nineteen Group F10A capsules indicated that the bottom weld on one capsule was not satisfactory and three capsules had low sodium levels. All four capsules were reencapsulated. Inspection of all nineteen capsules was completed.

The approval-in-principle request for the Group F10C experiment was completed and transmitted to AEC-RDT for consideration.

Work was initiated on the attachment of triangular spacers to the Group F10A and B capsules which will be located adjacent to the subassembly hex channel.

7.6 UNENCAPSULATED FUEL PIN IRRADIATIONS

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

Nondestructive examinations of the sixteen Group F3A fuel pins was completed and results are presented below:

Visual Examination

The pins were found to be in good condition when received at the GE Vallecitos Nuclear Center with the exception of several dark tarnished spots probably caused by exposure subsequent to disassembly of the bundle. The surface of the pins was observed to be bright and shiny when the subassembly was disassembled at EBR-II.

Gamma Scans

Gross gamma scans were made of all sixteen pins to determine the axial distribution of activity. Typical gross activity scans of pins #4 and #8 are presented in Figure 7-26. In almost all cases Nb-95 was found to be the nuclide producing the majority of the activity in the peaks scanned.

Neutron Radiography

Neutron radiographs were taken of each pin and showed that central voids were present in each fuel column and that in most pins the fuel pellet interfaces were discernible.

Diameter and Length Measurements

Continuous diameter profile scans were made along the length of each fuel pin at four orientations 45 degrees apart and while rotating. Typical results of these scans accompanied by original pellet line-ups for a 304L and 316 clad pin are presented in Figures 7-27 and 7-28. The profiles indicated that significant diameter changes occurred in the region of each pin exposed to the EBR-II core. The gas plenum sections of the fuel pins, which were above the reactor core, generally were uniform, with slight diametral variations typical of the as-fabricated tubing. These results indicate maximum diametral increases of ~0.4% for the 316 clad fuel pins and ~1.2% for the 304L clad pins.

Length measurements indicated increases of ~0.22% for the 316 clad fuel pins and ~0.66% for the 304L clad fuel pins.

Cladding density will be measured under Task L on selected samples to verify whether these differences were caused by different creep behaviors or different rates of irradiation induced void formation.

Examination of the profilometer traces of the F3A pins provides at least a partial explanation of the difference in diameter change between the 304L and 316 clad pins. Careful examination of the traces indicates a periodicity of peaks and valleys along the fueled length of the pins. Correlation of these peaks and valleys with the position of the wire wrap of pins #4 (316 clad) and F3A-8 (304L clad) were made from photographs of the pins. Shown in Figures 7-27 and 7-28 are the points where the wire wrap passed through the plan in which the 0° orientation profilometer traces were taken on pins #4 and #8. The position of the wire is indicated by the vertical "t"s located along the profile traces. The dotted lines indicate the general direction in which the wire wrapped around the pins. These lines could not be accurately located because of variations in the pitch of the wire as it was wrapped on the pins. This is the first correlation of wire wrap pitch to low points in the clad profile. The previous interim examination of these pins was done with the wire in place and provided no opportunity for this type of measurement. All previous profilometer traces under this program were done on double contained capsules. The profile traces from these pins showed no obvious correlation between the spacer wire and clad profile.

If the 0° and 90° profilometer traces and 45° and 90° diameter profiles are compared a reasonable correlation between peaks and valleys in the traces is observed. This correlation indicates that either the wire wrap or pin-to-pin interactions through contact with the wire and subassembly hex are significant enough to cause the pins to deform into oval cross sections, i.e. the pin is assuming a slight corkscrew shape. This phenomena is illustrated in Figure 7-29. This figure illustrates the effect of a spirally wrapped wire on the 0° orientation plane of a fuel pin assuming the wire restrains any change in diameter. Comparison of the 304L

Figure 7-26. Gross Gamma Activity Profile of Group F3A Pins #4 and #8

Figure 7-27. Profilometer Traces of Type 304L Clad Group F3A Fuel Pin #8 Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters

Figure 7-28. Profilometer Traces of Type 316 Clad Group F3A Fuel Pin #4 Compared to Pre-Irradiation Fuel Pellet Lengths and Diameters

Figure 7-29. Schematic Representation of Effects of Wire Wrap on Pin Ovality

and 316 clad pins indicates that the difference in diameter between pins clad in each clad cannot be entirely attributed to ovality differences. Several 304L clad pins exhibited diameters greater than the 316 clad pins in all orientations measured. Analysis of these differences is continuing.

Destructive Examination

Destructive examination of pin #8 was begun and examination of pins #1, #2, #3, #4, and #5 will follow. These six fuel pins were selected in order to allow various comparisons among the following parameters: annular pellet vs solid pellet, 1.98 vs 2.00 O/M values, and 88% vs 82% smeared densities. They also include the peak power and peak temperature fuel pin, F3B-3, which will be examined for fuel-clad reaction.

The initial step in the destructive examination of these fuel pins is the puncturing of the pin and subsequent measuring of the total volume of free gas in each pin. This was completed on three of the fuel pins, #1, #2, and #7. Pin #7 was punctured at the top in its plenum region, which resulted in a rapid release of fission gas. Essentially all of the gas was released by the time the first pressure reading was made approximately one minute after puncture. Pins #1 and #2 were punctured in the region of the bottom insulator. This resulted in a much slower release of fission gas as shown in Figure 7-30. After three hours pressure was still building up in the collection bomb used for pin #2, indicating that fission gas was still being released through the puncture hole. Continued work is planned in this area for the remaining fuel pins. Punctures will also be made in areas such as the fuel region and upper insulator to attempt to define possible effects of position of pin failure on fission gas release.

Metallographic Examination

Metallographic examination of pin #7 was completed. Figure 7-31 shows a transverse section taken from the peak flux region of the pin, where the end-of-life linear power was ~14.9 kw/ft and the clad I.D. temperature was ~930°F. The included 500X photo of the fuel-clad interface shows no indication of fuel-clad interaction. Examinations of additional areas of the fuel pin revealed no unusual occurrences.

7.6.2 Groups F9A, F9B, F9C and F9D (Unencapsulated Fuel Pin Irradiations at High and Medium Powers to High Burnups)

At the request of ACE RDT a memorandum providing comments and recommendations concerning the AEC RDT and EBR II Projects proposal for continuing irradiation of the Group F9A and F9C unencapsulated fuel pins in Mark E subassemblies was prepared and submitted to AEC-RDT. The F9A and F9C subassemblies were removed from the EBR II core on May 18, and July 16, 1970,

respectively. The subassemblies were stored in the EBR-II fuel storage basket until October 7, 1970, pending decisions on.

- 1) Whether sodium drainage from the Mark F-37 subassemblies in which the pins were contained was sufficient to permit cleaning the subassemblies using the standard EBR-II cleaning procedure without risk of damaging the fuel pins due to a possible explosive reaction between water and sodium remaining in the subassembly interstices, or
- 2) Whether a greater (or lesser) risk to the pins existed by disassembling the subassemblies without the standard water wash and then recleaning the fuel pins.

Three mockup test assemblies were placed in the reactor tank then removed and cleaned using the standard EBR-II cleaning procedure to provide an assessment of the risk involved with regards to an explosive sodium-water reaction. All three tests gave negative results. Based on these tests a decision was made to clean the subassemblies using the standard EBR II cleaning procedure. The subassemblies were removed from the reactor, cleaned, and disassembled without incident. All pins were in good condition and their interim examination initiated.

Irradiation of Groups F9B and F9C in EBR II subassemblies X062 and X058 continued. The peak burnup pins in the X062 and X058 subassemblies have reached average calculated burnups of 44,900 MWd/Te (5.04 at %) and 44,000 MWd/Te (4.95 at %), respectively.

7.6.3 Group F9E (High Power, High Burnup, Unencapsulated Fuel Pin Irradiation Using Grid Spacers)

Thirty-seven Group F9E fuel pins were loaded and welding of the top end plugs was completed. Final inspection of the pins is in progress.

Drawings of the flow test and in-reactor grid spacer subassemblies were completed. Procurement of the flow test subassembly hardware was initiated and fabrication of the parts completed. Fabrication of dummy pins for the flow test was completed.

A preliminary draft of equipment and interim examination requirements for the Group F9E examination was prepared and forwarded to AEC RDT at their request.

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

The wire wrap deflection fixture was completed and measurements were made on the wire wrap of dummy pins. The dummy pins were wrapped with wire put under tension by varying amounts of dead weight during the wrapping operation. As a result of these measurements, it was determined that the deflection of the wire could not be correlated to the dead weight used for wrapping unless there was a difference of approximately four pounds in the dead weight.

Figure 7-30. Rate of Fission Gas Release from Group F3A Pins #1, #2, and #7

25 X

500 X

Figure 7-31. Transverse Section of Group F3A Pin #7 Taken from Peak Flux Region
(Fuel-Clad Interface is also shown)

Each of the 37 pins in the group was assigned a 4, 8, 12 or 16 pound wire load to be used during the wire wrap step.

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

Fabrication of capsule F12B was suspended. Evaluation of bids from transducer manufacturers and the costs required to modify the capsule design to use different transducers than originally intended indicate that the cost of the capsule would be doubled. Based on this evaluation the experiment was judged too costly for the technical return to be obtained.

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

The interim examination of fuel pins F12P and F12Q was completed and the pins reencapsulated for additional irradiation in the GETR. Current burnup on the fuel pins is ~55,000 MWd/Te.

The diameter of pin F12P (15 mil wall clad) was found to have increased 1 to 1.5 mils since the previous examination at ~40,000 MWd/Te. The diametral change for this pin at ~40,000 MWd/Te was ~1.5 to 3.0 mils greater than at the start of the irradiation.

Pin F12Q had not increased in diameter within the accuracy of the measurement, ± 0.0002 inch, since its examination at ~40,000 MWd/Te.

7.9 CHARACTERIZATION OF CAPSULE FUEL PIN HARDWARE

An essential requirement in the design and implementation of fuels and materials irradiation experiments is the maintenance of complete and accurate histories on all materials used. The response of a material to irradiation can only be obtained by first knowing the precise condition of the material prior to irradiation. The characterizations that are performed on incoming materials serve this function. Items in the characterizations include chemical composition, structural condition, dimensional properties and mechanical properties. Details of the materials characterization procedures established for BRDO have been discussed previously.⁽¹⁾

Two materials evaluations were completed during the current quarter: a quality characterization of a single lot of austenitic stainless steel tubing procured for use as fuel cladding and a characterization of two lots of austenitic stainless steel bar procured to provide end plug material for fuel pins. The characterizations performed and the results established are summarized briefly below.

K3410G Tubing

The material procured under Purchase Order K3410G consists of 1000 feet of cold worked tubing of Type 316H stainless steel produced by the seamless process from vacuum-melted raw stock. The tubing — a single lot of 0.250-inch diameter, 0.015-inch wall material, cold reduced 20 percent — was manufactured to specifications established by BRDO for nuclear grade cladding. The quality characterization performed on representative samples of the lot showed that the material conforms in all respects to the terms and provisions of the ordering specification. Chemical analyses performed by the steelmaker, the tube vendor, and a third laboratory selected by BRDO yielded mutually confirming results. The levels of all elemental constituents are well within those specified for Type 316H. Results of the metallographic examination showed this material to have a cold worked microstructure of a magnitude comparable to previous 20 percent cold worked tubing. The material is extremely clean.

Mechanical properties tests of selected samples showed that 20 percent cold working imparted an approximately 50 percent increase in yield and a 25 percent increase in ultimate tensile strength over that of comparable annealed material. Elongation, on the other hand, showed reductions ranging from 60 to 75 percent.

K3524G Round Bar

Two lots of round bar — one each Type 304 and Type 316 stainless steel — were received on Purchase Order K3524G. The 500-pound lots, each representing approximately 1800 feet of 5/16-inch diameter bar, were custom produced from raw stock to assure conformance to the ordering specification for nuclear quality bar. Chemical and metallographic characterizations performed on representative samplings of each lot showed that the materials do in fact represent high quality bar. The chemistries of both materials lie well within the levels specified; the metallography revealed very equiaxed microstructures and inclusion contents within the maximum ranges allowed. Both materials were accepted for use in experimental fuel assemblies.

7.10 TASK F QUALITY ASSURANCE

The formulation of a Quality Assurance Plan for fuel pin fabrication was initiated.

An audit of the Task F fuel pin quality assurance and quality control was conducted by an AEC-RDT Team on October 12-13, 1970.

(1) T. Lauritzen, J. J. Albanese, G. C. Berg, G. G. Gaul, GEAP-13553, December 1969.

7.11 NEUTRON RADIOGRAPHY

Fabrication of neutron radiography standards for use at EBR-II was initiated. The purpose of the standards is to enable accurate measurements of irradiated fuel dimensions to be made during interim and final examinations using neutron radiography techniques.

Four standards are being fabricated. Two of the standards will be unencapsulated pins and two will be encapsulated. The variables included in the standards are U-235 content, fuel density, central void size and pellet diameter.

7.12 AUTOMATION OF PROFILOMETRY

Work to automate the digitizing of profilometry, previously described in Section 12.2.2 of the 35th Quarterly Report, was completed. It was determined that the instrumentation tape recorder being used to record the signal from the profilometer is too difficult to operate and prone to malfunction to be used routinely by the profilometer operator.

7.13 CORRELATION OF FAST FLUX FUEL PIN DATA

Direct immersion density measurements of sections of cladding from pins F2H and F2T, (reported under Task L), indicate that the clad swelling component attributed to the two pins in the correlation displayed in Figure 12-4 of the 34th Quarterly, was excessive by about 0.5 mils for F2H and 1.0 mil for F2T. The new data increases the scatter in this correlation which tests the hypothesis that all the diametral increase for these pins is due to the combined effort of differential fuel-clad thermal expansion at the first startup of a pin plus subsequent swelling of the cladding material. It is believed that the scatter of the data is still within the error band caused by uncertainties in parameters used to calculate the thermal expansion interference at startup. Thus the assertion that the diametral increases in these two pins resulted only from the startup interference plus clad swelling has been rendered less convincing but has not been disproved by the new data.

Analysis was carried out on data from the four high burnup Group F2 pins reported elsewhere in this section. Two of the three intact pins, F2G and F2V, exhibit large amounts of diametral increase. If the relatively small diametral increase of F2X represents the clad swelling component for the other two pins, these two pins exhibit

diametral increases greater than predicted by the above model (i.e. that all the ΔD can be attributed to the initial startup interference of fuel and cladding plus subsequent swelling of the cladding). This model, reported in section 12.3 of the 34th Quarterly Report, was successful in correlating most of the lower burnup fast flux data. The large diametral increases observed in the two PA-10 pins and in the NUMEC D-5 pin suggest that additional mechanisms may be operative in these pins.

The following observations were made after a review of the F2G, V, and -X microstructure and profilometer data.

1. F2G and F2V pins exhibited radially oriented cracks at the periphery of the fuel. The presence of this type of crack makes it doubtful that the diameter increase was caused by steady-state fuel swelling of the outer annulus of fuel.
2. The evidence of a center void in these pins implies that fuel swelling in the high temperature region was insufficient to collapse the center void or to produce clad strain.
3. A second phase of fuel material was observed in the microstructure of F2G, at the center of the fuel. What is apparently the same material partially fills some of the radial cracks in the peripheral fuel. The presence of this material may indicate a higher linear power that occurred fairly suddenly expelling the material outward into the cracks.
4. The highly peaked, symmetrical shape of the profilometer curves for these pins is indicative of a relatively high strain rate yielding of the cladding rather than low strain rate flux induced or thermal creep.
5. The magnitude of the diametral increases with increasing fuel density.

It is hypothesized that clad strain produced during power changes may be more consistent with these observations than the steady load of fuel swelling. For instance, an increase in average fuel temperature resulting from a power increase would liberate previously retained fission gas to the central void. If there was insufficient time for the gas to migrate to the plenum, and if insufficient void volume were available to accommodate this increased gas volume without exceeding the cladding yield strength, strain of the cladding would result to provide the needed volume. This and other mechanisms are being studied.

8. TASK G-6—CORE DESIGN ANALYSIS

8.1 GENERAL

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 Summary Description of the Conceptual Design of a Grid Spaced 1000 MWe Fuel Assembly for Test in an Open Loop of the FFTF

The experimental fuel assembly sketched in Figure 8-1 and summarized in Table 8-1 has 169 pins at a pitch-to-diameter ratio of 1.25 and a pin O.D. of 0.250 inch. With this geometry, 169 pins are the maximum number that will fit into an FFTF channel. A bundle of this size will provide test data that are representative of the generally larger LMFBR bundles. In this sense, the grid-spaced fuel assembly described is a simulation of the larger 1000 MWe LMFBR fuel assembly. The channel is a standard FFTF channel with a liner inserted to reduce the inside dimension; this change was necessary because the pitch in the test assembly (0.3125 inch) is larger than that in the FFTF reference design (0.286 inch). The difference is such that the test assembly must contain one ring of pins fewer than the FFTF reference assembly. This change in the number of pins would leave a large gap between the outer pins and the channel wall if a liner were not present. The pin-to-liner spacing shown in Table 8-1 was selected so that thermal gradients on the outer pins are minimized.

The fuel pin dimensions are representative of a 1000 MWe Plant design, except that the lower blanket region of the test pin is 8 1/2 inches shorter than that of the 1000 MWe plant design. This change was made so that the active fuel region of the test pin would occupy the correct axial location in FFTF. The 8 1/2 inches that was removed at the bottom of the pin was added to the plenum region at the top of the pin, this maintains a representative overall fuel pin length of 92 inches.

The irradiation is based on a maximum fluence of 2.8×10^{23} (>0.1 Mev) for the 1000 MWe fuel clad. Because of the difference in fluence to burnup ratio in FFTF, this results in a 100,000 MWd/Te maximum burnup in the test assembly when using 1000 MWe fuel loading.

The principal result of this study is the definition of the wave beam grid dimensions. The beam material thickness is 0.012 inch and the effective beam height (between pin-grid load points) is 0.350 inches, giving a height-to-thickness ratio of 29. This design is based on a fuel pin lateral deflection, resulting from all loads and including creep, of 5% of the linear distance between rods (5% of 0.0625 inch). These beam dimensions are consistent with current fabrication experience and the material thickness is compatible with the straight-through slot between fuel pin rows.

An assembly drawing of the test assembly appears as Figure 8-1. This figure also shows the axial locations of grid supports with respect to the fueled section of the assembly. Figure 8-2 shows the fuel pin, the special features of which have been described above. Fuel pins for a 1000 MWe plant that have been designed with a 0.010 inch wall thickness normally have a fission gas vent device to prevent excessive buildup of gas pressure. Details of this device have not been shown.

Plan and elevation views of the grid structure appear as Figure 8-3. This figure shows each row of grid beams to be angularly displaced 60° with respect to the grids above and below it. Figure 8-4 is a detail drawing of a wave beam and its attachment to the channel liner. Attachment is made by means of tabs located on the ends of each beam; these tabs are 80 mils long and 0.125 inch high, and are bent about 60° from the plane of the wave beam. As shown in Figure 8-4, each tab is designed to snap into a recess which has been formed in the liner wall. The dimensions of this recess are such that small lateral and axial displacements of the grid structure can be accommodated by movement of the tabs. Gross movement of the grid is restrained, however, since the tabs are securely locked into their recesses by the action of the tab as it is restrained by the heel of the recess. The presence of the fuel pins in the grid also serves to prevent gross disassembly movement of the tabs out of the recess. An advantage of this method of attachment is that the clearance between the tab and its recess is used to accommodate axial differential movement between the wave beam and the liner. In addition, bending of the tab can accommodate some of the lateral differential swelling.

The grid structures are assembled by projection welding the upper set of wave beams to a lower set. In the welding process, a certain amount of egg-crating is produced between the upper and lower beams. The welds are intended to hold the assembly together so that it can be inserted into the channel for assembly. The welds also act as

Table 8-1
SUMMARY DESCRIPTION OF FUEL ASSEMBLY GEOMETRY

Parameter	Value
Fuel Assembly Pitch (inch)	4.715
Fuel Assembly Outside Distance Across Flats (inch)	4.615
Fuel Assembly Inside Distance Across Flats (inch)	4.335
Inside Distance Across Flats of Liner (inch)	4.109
Channel Wall Thickness (inch)	0.140
Liner Thickness (inch)	0.113
Channel-Liner Gap (inch) (Liner is used to reduce I.D. of existing FFTF Channel)	0.0
Fuel Pin-to-Liner Space (inch)	0.035
Pitch-to-Diameter Ratio	1.25
Fuel Pin O.D. (inch)	0.250
Clad I.D. (inch)	0.230
Number of Pins in Assembly	169
Length of Fuel Region (inch)	36.0
Length of Lower Axial Blanket and Insulator (inch)	6.50
Length of Upper Axial Blanket (inch)	15.0
Overall Pin Length (inch)	92.0
Total Fuel Assembly Length (feet)	12.0
Flow Area of Central Cell (inch ²)	0.0355
Hydraulic Diameter of Central Cell (inch)	0.181
Total Fuel Assembly Flow Area (inch ²)	6.330
Axial Spacing of Grid Spacers (inch)	6.0
Total Spacer Beam Height (inch)	0.500
Effective Spacer Beam Height (inch)	0.350
Spacer Beam Thickness (inch)	0.012
Apparent Beam Thickness (inch)	0.022

Figure 8-2. Test Assembly Fuel Pin

Figure 8-3. Wave Beam Spacer

Figure 8-4. Wave Beam Spacer Locking Tab Design

pivots, allowing partial collapse of the grid structure during insertion. Once the grids are inserted opposite their respective tab recesses the grids are re-expanded and the pins are inserted into the grids. The pins lock the tabs in position and the welds no longer are required for structural integrity. On this basis, weld failure during irradiation is of no concern.

The fuel used in the test assembly is representative of 1000 MWe Demonstration Plant fuel (Ref. 1). This choice results in linear power densities that are closer to the average than to the maximum power densities expected in LMFBR cores. Specifically, the power density of the test assembly fuel in FFTF is about 13% lower than that of FFTF driver fuel operating at the same location as the test assembly.

8.2.2 Assembly Operating Characteristics

The operating characteristics of the test assembly are described in Table 8-2. Two sets of parameters are shown, corresponding to the two possible test locations in the FFTF core.

The coolant temperatures and thermal gradients shown in Table 8-2 are the result of the thermal-hydraulic analysis performed as part of the design of this assembly. Two types of fuel pins are described in Table 8-2; these are the "corner pin" and the "edge pin." Both pins are located adjacent to the hottest flat edge of the hexagonal channel. The corner pin lies at the corner of the hexagon and the edge pin lies at the center of the flat side. The local temperature difference (ΔT) is computed as the change in coolant temperature around 60° of pin circumference. The temperature differences appearing in Table 8-2 are maximum values.

Table 8-2 summarizes the radial and axial variations of coolant temperature by giving axial maximum and radial average values of coolant temperature. These values are based upon a coolant inlet temperature of 600°F; they will change during the lifetime of the test as the reactor approaches advanced core operation. (It is assumed that the test assembly is loaded in the first core of FFTF; core conditions will change as the equilibrium cycle is approached.)

The pressure drop data presented in Table 8-2 include only the contributions of fuel pin friction and grid spacer losses to the total pressure drop. The heat transfer coefficients shown in the table are drawn from Reference 2. The times required to reach 100,000 MWd/Te burnup in each of the two possible core positions are approximations. Corrections were made for the loss of fissile material during irradiation and for conversion of U-238 to Pu-239.

8.2.3 Fuel Pin Bending

Fuel pins operate for long periods of time in an environment where both the neutron flux and the coolant temperature vary spatially. The LMFBR environment results

in the deformation of structural members (cladding, grids and channels) by thermal expansion, irradiation-induced swelling, thermal creep, and irradiation-induced creep. The swelling and creep correlations used are for the PNL cold worked type 316 stainless steel.

The radial neutron flux and the coolant temperature gradients tend to reinforce or cancel each other, depending on the position in the assembly. In general, a corner pin nearest the center of the core will have the worst combination of environmental conditions.

The direct effect of these transverse gradients is that the fuel pin tends to bow. If unrestrained, the pin would bow with a large radius of curvature. For a corner pin, the bowing is in the direction of the core center. The fuel pin is restrained by a set of grid structures located six inches apart along its length. This causes the assembly of pins to act as a set of short, coupled, indeterminant structures.

In general, the lateral deformation that results from the imposed gradients consists of alternating deflections between successive grid supports. Thus the pin assumes a roughly sinusoidal shape. This pin deflection also results in bending stresses and lateral loads on the grid support structures. Since the pin operates to a high fluence, irradiation and thermal creep occurs which tends to reduce the stresses in the rod while increasing the deflections of the grids and increasing the grid restraint loads.

The analysis of the pin bending has been performed using CRASIB I⁽³⁾ which is a finite element procedure for the two-dimensional analysis of the creep of a statically indeterminant beam. The pin model is shown on Figure 8-5. The model simulates the 16 grid locations and includes 6 additional deflection nodes in intermediate positions between grids in the core area.

A series of CRASIB I problems was run using this model for an irradiation time of 19,200 hours. The reference case includes nominal swelling and creep. The grid loads and intermediate deflections are tabulated in Table 8-3 for time zero and for the end of irradiation. The maximum initial grid load is 0.117 lbs. and the maximum final grid load is 0.417 lbs.; these loads occur at nodes 2 and 15, respectively. The maximum initial deflection is -0.05 mils (at node 3), and the maximum final deflection is 1.45 mils (at node 9). The maximum fuel pin stress is 14,192 psi at node 10. In order to evaluate the uncertainty for irradiation-induced creep, CRASIB I cases were run for 50% and 125% of the nominal creep values.

It was found that rod deflection is relatively insensitive to the limits of creep, while pin stress and grid load vary almost inversely with the extent of creep. A 50% reduction in creep doubles both stress and load. Creep strains in fuel cladding at 19,200 hours are 0.29%, 0.34%, and 0.35% for 50%, 100%, and 125% of nominal creep.

In order to evaluate the effect of creep in more detail, a case was run with swelling and thermal creep only.

Table 8-2
TEST ASSEMBLY OPERATING CHARACTERISTICS IN FFTF CORE

<u>Parameter</u>	<u>Row 2 Value</u>	<u>Row 4 Value</u>
Maximum Coolant Inlet Temperature (°F)	600 (900)*	600 (900)*
Average Coolant Outlet Temperature (°F)	900 (1200)*	900 (1200)*
Maximum Hot Channel Coolant Temperature at the Top of the Fuel Region (°F) (including uncertainties)	949	949
Maximum Mid-Clad Temperature Averaged over 60° of Arc (°F)	927	930
Thermal Bowing ΔT Average Central Pin (°F)	1	3
Maximum Thermal Bowing ΔT Edge Pin (°F)	30	30
Maximum Thermal Bowing ΔT Corner Pin (°F)	33	32
Maximum Local Circumferential ΔT Corner Pin (°F/60° Arc)	51	50
Maximum Local Circumferential ΔT Edge Pin (°F/60° Arc)	30	29
Total Coolant Flow in Assembly (lb/hr)	151,000	133,000
Coolant Velocity in Central Flow Cell (ft/sec)	18.5	16.4
Average Mass Flux (lb/hr ft ²)	3.43 x 10 ⁶	3.02 x 10 ⁶
Film Heat Transfer Coefficient at 900°F for a Central Flow Cell (Btu/hr ft ² °F)	37,950	37,300
Maximum Power Density of Hottest Pin (kw/ft)	9.57	8.61
Maximum Power Density of Central Pin (kw/ft)	9.38	8.23
Maximum Power Density of Coldest Pin (kw/ft)	9.19	7.85
Total Fuel Assembly Power (Mw)	3.83	3.37
Maximum Heat Flux for Hottest Pin (Btu/hr ft ²)	5.0 x 10 ⁵	4.5 x 10 ⁵
Fuel Bundle Pressure Drop (psi) (Pin Friction and 16 Grid-Spacers only).	34	27
Time at Full Power to Achieve 100,000 MWd/Te at Axial Midplane of Lead Pin (Days)	710	800
Atom Percent Burnup at 100,000 MWd/Te (%)	13.5	13.5
Total Fluence in Lead Pin at 100,000 MWd/Te (n/cm ²)	4.28 x 10 ²³	4.28 x 10 ²³

*Values in parentheses are estimates of advanced core operation

Table 8-3
LOAD AND DEFLECTION SUMMARY FOR A CORNER ROD VS. CREEP MAGNITUDE

WADCO COLD WORKED 316SS																	
SWELLING + CREEP																	
CORRELATIONS	(lb)	0.00009	0.147	0.141	0.0510	0.0796	0.303	-0.324	-0.143	0.417	0.0733	0.0185	0.00025	0.00053	0.000009	0.00003	0.0000024
TIME - 19200 HOURS	(mil)		0.914	0.368	0.077	1.45	0.981	0.554									
<u>CASE A</u>																	
100% CREEP	(lb)	-0.002	0.117	0.0183	0.0434	0.0234	0.0450	0.0405	-0.0856	0.0771	0.0459	0.0181	-0.000047	0.000012	-0.000003	0	0
TIME - 0 HOURS	(mil)		0.062	0.037	0.013	0.04	0.003	0.031									
THERMAL ONLY																	
<u>CASE B</u>																	
50% CREEP	(lb)	0.00025	0.276	0.275	0.0907	0.172	0.606	0.606	0.141	0.741	0.119	0.025	0.00046	0.00051	0.000021	0.000026	0.000003
TIME - 19200 HOURS	(mil)		0.923	0.350	0.108	4.34	1.035	0.436									
<u>CASE C</u>																	
125% CREEP	(lb)	0.00007	0.119	0.113	0.0416	-0.063	0.242	-0.261	-0.126	0.339	0.061	0.016	0.00019	0.00053	0.000007	0.00003	0.000002
TIME - 19200 HOURS	(mil)		-0.927	-0.381	0.0733	1.484	0.982	0.578

0.250 O.D.
0.220 I.D.

Figure 8-5. CRASIB I Model

This resulted in a maximum grid load of 3.1 pounds, a fuel pin bending stress of 84,790 psi, and a maximum pin deflection of 5.47 mils. A subsequent calculation (on 15 mil cladding) eliminated the thermal gradient without changing temperature, flux, or flux gradient. The case was rerun for nominal creep and resulted in a maximum pin bending stress of 1000 psi, a maximum deflection of 0.07 mils, and a maximum grid load of 0.025 pounds. This shows that differential thermal expansion of the pin and clad differential swelling due to the temperature gradient are the predominant effects and that the flux gradient plays only a minor role in distortion of the pin.

8.2.4 Local Pin Stresses

Local stresses are produced in the fuel clad due to local bending at the grid supports. The clad thickness falls outside the range that could be considered either thin-wall or thick-wall tubing. A conservative estimate of local stress can be made by considering the clad to be thin-wall tubing and by using an empirical formulation for the stress distribution (Reference 8-4).

The maximum value of the stress concentration factor occurs under the load; its value is 1.9 for longitudinal stress and 2.2 for circumferential stress. For our nominal creep case this results in a longitudinal stress of 8000 psi and a circumferential stress of 9240 psi. These stresses reduce to 10% of maximum within 0.020 inches of the point of application of the load.

8.2.5 Vibrations

Fuel pins subjected to fluid flow parallel to their axes exhibit continuous random lateral vibration. This vibration is self-induced through interaction between pins and fluid, and generally occurs near the natural frequencies of the fuel pin. The largest displacement amplitudes occur at the lowest natural frequency. Although the pin stress, grid loads, and deflections occur at discrete points along the pin, they have been plotted on Figure 8-6 as continuous functions of axial position. The frequency is also shown as a function of axial position. Pin frequency varies only slightly with position as a result of temperature variation. Grid load, pin stress, and deflection have larger variations and they generally decrease as flow proceeds up the assembly. This is primarily the result of flow smoothing that occurs as the flow proceeds through the assembly.

8.2.6 Grid Structural Analysis

The wave beam grid spacers described in this conceptual design are assumed to accommodate relative pin-to-grid swelling primarily through flattening of the wave section of the beam, (Figure 8-7a), with some (~10%) distortion of the beam in the transverse direction, (Figure 8-7b). As shown in this figure, the loading of a typical section of the wave-beam is in the form of point contacts on a finite flat plate; however, analysis of this flattening

mechanism has shown that a typical wave beam section may be analyzed as a uniform beam, redundantly supported at the contact points. On this basis it is relatively easy to define and investigate the parameters most affecting the operational characteristics and design of the wave-beam grid, i.e., its thickness and height.

A. Elastic Model

The elastic response of the wave-beam was based on this response of a uniform beam and includes: (a) wave-beam spring constant, (b) peak bending stress per unit load, and (c) peak bending stress per unit deflection. The results of this are shown in Figures 8-8 to 8-10.

B. Creep Model

Wave-Beam-Grid creep distortion during the application of long-term loads can be extremely important, since it directly affects the pin-to-pin or pin-to-channel spacing within the bundle. Normally this deformation could be neglected since the majority of the grid spacers will be operating at 900°F or less, which is below the temperature at which thermal creep is significant. However, since the discovery of irradiation-induced creep, which is greatest at the lower inlet temperatures, this is no longer true.

As with the elastic analysis, a uniform beam model has been used to evaluate the effects of long-term creep. This included both the deflection from a constantly applied load and the creep relaxation due to a unit displacement.

C. Pin Bowing Loads

The grids must always be capable of supporting the loads imposed by pin bowing; thus defining the lower limit of grid geometry becomes limited by creep deflection considerations of the grid support which, if unlimited, would result in severe reductions in the pin-to-pin or pin-to-channel spacing.

The elastic deflection at end-of-life was computed based on the creep deflection model by integrating the effects of a time varying load. The resulting procedure was followed at each grid location and included the variations in beginning, mid, and end-of-life load, fluence, and temperature which exist at each point.

For the peak deflection point, grid point 10, the deflection for various grid geometries is shown in Figure 8-11. This includes the effects of 50% and 125% of the reference irradiation creep value. As shown, however, there is very little effect due to this variation since the grid loads at this point also vary in a proportional fashion.

A reference grid geometry of 0.350 "x 0.012" (h/t = 29) was chosen to limit the creep deflection of the outer pin rows to approximately 5% of the pin-to-channel clearance.

D. Pin-to-Grid Swelling

While certain components of the core must be designed to accommodate the total swelling which has been

Figure 8-6. Fuel Pin Vibration Predictions

Figure 8-7. Grid Beam Structural Model Used for Conceptual Design

Figure 8-8. Wave-Beam Grid Design Spring Constant

Figure 8-9. Wave-Beam Grid Design Stress/Unit Load

Figure 8-10. Wave-Beam Grid Design Stress/Unit Deflection

Figure 8-11. Grid Spacer — Deflection vs. Height to Thickness Ratio with Irradiation Creep as a Parameter

predicted to date, the grids are only required to accommodate that amount of relative swelling which occurs between the fuel pins, grid structure, and channels.

Considering a unit cell within the spacer, the growth of the grid structure itself is expected to be the sum of irradiation-induced and thermal expansion. For the analysis, the relative swelling between the fuel rods and grid structure was taken as the sum of the thermal growth differential and the irradiation swelling differential caused by differences between the clad and grid temperatures. No fluence difference was used due to the close proximity of grid and pin.

The resulting relative pin-to-grid growth becomes very strongly dependent on both linear power and fluence level with the result that the peak relative growth occurs near the peak power point in the pin. The results of this work are summarized on Figure 8-12.

As swelling occurs, the resulting grid distortion produces a reaction load between the pin and grid. If no creep were occurring, this reaction would merely be related to the grid distortion in an elastic manner. This is not the case, however, and the effects of creep will play an important role with the peak reaction occurring during a period of maximum growth rate not total growth.

With pin-to-grid growth related primarily to irradiation swelling, peak reaction loads will occur at end-of-life due to the ever increasing pin-to-grid growth rate. This growth rate which equals the deflection rate of the grid was used to determine the reaction of the grids. The results of this work are shown in Figure 8-13 as a function of axial position and include the variations in grid temperature, fluence, and linear power. As shown in this figure, the resulting grid reaction peaks just above the center of the core near the point of peak relative growth.

E. Grid Expansion

The undercooled nature of the channel compared to the majority of the internal pins and coolant will result in differential grid expansion relative to the channel. This differential expansion will swell the grid beams to a larger extent than the channel at any axial position. For the longest beam in the grid, this differential growth was computed to be 0.027 inches. This value includes both the effects of thermal and irradiation swelling and is found to peak just beyond the core mid-plane.

Fully restrained, differential growth would result in substantial axial loads and stresses on the grid beams. However, design of the grid-to-channel interface will provide clearance to prevent interference near end-of-life and to prevent either grid or local channel deformation, thus not contributing to stresses in the grid structure.

F. Axial Thermal and Fluence Gradients

The grid spacer structure will distort due to swelling gradients in the axial direction caused by both thermal and

irradiation effects. Assuming that the local grid temperature is equal to the coolant temperature at that location, this results in an axial thermal gradient of approximately 8°F/inch for FFTF.

Considering these gradients by themselves, they represent a negligible loading source on the grid, resulting in insignificant deflections and bending stresses. However, when considered in conjunction with the effects of irradiation swelling, these thermal gradients combined with the axial fluence gradient can result in significant loadings on the grid beams.

As shown in Figure 8-14, these gradients are additive in the lower half of the core where all gradients are positive. However, the peak gradient occurs at the top of the core since the fluence portion of the total gradient predominates.

G. Axial Pin Growth

With the grid spacer attached to the channel, significant relative motion can occur between the grid and fuel pin due to the under-cooled nature of the channel relative to the hotter fuel pin cladding and due to irradiation — thermal distortion of the grid itself.

Including the effects of creep, the peak normal load near end-of-life was found to be 0.23 pounds which, with a coefficient of friction of unity, will equal the axial grid load per contact point. This load is typical of most internal cell locations where thermal bowing gradients are small. At the edge row of pins, the normal grid loads can be higher; however, they are not typical of the majority of the cells within the grid and thus have not been considered.

The longest top layer, corner to corner, beam will see a total axial load from 29 pins. For this analysis it further has been assumed that the total axial load was not only due to pins in contact with it; but also includes a contribution from the lower beam level. Thus the total load for this beam is then given as 13.4#, resulting in a bending stress in the reference beam geometry (0.350 x 0.012) of 15,900 psi.

This stress is classified as a secondary bending stress since small deflections will reduce its magnitude, i.e. it is self-limiting.

8.3 STEADY STATE AND DEFECTED FUEL IRRADIATION TESTING NEEDS

A topical report on this subject has been prepared and is undergoing final review prior to release as a GEAP. The general contents are indicated below:

- Fuel design range of interest
- Economic considerations in the design of LMFBR fuel
- Review of actual and planned UO₂-PuO₂ fast flux irradiation
- Analyses of irradiation test results

Figure 8-12. Pin-to-Grid Growth

Figure 8-13. Pin-to-Grid Swelling Reaction Grid Height (0.35 in.)

Figure 8-14. Contributions to Axial Gradient

- Tests to permit extrapolation of irradiation results to LMFBR design conditions
- Defected fuel pin testing
- Fuel testing for safety and licensing—normal operation
- Fuel pin thermal performance testing needs
- Internal cladding attack
- Fuel bundle testing
- Priorities test recommendations

8.4 FUEL ELEMENT FAILURE, COOLANT BOILING AND OPERATING SAFETY LIMITS

The objective of this task is to make a comprehensive assessment of the information on the effects of fuel element

failure, coolant boiling and operating safety limits needed to design LMFBR fuel assemblies.

A broad assessment of these information needs was completed during this quarter. Consideration was given to the influence of these considerations on core design, safety, and fuel cycle economics. A fault tree using a mean-annual cost technique was used to weigh the relative importance of suspected accident mechanisms and core design variables. Recommendations for analytical and experimental development were made based on this evaluation. A draft report has been completed and is undergoing review.

REFERENCES

1. Gee, W. R., "Conceptual Plant Design, System Descriptions, and Costs for a 1000 MWe Sodium Cooled Fast Reactor", GEAP-5678, December 1968.
2. D. J. Bender and P. M. Magee, "Turbulent Heat Transfer in a Rod Bundle with Liquid Metal Coolant", GEAP-10052, July 1969
3. Sutherland and Watwood, "Creep Analysis of Statically Indeterminate Beams", BNWL-1362, March 1970.
4. R. J. Roark, "Stresses and Deflections in Thin Shells and Curved Plates due to Concentrated and Variously Distributed Loading" NACA Technical Note No. 806, 1941.

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 the evaluation of the safety of fast reactors, and to provide guidance for effective planning and coordination of experimental programs. The results of these studies will guide further development of analytical methods and form a basis for establishing the 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

A version of the VENUS⁽¹⁾ coupled hydrodynamics and point kinetics calculation was obtained from HEDL/WADCO and made operational locally on a UNIVAC 1108. Conversion of the FREADM code to the UNIVAC 1108 was initiated to facilitate coupling with the VENUS code.

The FREADM-1⁽²⁾ documentation was completed and the code package prepared for release to the Argonne Code Center.

Modifications to the primary loop hydraulics calculation were made to permit calculation of either tank or loop type concepts. Checkout of the modifications is in progress. Specifications have also been written to permit a zero cover-gas volume above the intermediate heat exchangers.

A core model of a 1000 MWe LMFBR was developed to provide a basis for the analysis of primary pump coast-down accidents. The effects of burn-up and various assumed fuel failure and coolant boiling criteria will be investigated.

The general layout of the FREADM-2 code was specified. A generalized flow chart showing the linking of the principle modules is shown in Figure 9-1 with a summary of the function of each link.

9.2.2 Thermal-Hydraulics

The improved core heat transfer and voiding computational model has reached its operational state as an independent model and has been used for a limited sensitivity study. Based upon input from Task G-6.4, the influence of different parameters on the voiding dynamics in a coolant channel was examined.

The main parameters considered were:

- 1) The final amount and rate of onset of flow blockage.
- 2) The initial superheat.
- 3) The initial thickness of the liquid film retained on the fuel cladding.
- 4) The evaporation and condensation coefficients.

The model assumes a uniform temperature across the coolant channel, except in the voided region, where a temperature gradient exists between the liquid film on the wall and the vapor in the bubble. Thus, radial condensation effects are not adequately described. This will be the subject of a later study.

A typical result of a simulation run is given in Figure 9-2, as a function of time for the case of a flow reduction to 20% of nominal in 0.5 sec. The upper and lower lines in the voiding pattern represent the location of the top and bottom of the bubble respectively.

The initial superheat in this case was about 20°F, resulting in generation of the first bubble early in the coast-down. The rapid reduction of flow to a relatively low rate, creates a significant difference in clad temperatures, between the hot core region and the cold blanket and plenum regions. Thus, the void, which is carried downstream by the residual flow, condenses rapidly in the upper blanket.

The assumption of a higher initial superheat would have had the following effect. Depending on the amount of superheat, the first and perhaps the second bubble would not have been formed, and voiding would have started in the last core node at a higher temperature and pressure than in the case shown. However, the first two bubbles do not grossly affect the temperature distribution in the blanket, and it appears that a higher initial superheat (corresponding to boiling at the point of the third bubble initiation) would not significantly modify the voiding pattern beyond this time.

Also of great importance in calculating the bubble dynamics, are the assumptions about the evaporation (and condensation) coefficients (assumed 10,000 Btu/hr ft²°F), and the initial film thickness. The latter has been found in various sodium expulsion experiments, to lie between 0.05 and 0.7 mm. In this case, the initial value used was 10 mils (0.25 mm), which is considerably below the upper experimental value. Nevertheless, no dry-out was calculated. This is partially due to the particular type of flow blockage considered, since the residual flow drives the bubble out of the initially voided area, and wets the clad again after only a short exposure to evaporation.

For the same reason the last core-node is not cooled significantly and the code generates the subsequent bubbles at the same location. Flow reversal of the lower slug is found to be significant during the initial ejection process only at very high clad temperatures, as the channel inlet pressure is much higher than the outlet pressure, and the major flow restrictions are located at the inlet. Thus, the high initial vapor pressure is essentially relieved by the fast ejection of the upper slug.

9.2.3 Fuel Failure and Movement

Specifications are being written to permit inclusion of the transient overpower fuel failure model (TOFF) as a subroutine of the FREADM code. Results of a study using this code were reported in the 34th Quarterly Report, with corrected figures appearing in the 35th Quarterly Report.

9.2.4 Parametric Studies

Figure 9-1 in the 35th Quarterly Report illustrated results of a simple model used to calculate channel expulsion rates due to fission gas release. Comparison of this figure

- FREM — Controls problem initialization, wrapup and restart options.
 FREADM — Control link between coupled kinetics and heat transfer and primary loop hydraulics.
 FRDM — Controls coupled point reactor kinetics and multichannel heat transfer.
 INIT — Problem initialization from steady state conditions.
 HTRSFR — Multichannel heat transfer calculations.
 KNTCS — Point kinetics and reactivity feedback calculations.
 VOIDM — Primary loop hydraulics and channel voiding calculation.
 VDINT — Initialization for primary loop hydraulics and channel voiding.
 VEDIT — Edit control for primary loop hydraulics and channel voiding module.
 FDCALL — Transient hydraulics for primary loop and channel voiding calculations.
 HYDYN — Interface link between pre-melt and post-melt calculations.
 MARS — Modified Bethe-Tait analysis in R-Z geometry.
 VENUS — Coupled point reactor kinetics and hydrodynamics in R-Z geometry.

Figure 9-1. FREADM-2 Calculation Links

Figure 9-2. Void Pattern as a Function of Time

with results from the channel voiding model (section 9.2.2) revealed a 10X time scale error. Corrected curves are shown in Figure 9-3.

REFERENCES

1. W. T. Sha and T. H. Hughes, "VENUS—A Two-Dimensional Coupled Neutronics-Hydrodynamics Fast Reactor Power Excursion Computer Program", ANL-7701, July 1970.
2. D. D. Freeman, et. al., "The FREADM-1 Code, A Fast Reactor Excursion and Accident Dynamics Model", GEAP-13608, September, 1970.

Figure 9-3. 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 general objectives of the core materials development program are to contribute to the attainment of the desired levels of 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 LMFBR environment.
- Investigate the rate of alloy swelling and the environmental and metallurgical factors which influence and control it in LMFBR primary alloys and those which are potential candidates.
- Determine the minimum criteria for the performance of core materials which will permit attainment of overall LMFBR objectives. Determine the quality control techniques to measure performance against these criteria.
- 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 LMFBR applications.
- Interpret the in-depth behavior of core materials under simulated LMFBR conditions and recommend specific material applications and design limits for LMFBR use.

10.2 MATERIALS IRRADIATION PERFORMANCE

The objective of this subtask is to provide and follow capsules used as irradiation vehicles for irradiating clad 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.

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

The series L-6, L-15 and MT-4 materials capsules are at EBR-II in various stages of their scheduled irradiations. The three L-6 capsules (Mark-B7 type) were discharged from EBR-II following run 45 (mid-September) and will

be shipped to the VNC during the next quarter. These capsules have accumulated a total fluence* of 8.3×10^{22} n/cm² in S/A X018B.

The four L-15 capsules and the MT-4 capsule (Mark-A19 type) have been reconstituted into S/A X080 for further irradiation. This S/A was loaded into core position 6E5 of EBR-II for a scheduled irradiation of 2,900 MWd starting with run 46. As of mid October, the four L-15 capsules and the MT-4 capsule have accumulated 1×10^{22} and 7.0×10^{22} n/cm² total fluence respectively.

10.2.2 Series L-2 EBR-II Materials Capsules

The four L-2 series capsules irradiated in X050 to 9.05×10^{22} n/cm² total fluence are presently undergoing post irradiation examination. This examination includes gross gamma scanning, profilometry, visual examination, and sectioning of the capsule and specimen inserts to provide specimens for clad alloy swelling and mechanical property studies. The examination is nearing completion. All of the nondestructive testing is complete. The capsules have been sectioned and the specimen inserts removed from the capsule tubes. The tubular test sections were profiled, and then the tensile specimens were removed from the test sections. Most of the tubular and tensile specimens exhibited some irradiation induced clad swelling as was apparent from the observed diameter (Figure 10-1) and length increases in the specimens (Figure 10-2). The core tubular specimen in L2E (type 347) and L2G (type 304) also exhibited gross mechanical induced swelling (17 to 29 mils) (Figure 10-1) which probably occurred due to the pre-irradiation pressurization of these capsule inserts. The other two capsules, L2C (type 316L) and L2A (Incoloy-800), apparently leaked prior to or upon the initial reactor startup since they did not exhibit this gross mechanical induced swelling.

All of the L-2 specimens were stored for future work on other subtasks in Task L.

10.2.3 Series L-16 Materials Capsules

The new group of materials capsules designated L-16 series capsules is being designed to provide quantitative data on the swelling behavior of candidate clad alloys Type 304 and Type 316 as a function of temperature of irradiation, metallurgical condition and fluence.

10.2.3.1 L-16A Capsules

These capsules are designed to accommodate various temperature monitoring devices and flux wires to measure the temperature of irradiation and neutron flux in a typical

* Unless otherwise noted, all fluence reported are based on EBR-II physics predictions without any correction factors.

Figure 10-1. Diameter Increases in L-2 Tubular Specimens Due to Clad Alloy Swelling (L2A-3) and Pressurization (L2G-3)

Figure 10-2. Post Irradiation Appearance of Type 347 Tensile Specimens from L2E3

L-16 type Mark-A19 subassembly. Nineteen of these capsules are being fabricated. All materials are available and assembly of capsule inserts is scheduled for early November. The final encapsulation of these capsules will be done by ANL-Idaho at the EBR-II Fuel Evaluation Facility.

10.2.3.2 L-16B Capsules

The L-16B capsules were returned to BRDO-GE from EBR-II, Idaho for rework which was required in order to change the insulation gas in 15 of the 19 capsules from 100% argon to 80% argon and 20% helium. This change was necessary to accommodate additional heat transfer by conduction since recent bench tests — described below — have shown the heat transfer by radiation in the L-16 capsule design to be less than originally allowed for.

10.2.3.3 L-16 Bench Tests (Emissivity Tests)

The accuracy of design temperatures of the L-16 material capsules depends heavily on the accuracy of the thermal property values used in the calculation. Specifically, the emissivity of the stainless steel tubing carries a relatively large uncertainty margin. In capsules with argon as the thermal bond more than 1/3 of the total gamma heat absorbed is dissipated by way of thermal radiation. The resulting uncertainty in the emissivity of the tubing affects the overall result significantly. To narrow down the uncertainty in emissivity, a bench test was designed specifically to have the same material and configuration as used in the L-16 series of material capsule. The result obtained from this test would, therefore, be directly applicable to the design of all L-16 capsules with similar material and configuration.

The apparatus consisted of two concentric type 304 steel tubes. The inner tube was 1/4" O.D. and contained a calibrated heater which could heat this tube to temperatures in excess of 1300°F. The outer tube was 3/8" O.D. with a 0.020" wall. The temperature of this tube could be held constant at any desired temperature from ambient to 700°F by proper cooling or heating. Temperatures of the inner and outer tube were monitored by thermocouples fixed to both tubes. The annulus between the tubes could be controlled to contain a vacuum or any desired gas mixture.

The results of these tests are best shown by plotting the apparent emissivity of both tubes as a function of the temperature of the inner tube, Figure 10-3. The apparent emissivity is defined here as the value obtained by assuming equal emissivity values for the two surfaces although they are at different temperatures. The emissivity values shown in Figure 10-3 were calculated using a one dimensional heat transfer model for a vacuum annulus. The emissivity values obtained for argon runs were quite similar. The most important observation shown in Figure 10-3 is that the emissivity values range from 0.24 to 0.31 which is much less than the assumed 0.5 used in the original L-16 capsule designs.

In Figure 10-3, the apparent emissivity for a temperature of inner tube of 700°F is extrapolated to be 0.245. Remembering this figure is constructed from test runs in which the nominal temperature of the outer tube is 700°F, it is reasonable to assume that the asymptotic value of the *actual* emissivity is 0.245.

Using the value of 0.245 as the *actual* emissivity of surface at 700°F together with the apparent emissivity from Figure 10-3, the actual emissivity at temperatures were

Figure 10-3. Apparent Emissivity of SS Tubes Plotted as a Function of Temperature of Inner Tube

Table 10-1
CLAD SWELLING DATA FROM PINS F2H AND F2T

Specimen	Material	Specimen Location (in. above bottom of core)	Fluence* n/cm ² x 10 ⁻²² (E > 0.1 Mev)	Temperature*		$-\Delta\rho/\rho_0$ ** %
				°F	°C	
F2H 4	316	3.6-4.6	3.85	942	506	1.41
F2H-7B	316	6.6-7.8	3.93	982	528	1.47
F2H-7A	316	7.8-9.1	3.77	993	534	1.38
F2H-8C	316	10.9-11.9	3.10	1000	538	0.68
F2T-8	347	7.1-8.1	3.92	986	530	2.25
F2T-9	347	8.1-9.3	3.76	993	534	1.99
F2T-11	347	10.3-11.3	3.27	1000	538	1.35

*Mean values over length of specimen

**Based upon unirradiated values of

316 Tube No. F52, archive ballized once, density = 7.929 g/cm³

347 Tube No. F12, archive ballized once and swaged once, density = 7.901 g/cm³

determined. Figure 10-4 shows the *actual* emissivity plotted as a function of temperature. Also shown in the figure is the *apparent* emissivity (dashed line).

The temperature dependent emissivity shown in Figure 10-4 can be represented by the equation below:

$$\epsilon = 0.27078 - 1.503 \times 10^{-4}T + 1.626 \times 10^{-2}T^2$$

where T = surface temperature (°F),

Equation is applicable to the temperature range of

$$600^\circ\text{F} \leq T \leq 1400^\circ\text{F}$$

The overall uncertainty of the values of emissivity was estimated to be $\pm 15\%$. This estimate is based on inspection of plots similar to Figure 10-3 but including points calculated from using linear power values of $\pm 5\%$. It was observed that the *apparent* emissivity as represented by the curve of best fit together with a band of $\pm 15\%$ encompasses approximately 90% of all points.

10.3 CLAD ALLOY SWELLING

The objective of this program is to monitor the swelling of clad alloy materials and S/A hardware in available PA-10 irradiated specimens to determine the relationship between swelling and fluence, irradiation temperature, alloy microstructure, and stress history. The techniques used to measure clad swelling are liquid im-

mersion density, dimensional analysis, and transmission electron microscopy.

10.3.1 Immersion Density Measurements

Immersion density measurements of irradiation induced metal swelling have been completed on seven alpha cleaned cladding sections from pins F2H (316 clad) and F2T (347 clad) irradiated to peak burnups of 65,000 MWd/Te. The data are listed in Table 10-1.

The specimens were in the form of clad sections 1' 1 1/4" in length. The fuel was removed by drilling, reaming and subsequent electro polishing of the inside surface. These operations resulted in reduction of the wall thickness from a nominal value of 0.015 in. to 0.011 - 0.013 in. A minimum of six density measurements were made on each specimen using boiled distilled water with a few drops of wetting agent added. The unirradiated control data were obtained from archive specimens of the tubes used to fabricate F2H and F2T.

The data are presented in Figures 10-5 and 10-6 in which the measured density decreases have been converted to diameter increases assuming isotropic swelling. These increases in diameter are compared with the values measured on the pins by profilometry and also with the extent of swelling as predicted by empirical swelling correlations. The bands about the density data points represent the data limits and those about the profilometry measurements represent the maximum and minimum values over the specimen lengths. The fluence values were based on EBR-II

Figure 10-4. Actual and Apparent Emissivity at Different Surface Temperatures

Figure 10-5. Pin F2H 316 Clad

Figure 10-6. Pin F2T 347 Clad

physics calculation with 1/1.2 correction factor and the temperature values represent the mean clad wall temperatures over the specimen lengths for end-of-life conditions ($Q = 15.16$ Kw/ft peak for F2H and 15.25 Kw/ft peak for F2T). The fluence and temperature values are listed in Table 10-1.

It can be seen from Figures 10-5 and 10-6 that the clad swelling measured by immersion density accounts for less than half of the measured total pin diameter changes. The remaining ΔD appears to be too large to be accounted for solely by clad strain caused by differential thermal expansion of fuel and clad on startup, so that it appears that significant thermal creep of the cladding caused by fuel swelling has occurred. Whether evidence of thermal creep can be detected under transmission electron microscopy of the cladding remains to be seen. It was found that the 347 clad had swollen more than the 316 clad. Indeed swelling of the 316 clad is appreciably below that predicted by available empirical swelling correlations.^(1,2) When the F2H 316 data are compared with UK immersion density from solution treated and cold-worked 316 pin claddings, it is found that they fall closer to the UK cold-worked data. This observation would appear to support the postulate that the sizing operations carried out on the tubing (ballizing and swaging) have resulted in some degree of cold work which has subsequently affected the degree of swelling. However, optical microscopy of archive material* reveals little or no effect of the sizing operation. It is appreciated that swelling will be influenced more by sub-microscopic defects than by anything visible by optical microscopy. Hence, transmission electron microscopy of unirradiated archive material as well as the irradiated specimens is currently underway and results of this work will be reported as they become available.

10.4 CLAD ALLOY MECHANICAL PROPERTIES

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

10.4.1 Biaxial Creep Program

The biaxial creep facility described earlier⁽³⁾ 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.

Post irradiation creep rupture testing of 304 stainless steel is continuing. All specimens presently being tested are three inch long tubular sections of 3/8" diameter fuel pin capsule tubes and materials capsule tubes. The specimens are pressurized and inserted in a furnace at a controlled temperature. Strain is recorded by removing the capsules from the furnace and after allowing the specimens to cool, scanning the diameter along the complete length with an optical scanning unit. Originally a mechanical profilometer was used for this purpose, however, considerable difficulty was encountered in specimen alignment and the following system was evaluated.

Measurement System

An optical profilometer system which comprises an optical head, amplifiers, and an X-Y recorder is presently in use.

The optical head is a high linearity PhysiTech photo multiplier tube which is positioned so that the field of view contains both edges of the tubular specimen. The edges are shown in profile by suitable positioning of the lighting so that the edges represent optical discontinuities. The head rapidly scans the specimen (sample period approximately 20 microseconds) and the continuous pulses at the optical discontinuities are monitored by a PhysiTech Model 441 Gaging Control Unit. The displacement of each optical discontinuity (left and right edges of specimen) from a reference position are indicated independently on the Gaging Unit. To determine the diameter, the Gaging Unit signals representing the displacements of both edges are transmitted to a PhysiTech Model 41B Differential Amplifier, which adds the signals algebraically and transmits the result to the X-Y recorder as a single diameter signal.

During the continuous diametral scanning, the specimen is moved along its longitudinal axis within the field of view of the optical head. The speed of movement is synchronized with the recorder sweep to provide a continuous recording of the specimen diameter as a function of specimen length. The reference dimensions for the specimens are provided by two precision cylindrical mandrels, one mandrel of greater diameter than the specimen, the other mandrel of lesser diameter.

The warranted absolute accuracy of the system is 0.1% of the full optical range. The field of view used is 0.4 inches, and so the warranted accuracy is ± 0.0004

*It should be noted that the available archive material is not wholly typical of the irradiated cladding. The archive samples were taken from the same tubes used to fabricate the pins after intermediate sizing operations but not after the final ballizing and/or swaging pass. The F2H archive was ballized once, whereas the actual cladding was ballized once and swaged once. The F2T archive was ballized once and swaged once, while the final tube had received two each ballizing and swaging operations. Post irradiation optical metallography⁽³⁾ also showed little effect of the preirradiation sizing operations on the as-irradiated microstructure.

Table 10-2
CREEP TEST CONDITIONS – BATCH NO. 2

Test Temp. °F	Hoop Stress Ksi	Irradiation Temp. °F	Total Fluence n/cm²
1300	12	795-840	3.6-3.9 x 10 ²²
1300	12	710	2.2-2.8 x 10 ²²
1300	10	875-840	3.5-3.9 x 10 ²²
1300	10	710	2.8-2.9 x 10 ²²
1300	8	875-900	2.9-3.6 x 10 ²²
1300	8	710	2.7-2.9 x 10 ²²
1300	6	745-800	3.0-3.7 x 10 ²²
1300	6	710	2.2-2.7 x 10 ²²
1300	4	800-850	3.7-3.9 x 10 ²²
1300	4	710	2.7-2.9 x 10 ²²
1100	30	795-846	3.8-4.0 x 10 ²²
1100	25	710	2.2-2.7 x 10 ²²
1100	25	845-885	3.6-3.9 x 10 ²²
1100	20	710	2.7-2.9 x 10 ²²
1100	20	840-875	3.8-4.0 x 10 ²²
1100	15	710	2.7-2.9 x 10 ²²
1100	15	800-850	3.6-3.9 x 10 ²²
1100	10	710	2.3-2.7 x 10 ²²
1100	10	850-885	3.6-3.9 x 10 ²²
900	35	745-795	2.9-3.6 x 10 ²²
900	35	885-905	2.9-3.6 x 10 ²²
900	35	710	2.2-2.8 x 10 ²²
900	30	710	2.7-2.9 x 10 ²²
900	20	710	2.7-2.9 x 10 ²²

Table 10-3
INTERIM CREEP TEST RESULTS – BATCH NO. 2

Temp. (° F)	Stress (Ksi)	Diametral Strain (Percent)				Average % Reduction in Strain Rate, Attributable to Radiation Effects
		at 211 hours		at 330 hours		
		Irrad.	Unirrad.	Irrad.	Unirrad.	
1300	12	0.21	3.24	0.45	4.39	~90
1300	12	0.31	3.24	0.58	4.39	
1300	10	0.34	1.3	0.53	1.76	~60
1300	10	0.34	1.3	0.63	1.76	
1300	8	0.26	0.42	0.39	0.57	~40
1300	8	0.32	0.42	0.34	0.57	
1300	6	0.05	0.10	0.10	0.137	~7

inches. The major error contributing to the accuracy limitation is the stability of the Gaging Unit.

However, typically, the signal disturbance due to minor stability shifts average ± 0.0001 inches, and rarely reach ± 0.0002 inches. The larger shifts of ± 0.0004 inches occur infrequently, and are identifiable by their shape. If the latter occurs during a run, the run is repeated. In addition the reference dimensions are checked before and after a run.

Creep Testing

The biaxial creep tests on a comparatively low fluence ($\sim 1 \times 10^{22}$ n/cm² total*) group of specimens have been interrupted at 3000 hours and tests on higher fluence ($\sim 3 \times 10^{22}$ n/cm² total*) specimens have now been initiated.

The low fluence group of specimens had shown a marked reduction in strain rate by comparison with unirradiated specimens. The amount of reduction appears to be stress dependent, the differences between the creep rates of irradiated and unirradiated material is more pronounced at the higher stress levels. At diminishing levels of stress, strain rates for irradiated and unirradiated specimens begin to converge. It is intended that the second batch of specimens will better define this trend and establish the lowest level of stress at which strain rate divergence is present.

The second batch of specimens has now been on test over 300 hours. Test conditions are shown in Table 10-2. Data taken up to 330 hours on the higher temperature specimens are consistent with the observations from the first test series. The irradiated specimens are demonstrating

a reduced creep rate, and once again the reduction in creep rate is more pronounced at the higher stress levels (see Table 10-3). Control data for these tests were obtained using data from tests at 1300°F and stress levels of 7,100, 9,200 and 12,300 psi to derive the equation

$$e = A \sigma^m t^n$$

where e = Diametral strain in/in
 σ = Hoop stress ksi
 t = Test time hours

$A, m \& n$ = Material constants (2.45×10^{-9} , 5.0, 0.75 respectively, applicable at 1300°F only)

A third group of twenty eight specimens have now been pressurized. Testing of this group is expected to start early in November. The fluence for this group is the highest for any material tested in creep at this laboratory 7×10^{22} n/cm² total. The test conditions will correspond as closely as possible to the conditions for the present batch of specimens, (Table 10-2). In this way the effect of an increase in fluence can be more readily assessed.

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

* Fluence values were based on EBR-II physics calculations with 1/1.2 correction factor

under purchase order 5G430G is continuing. The material consists of Type 304H and Type 316H stainless steel with 0, 10, 20, 30 and 50 percent cold work, the figure representing the area reduction impressed during the final cold draw. Test parameters include a series of hoop stresses, selected to produce failure in 400, 1000 and 4000 hours at a test temperature of 1300°F. All material is seamless, 0.250-inch in diameter with a 15-mil wall. All tubular test sections contain welded closures; the tubes are internally pressurized with argon, producing an essentially constant stress on the tube wall. Each test section is encased in a monitored low pressure tube; failure of the section is inferred by an alarm signalling a pressure rise of 20 psi in the outer tube. Examination of each failed tube included (1) establishment of point of failure (in some cases, this required pressurizing after failure to as much as 1000 psi before a leak was detectable), (2) diametral measurements for the assessment of failure strain and (3) metallographic sectioning (in selected cases) to evaluate the mode of failure. Results are presented and discussed below.

The relationship of initial hoop stress to stress-rupture life is illustrated in Figure 10-7 for the four cold worked lots and one annealed lot of Type 316H. A similar data display on the Type 304H material was presented in an earlier document⁽⁴⁾. As the earlier results on the Type 304H cold worked materials indicated, these results confirm that moderate cold working imparts additional stress-rupture strength to Type 316H at 1300°F, only in short-term applications. Moreover, heavily cold worked materials (in this instance, levels of cold working greater than 30 percent) are at all times inferior to comparable annealed material.

Variations in maximum diametral ductility shown by the several cold worked grades tested continue to follow the trends established by the earlier data. These trends indicated a reduction in ductility with light cold working and a ductility level comparable to annealed material for the heavily cold worked materials. The latter behavior is thought to be evidence of the onset of recovery and grain refinement mechanisms in a structure that is highly unstable under the conditions of this test.

Stress Rupture Properties of Austenitic Stainless Steel: the effect of surface defects on cold worked tubing

An experimental program was initiated during the current period to evaluate the effect of surface defects on the long-term integrity of cold worked tubing of austenitic stainless steel tested in stress-rupture. This program is similar in scope to that performed on annealed material, described in an earlier document.⁽⁵⁾ Although only four failures have occurred to date, the results are significant. Of the four monitored failures, three occurred in tubes inserted as controls for simultaneously-tested samples containing longitudinal defects as large as 10 percent of the wall in depth. The one defected sample that failed (20 percent cold worked) outlasted its control by two days; the location of rupture was remote from the defect. These tests are being performed on 10 and 20 percent cold worked tubing of Type 316H stainless steel tested earlier. This suggests that surface defects of moderate depth are not likely to reduce the stress-rupture strength of annealed or cold worked tubing.

REFERENCES

1. S. Oldberg, D. Sandusky, P. E. Bohaboy and F. A. Comprelli, "Analysis of Swelling of Austenitic Stainless Steels in Fast Reactors", Trans. ANS 12, No 2, 588 (1969).
2. WARD - 4135 - 1, Page 6, Oxide Fuel Element Development Quarterly Progress Report for Period Ending September 30, 1969, Westinghouse Electric Corp. (November, 1969).
3. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, 30th Quarterly Report, Feb.-April, 1969", May 1969, (GEAP-10028).
4. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, 34th Quarterly Report, Feb.-April, 1970", June, 1970, (GEAP-10028-34).
5. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, 33rd Quarterly Report, Nov., 1969-Jan., 1970", March, 1970, (GEAP-10028-33).

Figure 10-7. Stress-Rupture Properties of 5G430G Tubing: Type 316H, Heat No. M2783, Test Temp.: 1300°F

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 COMPILATION AND DISTRIBUTION OF EVALUATED DATA

ENDF/B Tape Numbers 202, 203 and 204 have been received from BNL and have been processed for use on the GE generalized files. The three tapes, along with Tape Number 201 which was received in May, 1970, constitute Version II of the ENDF/B library and include changes in data and data formats as recommended by the Cross Section Evaluation Working Group. Recently revised data for Fe, Ni, Cr and C have been used in the benchmark critical evaluations which are being carried out as part of the activities of the Data Testing Subcommittee of CSEWG.

Ten benchmark problems were prepared for the CSEWG Phase II data testing of ENDF/B. The results of the benchmark evaluations will be used to check out multi-group preparation procedures and to determine the applicability of the ENDF/B, Version II data to the present LMFBR and demonstration plant design effort. Several different national and private laboratories were assigned five or more of the critical assemblies for standardized analyses.

General Electric has accepted the responsibility for evaluating three plutonium criticals — VERA 11A, ZPR 3 Assembly 48 and ZEBRA 3- and two uranium criticals — ZPR 3 Assembly 11 and ZPR 3 Assembly 12. A simplified computational model was used for each of the assemblies based upon the prescription of Davey and Hess⁽¹⁾. Adjustments for differences between the actual experimental reactor systems and the simplified models have been included (i.e., corrections for heterogeneity, worths of interface gaps, edge irregularities and the cylinder to sphere volume shape factors).

Preliminary results of the G E benchmark evaluations are given in Table 11-1. Some identifying characteristics of the critical models are also included in Table 11-1 to indicate the basic differences between the assemblies.

The ENDF/B, Version II data tends to give slightly lower eigenvalues than the Version I data (e.g., about 0.005-0.01 lower for ZPR-3 Assembly 48). The principal differences between the Version I and Version II data which could lead to the lower eigenvalues are higher alpha values for Pu-239 and higher capture cross sections for Fe, Ni and Cr. Changes in the Version I data which would tend to increase the eigenvalue include slightly higher values of ν for Pu-239 and, in the case of ZPR 3 Assembly 48, lower scattering cross sections for carbon.

The eigenvalues which were computed for the plutonium fueled assemblies are about 3% lower than the measured values and this tends to place a low confidence on the use of ENDF/B, Version II data for present LMFBR design efforts. An examination of the fission ratios in Table 11-1 does indicate that both the U 238 and Pu-239 fission cross sections may be low and increasing these parameters would be in the direction of improving the agreement between calculated and experimental eigenvalues.

The preliminary results of the benchmark evaluations were the principal topic of the CSEWG meeting at BNL on October 7-8. Other reported results were minimal but tended to be in agreement with the results given in Table 11-1. The low eigenvalues which were reported for the plutonium fueled assemblies prompted an increased effort to complete and expand the testing and evaluation of the ENDF/B, Version II data. Specific commitments by G E, in addition to reporting the results of critical calculations, are to review and evaluate all new experimental measurements for Pu 239.

High resolution total cross sections for U 238 between 500 and 700 keV have been reported by Kopsch, et al.⁽²⁾ These data indicate significant fine structure in this energy range which had not been detected previously. The possibility that this fine structure introduces significant self shielding to the capture cross section and could account for some differences between integral and differential measurements (such as low K_{eff}) was investigated by preparing the data in the ENDF/B format and computing self shielding factors with the cross section processing code ENDRUN. The capture cross section was assumed to vary by the same proportion as the total cross section and it was found that for an effective potential scattering of 10 barns (more self shielded than in most fast critical assemblies) the self shielding effect was only 1 percent. Thus, it is concluded that the fine structure in the U 238 cross sections from 500 to 700 keV have only a negligible self shielding effect. However, self shielding between 50 keV, the upper energy range of the unresolved resonances in ENDF/B, and 500 keV could have a significant effect if the fine structure becomes more pronounced in this range.

Table 11-1
PRELIMINARY RESULTS OF BENCHMARK CRITICAL EVALUATIONS

	ASSEMBLY				
	VERA-11A	ZPR-3 Assembly 48	ZEBRA-3	ZPR-3 Assembly 11	ZPR-3 Assembly 12
ASSEMBLY CHARACTERISTICS					
Core Radius (cm) ⁽¹⁾	13.985	46.46	23.67	31.61	28.76
Enrichment	100% Pu	~18% Pu	~10% Pu	~12% U ²³⁵	~21% U ²³⁵
Moderator	Graphite	Graphite-sodium	-	-	Graphite
Mean Fission energy (keV)	~570	~240	~610	~370	~270
COMPUTATIONAL MODE					
	S8 Transport	S4 Transport ⁽²⁾	S8 Transport	Diffusion	S4 Transport
RESULTS					
k_{eff}	0.977	0.970 ⁽²⁾	0.963	0.959	0.996
CALCULATED/MEASURED⁽³⁾					
$\sigma_f^{238}/\sigma_f^{235}$	1.062	0.954	0.925	0.880	0.909
$\sigma_f^{239}/\sigma_f^{235}$	1.035	0.926	0.960	0.943	0.951
$\sigma_f^{240}/\sigma_f^{235}$	0.993	0.918	0.999	0.945	-
$\sigma_c^{238}/\sigma_f^{235}$	-	1.007	-	1.067	-
Pu ²³⁹ Material Worth	-	1.28	-	1.07	-
U ²³⁵ Material Worth	-	1.35	-	1.17	-
U ²³⁸ Material Worth	-	1.29	-	1.20	-
Fe Material Worth	-	1.41	-	1.00	-
B-10 Material Worth	-	1.14	-	1.03	-

1. Simplified sphere — In establishing the benchmark prescriptions, all adjustments for differences between the actual experimental reactor system and the simplified models have been included. Corrections for heterogeneity effects, worths of interface gaps, edge irregularities, and the cylinder-to-sphere volume shape factors were adopted on the basis of experiments or semi-empirical studies.⁽¹⁾
2. The transport mode was used only to calculate the eigenvalue of this assembly. All other results are based upon diffusion theory calculations.
3. Given as the ratio of the calculated value to the measured value as reported by Davey and Hess (Reference 1).

11.3 SUPPLEMENTARY ANALYTICAL INTERPRETATION OF INTEGRAL DATA

In order to evaluate the validity of design methods using experimental results from ZPPR, it will be necessary to perform the analyses with a degree of accuracy equivalent to that currently used for reactor design. In some cases, acquiring this degree of accuracy will be time consuming and expensive. Maximum use will be made of analyses performed at Argonne National Laboratory and elsewhere.

Calculations of ZPPR, Assembly 2, are being performed according to the following procedure:

1. Generation of effective cross sections in 60-groups for each core region and blanket, based on ENDF/B-II.
2. Correct the effective cross sections for spatial heterogeneities.
3. Condense to 16 groups using space-dependent spectra.
4. Perform two-dimensional calculations of the basic ZPPR-2 measurements using both direct and synthesis solutions.

Following this analysis, consideration will be given to uncertainties in design quantities which are not measured in ZPPR-2, by using the differences obtained between measured and calculated quantities in ZPPR-2 along with design computer codes, such as the fuel management code, FUMBLE

Cell heterogeneity calculations have been run for the inner and outer core regions of ZPPR-2 using S_n transport theory. The degree of quadrature approximation required for this calculation is S_{16} and the results indicate about a +0.013 high-energy heterogeneity effect for K_{eff} in the inner core cell and a +0.008 effect in the combined outer two cells.

Since heterogeneity effects of resonance absorption are to be considered in this analysis, particularly for rodged fuel, calculations of the radial flux and absorption distribution in a 1/4 inch O.D., U-238 oxide pin were performed with both the one-dimensional discrete ordinates code SN1D and the more efficient resonance code REAX,⁽³⁾ which uses approximate integral transport theory on a hyperfine group structure. The results over the energy range from 12 to 56 eV, given in Table 11.2, indicate that REAX provides acceptable accuracy for absorption and power distributions across rods which are surrounded by essentially homogeneous material.

11.4 IMPROVEMENTS OF EXISTING METHODS FOR GENERATING GROUP CONSTANTS

The ENDRUN (Version 2) code has been modified to handle the latest ENDF/B, Version II, materials. Changes were made in the resolved resonance routine for including the higher ℓ states.

Modifications are being made to the elastic transfer routine to calculate the matrix by using collision

probabilities instead of the present equal probability into equal energy widths. The Selengut-Goertzel method of elastic moderation in hydrogen is also to be included.

Testing of the TDOWN-2 code is presently about 80% completed. This testing is being performed in conjunction with the ENDF/B-II data testing activities described in 11.2 above.

11.5 IMPROVEMENT OF FUEL CYCLE AND MANAGEMENT SCHEMES

The fuel cycle code FUMBLE has been modified to include several options which are required for PDP fuel management studies. The principal change has been to increase the flexibility of the fuel reload option by making it possible to load fuel of the same fissile enrichment in core zones with different fuel volumes and/or different batch loadings. This option allows the users to analyze systems which are an accurate representation of the physical model. The option will have particular application in the analysis of small core regions which are being analyzed to determine the local shift in power density or magnitude of fuel exposure.

An extrapolation technique is used in FUMBLE to determine the end-of cycle reactivity based upon the operating interval and the reactivity associated with the reference input data. This end-of cycle reactivity was checked by running a BISYN problem using the end-of-cycle atom densities computed by FUMBLE. The BISYN and FUMBLE reactivity agreed to within 0.002 and confirmed the applicability of the FUMBLE approximation technique.

The FUMBLE report is presently being reviewed for final distribution.

11.6 COMPUTERS, COMPUTER LANGUAGE AND CODE CENTER

Two meetings of the *de facto* Committee on Computer Code Coordination, one on September 9-10 and the other on October 14-15, were attended by N. Barth, B. A. Hutchins and M. D. Kelley. In conjunction with the committee activities, reports for the Standard Data File and Programming Practices Subcommittees were prepared and distributed to Committee members. In accordance with the schedules set up by the Committee for arriving at recommended code development standards, a suggested standard card input format was also distributed.

The *de facto* Committee on Computer Code Coordination requested that each attending installation convert the 1DX, ETOX, ANISN, DAC, DOT2DB and MC² codes to its own computer. A report on the results of our conversion process was given at the committee meetings in Los Alamos and Chicago.

Table 11-2
COMPARISON OF RADIAL DISTRIBUTIONS OF CROSS SECTIONS AND FLUX

$(r/R)^2$	σ_a REAX	σ_a SNID	σ_s REAX	σ_s SNID	Flux REAX	Flux SNID	$\sigma_a \times \text{Flux}$ REAX	$\sigma_a \times \text{Flux}$ SNID
.05	4.512	4.227	14.838	13.579	95.447	95.651	430.7	404.3
.2	4.993	4.650	15.263	13.912	95.646	95.781	477.6	445.4
.4	5.596	5.555	15.710	14.293	95.869	96.018	536.5	533.4
.6	8.229	7.555	17.500	15.901	96.412	96.327	793.4	727.8
.8	13.074	14.771	21.352	19.134	97.342	96.906	1272.6	1431.4
.95	32.497	27.923	37.970	32.680	100.367	97.676	3261.6	2727.4
∞	61.592	-	50.991	-	100.000	100.000		
Average	10.195	9.376	19.341	16.972				

Problems encountered in the conversions include:

A. Non-Standard FORTRAN

1. "Include ABC" statement.
2. Non-standard FORTRAN subscripting.
3. Mixed mode arithmetic.
4. Improper use of data statements.
5. Improper use of argument lists (variables in the list were also in common).
6. Dimension statements encountered in the middle of a program.
7. Use of literals in format statements.
8. Variable names greater than six characters.
9. "DO" index as a nested "DO" index.
10. Illegal character in a format statement.
11. "IMPLICIT" statement.
12. Subscripted variables used as a subscript.
13. Several calls made to system routines.

All programs have been altered as necessary and compiled.

B. Comments on Execution

1DX, DAC, DOT2DB and ANISN have been successfully executed. ANISN caused the most problems because of the use of the difference between negative and positive zero as implemented on the CDC 6600.

ETOX and MC² have not been executed because required data tapes were not supplied with the codes.

Recommendations to the Committee have been made in the areas of:

1. Using USASI FORTRAN as a standard.
2. Adapting ANS 10.2 with minor revisions.
3. Adapting a standard call sequence to interface between the program and system dependent block data transfer routines. System dependent block data transfer routines provide the advantages of greater computer efficiency and more flexibility in data manipulation.
4. Adapting a standard card input format.
5. Adapting a standard interface between certain system routines and the program (such as accessing the computer clock).
6. Adapting a standard routine name which contains information as to the peripheral file structure.

11.7 SENSITIVITY STUDIES

The paper titled "Implications of Nuclear Data Uncertainties to Reactor Design",⁽⁴⁾ which was presented at the Helsinki meeting on Nuclear Data for Reactors in June, 1970, was revised to include recent nuclear data measurements and drafted for issuance as a topical report. The topical report is titled "Implications of Nuclear Data Uncertainties to LMFBR Design", and is currently in internal review.

REFERENCES

1. Memorandum from W. G. Davey and A. L. Hess to the Members of the Cross Section Evaluation Working Group, Additional Fast Reactor Benchmarks for Phase II Data Testing of ENDF/B, February 21, 1969.
2. Kopsch, D., Cierjacks, S., and Kirouac, E. J., "New Total Neutron Cross Section Measures of Uranium Between 0.5-4.35 MeV", KFK-1199 (May 1970).
3. Protsik, R., "Epithermal Capture and Fission Distribution in a Fuel Rod - The Computer Code REAX", GEAP-4973 (April 1966).
4. Greebler, P., Hutchins, B. A., Cowan, C. L. "Implications of Nuclear Data Uncertainties to Reactor Design", CN-26/102, 2nd International Conference on Nuclear Data for Reactors, Helsinki, Finland, June 15-19 (1970).

12. TASK P—FUEL ELEMENT DESIGN ANALYSIS

12.1 GENERAL

The objective of Task P is the development of a model which will predict fuel performance and its life-time. 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 PERFORMANCE AND FAILURE MODE STUDIES

12.2.1 Advanced Model of Fuel-Cladding Mechanical Interaction at Steady-State

The computer code "SS2" for the analysis of the fuel-clad system under stresses generated by steady-state swelling of the fuel and cladding has been debugged and is ready for use. The major features of this program were described in the 35th Quarterly Report. This program will be incorporated into the BEHAVE model of fuel performance.

12.2.2 Model of Fuel-Cladding Mechanical Interaction During Power Increases

A number of methods of modeling fuel-cladding mechanical interaction during power increases were studied to determine their feasibility.

Stress analysis of interaction effects during relatively rapid power increases differs fundamentally from stress analysis of a steady state condition in that, in the former case, the fuel deformation is dominated by thermal creep effects while in the latter case, fuel deformation is controlled by fission creep. This occurs because fission induced creep is too slow a process to produce much plastic strain of the fuel in the short time period of a rapid power increase.

Thermal stresses occurring during power increases tend to open up radial cracks in the outer region of fuel. Tensile stresses in the hoop direction cannot occur in the radially cracked region and thus cladding loads are carried by the hot, thermal creep dominated, fuel at the base of the crack. On the other hand, during slow power increases or at steady state the strain rate due to fission creep is high relative to all other strain rates in the system and stresses can be relieved by creep of the cold, fission creep dominated, outer region of the fuel. Stress analysis, in the two cases, requires somewhat different techniques due to the high radial gradient in stress in the thermal creep

dominated process compared to the low radial gradient of stress in the fission creep dominated case. In summary, the major difference between steady state and non-steady state processes is that cladding deformation is by creep in the steady state case and by yielding in the non-steady state case.

A necessary input to the power increase code is an "equation of state" for cladding under relatively high strain rates relating strain rate to fluence, temperature, stress, strain and the time history of these parameters. Such an equation is not available in the literature. However, development of such an equation is feasible based upon Task L tensile test data.

12.3 INTEGRAL PERFORMANCE AND FAILURE MODELING

12.3.1 BEHAVE: Fuel Element Performance Code

Construction of the BEHAVE code to simulate the performance of a fuel element throughout its operating history, including power changes and transients, is continuing. Subroutines have been added to complete the steady state model, as shown in Figure 12-1. The code in its present form computes the integrated effect of startup with fresh fuel, initial fuel restructuring, and the steady phenomena of clad swelling, loss of effective clad thickness, clad creep under the gas pressure load, and clad strain due to fuel swelling. It remains to extend the models to include the effects of rapid power changes.

Because the understanding of clad swelling is limited at this time, options are being developed to predict growth by any of the several empirical correlations (GE, WADCO, etc.), or by one of the semi-empirical models (Harkness and Li, Bullough, etc.). At present, only the empirical correlations are operational.

The effective thickness of the clad is defined as the amount of load bearing metal available to resist strain. Metal loss from the outside surface by corrosion in the flowing sodium is estimated with the aid of the GE-correlation developed and reported under AEC-PA-15 program. Reduction of strength of the clad due to reaction between the clad and fuel and fission products at the inside surface is accounted for by a temperature dependent reaction allowance, assumed uniform over the inside clad surface. The loss of strength due to leaching of clad alloy constituents into the sodium is being studied but has not yet been included.

The gas pressure inside the pin is taken to be the sum of (1) pressurization during fabrication, (2) release of gas adsorbed on the fuel, (3) release and vaporization of volatile fission products, and (4) release of gaseous fission products. All the fission gas generated is assumed to be

● COMPLETED EARLIER ■ NEW OR IMPROVED THIS QUARTER

Figure 12.1 Status of BEHAVE

released. The model of gas release as a function of temperature and burnup will be added when it is ready.

Predicting the clad creep to be expected in-pile is at a stage similar to that of clad swelling, and the multiple-option approach is also applied here. Several combinations of correlations for thermal creep, flux induced creep, and non-conservative-volume creep are available.

The modeling of the closure of the voids in the fuel (initial density less than 100% TD) and the subsequent clad strain due to further fuel swelling was described in the 35th Quarterly Report (GEAP 10028 35). At present, the simplified version, "SS1", is in operation in BEHAVE.

The design studies of fuel element performance in the demonstration plant, in progress under AEC-PDP program, require estimates of the effect on the mechanical interaction of fuel and cladding of the initial fuel density. The BEHAVE model in its present form (i.e. without power change effects) was used, under funding by the AEC PDP program, to scope the threshold of interference in two areas: differential thermal expansion between the fuel and clad during the first startup of fresh fuel, and clad strain due to fuel swelling of pins at several locations in the core.

Construction of BEHAVE will continue through the next quarter. The regimes of power change during operation will be added as they are developed. Work on the fuel modeling round robin problems will begin as the first version of the code nears completion.

12.3.2 STATUS: Local Deformation and Lifetime Code

The purpose of STATUS, the code for local deformation and lifetime, is to calculate local clad strain distributions and predict residual clad life as a function of operating history. The method of calculating the strain distributions was reported in the 35th Quarterly Report for the loading mechanisms of differential thermal and swelling expansion. During the current quarter, a method of dividing the total strain into components according to the cause of the strain was completed, and failure criteria based on a maximum allowable diameter and on a cumulative strain-to-failure were encoded. Figure 12-2 provides a review of the effects considered within STATUS and their degree of implementation. The details concerning loading mechanisms listed and the cumulative strain-to-failure criterion are available in subsection 8.3.4.2 of the 30th Quarterly Report. Pertinent material properties, how used and location of use within STATUS are shown in Figure 12-3.

Taking as input the average values of temperature and strain from BEHAVE at each axial node, superimposing the thermal hydraulic perturbation on temperature, it is possible to calculate the differential clad expansion strains. Such calculations at any point in time can be called "state"

calculations. The difference in strain between two "states" (different times, temperatures, etc.) is an accurate measure of the strain increment that would occur under actual conditions. However, the portion of this strain increment that occurs as short term strain (referred to as plastic), long term strain (creep), and elastic strains cannot be separated without additional information. The individual magnitudes of these components of strain is an important factor in the accumulated strain-to-failure criterion since the strain to failure is a function of the strain rate.

The method presently used for separating the strain increment into components versus a function of strain rate can be described with the aid of hypothetical conditions presented in Figure 12-4. The plot of strain versus time shows an existing condition at time t_1 . The state calculation is shown along with the level of its plastic, elastic and creep components. The increment of strain between state points at times t_1 and t_2 is shown as $\Delta\epsilon_2$. To determine where this increment will apply (elastic or plastic), the elastic strain level is needed at time t_2 . This is evaluated assuming a relaxation in which the elastic stress related to the elastic strain is assumed constant throughout the period and irradiation and thermal creep strain is determined. This approximation is shown as a linear increasing creep strain and a decreasing elastic strain level*. The level of elastic strain at time t_2 is used as a base on which to now add the state increment of strain $\Delta\epsilon_2$. Figure 12-4 indicates that the new level of elastic strain is less than the yield strain shown and therefore no plastic strain increment occurs. If the new level of elastic strain had exceeded the yield value as shown at time t_3 , in accordance with the assumed elastic perfectly plastic behavior, the increment exceeding the yield value would be accumulated as plastic strain as indicated. This process proceeds in succession from one set of data at one time to the next, accumulating strain increments in the two categories (creep and short term plastic flow) until the life criterion indicates no remaining life. At present the model considers only step changes in state points. Future work will include methods of incorporating steady clad strain rates (caused by fuel expansion or gas pressure).

The life limiting criterion can be reviewed with the aid of Figure 12-5. The cumulative strain to failure criterion takes the analytic form

$$\text{Damage Index} = \sum_i \left(\frac{\epsilon_i}{\epsilon_t} \right)$$

where ϵ_i = strain under condition i
 ϵ_t = total strain to failure under condition i

*This is a first approximation to the exponential creep strain relationship known to exist.

Figure 12.2 Status of STATUS—Local Deformation and Lifetime Analysis Code

PROPERTY	MODEL DEPENDENCE						CORRELATION USED				USING SUBROUTINE						
	TEMPERATURE	FLUENCE	FLUX	STRAIN	STRAIN RATE	STRESS	TIME	NONE	CONSTANT	LINEAR	STEPPED	CURVE FIT	MAIN	TEMPO	THERM	CSWELL	DAMCUM
YOUNG MODULUS	○	○							✓				✓				
POISSON'S RATIO	○								✓						✓		
YIELD STRESS	○	○							✓				✓				
YIELD STRAIN	○	○							✓				✓				
STRAIN HARDENING	○				○			✓									
RUPTURE STRENGTH	○	○					○	✓	✓								
RECOVERY	○	○						✓									
THERMAL COEFFICIENT OF EXPANSION	○								✓						✓		
THERMAL CREEP	●					●	●					✓	✓				
IRRADIATION CREEP	●		●				●					✓	✓				
CLAD SWELLING	●	●					○					✓				✓	
LOCAL STRAIN AT FAILURE	○	○			○				✓								✓
TOTAL ELONGATION	○	○			○			✓									

○ KNOWN OR SUSPECTED DEPENDENCE

● DEPENDENCE HAS BEEN INCORPORATED

Figure 12-3. Clad Material Properties Data Input to STATUS

Figure 12-4. STATUS Methods of Separating Components of Strain

The straight failure line connecting the normalized failure points in the creep and plastic modes represents the fundamental concept of equating damage index and life used. Paths A and B illustrate alternate histories leading to the same failure strain point while path C indicates termination at another condition.

STATUS is currently operating on dummy input data during the checkout phase. Future work will include account of steady strain rates (gas pressure, etc.), coupling to BEHAVE output, and study of other loading mechanisms.

Figure 12-5. Strain Damage Accumulation Diagram

APPENDIX A PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Number 1-107, from July 1959 through September 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-4480
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.

GEAP-5677

Twenty-Seventh Quarterly Report
May—July 1968.

GEAP-5700

Twenty-Eighth Quarterly Report
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

GEAP-10028-35

Thirty-Fifth Quarterly
May 1970—July 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:

GEAP-3287

Fast Oxide Breeder - Reactor Physics,
Part I - Parametric Study of 3000 MWe Reactor Core;
P. Greebler, P. Aline, and J. Sueoka
(November 10, 1959).

GEAP-3347

Fast Oxide Breeder - Stress Considerations
in Fuel Rod Design;
K. M. Horst
(March 28, 1960).

GEAP-3486

Fast Oxide Breeder Project -
Fuel Fabrication

Part I - Plutonium-Uranium Dioxide Preparation
and Pelletized Fuel Fabrication;
J. M. Cleveland and W. C. Cavanaugh

Part II - Fabrication of
Plutonium-Uranium Dioxide Specimens by Swaging;
M. E. Snyder and W. C. Cowden
(August 15, 1960).

GEAP-3487

Fast Oxide Breeder -
Preliminary Sintering Studies
of Plutonium—Uranium Dioxide Pellets;
J. M. Cleveland and W. C. Cavanaugh
(August 15, 1960).

GEAP-3646

Calculation of Doppler Coefficient

and Other Safety Parameters
for a Large Fast Oxide Reactor;
P. Greebler, B. A. Hutchins, and J. R. Sueoka
(March 9, 1961).

GEAP-3721
Core Design Study for a 500 MWe
Fast Oxide Reactor;
K. M. Horst, B. A. Hutchins,
F. J. Leitz, and B. Wolfe
(December 18, 1961).

GEAP-3824
Fabrication Cost Estimate
for UO_2 and Mixed PuO_2 - UO_2 Fuel;
G. D. Collins
(January 24, 1962).

GEAP-3833
The Post-Irradiation Examination
of a PuO_2 - UO_2 Fast Reactor Fuel;
J. M. Gerhart
(November 1961).

GEAP-3856
Experimental Fast Oxide Reactor;
K. P. Cohen, M. J. McNelly and B. Wolfe
(November 27, 1961).

GEAP-3876
Plutonium Fuel Processing and Fabrication
for Fast Ceramic Reactors;
H. W. Alter, G. D. Collins, and E. L. Zebroski
(February 1, 1962).

GEAP-3880
Comparative Study of PuC-UC and PuO_2 - UO_2
as Fast Reactor Fuel,

Part I - Technical Considerations;
K. M. Horst and B. A. Hutchins
(February 15, 1962).

Part II - Economic Considerations;
G. D. Collins
(November 15, 1962).

GEAP-3885
Experimental Fast Ceramic Reactor Design,

Status Report as of October 31, 1961;
Edited by K. M. Horst
(April 24, 1962).

GEAP-3923
Resonance Integral Calculations
for Evaluation of Doppler Coefficients -
The RAPTURE Code;
J. H. Ferziger, P. Greebler, M. D. Kelly,
and J. Walton
(June 12, 1962).

GEAP-4028
A Fuel Reprocessing Plant
for Fast Ceramic Reactors;
H. W. Alter
(February 1, 1962).

GEAP-4058
Analytical Studies of Transient Effects
in Fast Reactor Fuels;
R. B. Osborn and D. B. Sherer
(August 1962).

GEAP-4090
FORE - A Computational Program
for the Analysis of Fast Reactor Excursions;
P. Greebler and D. B. Sherer
(October 1962).

GEAP-4092
Doppler Calculations for
Large Fast Ceramic Reactors,
Effects of Improved Methods
and Recent Cross Section Information;
P. Greebler and E. Goldman
(December 1962).

GEAP-4130
Experimental Studies of Transient Effects
in Fast Reactor Fuels
Series I, UO_2 Irradiations;
J. H. Field
(November 15, 1962).

GEAP-4426
Conceptual Mechanical Design
of 565 MWe Fast Ceramic Reactor;
A. G. Silvestor
(April 1963).

GEAP-4271

Measurement of Oxygen-to-Metal Ratio
in Solid Solutions of UO_2 and PuO_2 ;
W. L. Lyon
(May 31, 1963).

GEAP-4283

Experimental Studies of Sodium Logging in FCR Fuels;
G. L. O'Neill, P. E. Novak, M. L. Johnson,
and W. E. Baily
(September 1963).

GEAP-4350

Fast Reactor Meltdown Excursions-
Analysis and Computer Program FARM I;
B. C. Luci, D. R. Riley, and N. F. Friedman
(December 5, 1963).

GEAP-4410

Calculations of Doppler and
Sodium Reactivity Effects;
P. Greebler
(November 1963).

GEAP-4414

Reactor Safety and Fuel Cycle Economics;
K. P. Cohen
(November 1963).

GEAP-4418

Liquid Metal Fast Breeder Reactor Design Study,
1000 MWe PuO_2 - UO_2 Fueled Plant,
Vols. I and II;
M. J. McNelly, et al.
(January 1964).

GEAP-4420

Influence of Doppler Effect on Meltdown Accident;
N. F. Friedman
(December 1963).

GEAP-4466

Experimental Studies of
Plutonium and Fission Product Redistribution
in Fast Reactor Fuels;
T. Lauritzen, et al.
(July 1966).

GEAP-4469

Experimental Studies of Transient Effects
in Fast Reactor Fuels,
Series III,
Pre-Irradiated Mixed Oxide (PuO_2 - UO_2),
Irradiations Final Report,
Transient Irradiations;
J. E. Hanson, J. H. Field
(February 1968).

GEAP-4471

Calculated Nuclear Parameters and
Their Uncertainties in a 1000 MWe FCR;
P. Greebler, et al.
(October 1966).

GEAP-4472

User Requirements for Cross Sections
in the Energy Range from 100 eV to 100 keV;
P. Greebler and B. A. Hutchins
(1966).

GEAP-4580

FLUX—A Computer Code
for Analyzing the Reactor Response
to Small Reactivity Oscillations;
P. Greebler, B. A. Hutchins, and M. D. Kelley
(March 22, 1965).

GEAP-4630

The Effects of Resonance Overlap on
the Doppler Coefficient in a Fast Ceramic Reactor,
Computer Codes RAVE and DOPIE;
B. A. Hutchins
(August 1964).

GEAP-4675

Some Electrical Properties of
the Plutonium-Uranium Mixed Oxide System;
W. E. Baily
(July 1966).

GEAP-4770

A Technical and Economic Evaluation of
Vented Fuel for Sodium-Cooled Fast Ceramic Reactors;
G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight,
and D. B. Sherer
(May 1965).

GEAP-4787

An Analysis of Fast Reactor Transient Response and Safety in Selected Accidents;
W. Meinhardt, Jr.
(August 1966).

GEAP-4804

Experimental Studies of
Transient Effects in Fast Reactor Fuels,
Series II;
J. E. Hanson, J. H. Field, and S. A. Rabin
(June 1965).

GEAP-4809

Fast Reactor Meltdown Accidents
Using Bethe-Tait Analysis;
R. Meyer and B. Wolfe
(January 1957).

GEAP-4819

Experimental Studies of
Transient Effects in Fast Reactor Fuels,
Series III,
Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$) Irradiations,
Interim Report, Steady State Irradiations;
J. E. Hanson and S. A. Rabin
(September 1965).

GEAP-4822

Studies of Alternate Core Design Concepts
for Fast Sodium Cooled Reactors;
C. L. Cowan
(November 1966).

GEAP-4870

Some Design and Irradiation Experience
with Capsules for Long Ceramic Fuel Elements;
F. W. Knight
(July 1966).

GEAP-4877

Diffusion of Plutonium and Uranium
in Oxide Fuel at 2400°C ;
J. Davies and P. Novak
(August 1966).

GEAP-4878

The Solid-Liquid Phase Diagram
for $\text{UO}_2\text{-PuO}_2$ System;
W. L. Lyon and W. E. Baily
(May 1966).

GEAP-4879

Thermal Conductivity of
Uranium and Uranium Plutonium Oxides;
T. E. Lannin
(January 1968).

GEAP-4897

Irradiation Behavior of
Hyper- and Hypo-Stoichiometric Plutonia-Urania Fuel
at $70,000\text{ MWd/Te}$;
E. L. Zebroski, W. E. Baily, and D. P. Hines
(March 1965).

GEAP-4922

BISYN—
A Two Dimensional Synthesis Program;
M. D. Kelley
(December 1966).

GEAP-4939

Safety and Economic Characteristics of
a 1000 MWe Fast Sodium Cooled Reactor Design;
K. P. Cohen and G. L. O'Neill
(April 1967).

GEAP-5271

Implications of Recent Fast Critical Experiments
on Basic Fast Reactor Design;
P. Greebler
(March 1968).

GEAP-5272

Evaluation and Computation of
Pu-239 Cross Section Data
for the ENDF/B Files;
P. Greebler, P. Aline, B. Hutchins
(January 1967).

GEAP-5273

FORE II, A Computational Program for
the Analysis of Steady-State and Transient
Reactor Performance;
B. E. Lawler, et al.
(September 1967).

GEAP-5340

Neutron Spectrum Measurements in a Fast Plutonium
Assembly ZPR-III Assembly 48;
P. S. Brown and A. Weitzberg
(April 1967).

GEAP-5344

Electron Microanalysis of Radioactive Materials I,
Preliminary Experiments and Plans for a Shielded Facility;
H. Rosenbaum, et al.
(February 1968).

GEAP-5355A

Burnup: A FORTRAN IV Code for Computing U and Pu
Fuel Burnup from U, Pu, and Nd Mass Spectrometric
Measurements;
B. F. Rider

GEAP-5411

Turbulent Velocity Distribution in a Rod Bundle;
D. J. Bender, D. M. Switick, J. H. Field
(January 1968)

GEAP-5493

The Effect of Density on Thermal Conductivity
of Uranium Dioxide;
R. Asamoto
(February 1968).

GEAP-5494

RAPFU—A Computer Code;
P. Greebler, C. Cowan
(November 1967).

GEAP-5556

In-Pile Comparison of Effective
Thermal Conductivity (Pu-UO₂) Mixed Oxides;
C. N. Craig, W. E. Baily
(March 1968).

GEAP-5583

The Role of Fission Products
in the Swelling of Irradiated UO₂-(U, Pu)O₂ Fuel;
F. Anselin
(January 1969).

GEAP-5633

Compatibility of LMFBR Fuel Pin Components;
T. A. Lauritzen
(June 1968).

GEAP-5635

Significance of Neutron Data Uncertainties
to Fast Reactor Economics and Power Plant Design;
B. A. Hutchins, et al.
(June 1968).

GEAP-5683

Thermodynamic Analysis
of Possible Chemical Interactions in the System,
UO₂-PuO₂ Fuel-Sodium-Stainless Steel;
E. A. Aitken
(July 1968).

GEAP-5744

Analytical Procedures and Applications
of Fluence Determinations from EBR-II Flux Wires;
A. Withop, B. A. Hutchins, G. C. Martin
(March 1968).

GEAP-5748

Fuel Cladding Heat Transfer Coefficients
for Oxide Fuel Rods;
C. N. Craig, et al.
(January 1969).

GEAP-5761

The Irradiation of Mixed Oxide Fuel
to 125,000 MWd/Te in a Thermal Flux;
B. F. Rubin and K. J. Perry
(March 1969).

GEAP-5570

Short Term Fast Flux (EBR-II) Irradiation
of PuO₂-UO₂ Fuel Pins;
S. A. Rabin, W. W. Kendall, W. E. Baily
(February 1968).

GEAP-10046

BISYN—PERT;
A Two-Dimensional Perturbation Module
for the BISYN System;
M. D. Kelley and T. R. Jones
(April 1969).

GEAP-10052

Turbulent Heat Transfer in a Rod Bundle
with Liquid Metal Coolant;
D. J. Bender and P. M. Magee
(July 1969).

GEAP-10054

Compressive Creep Characteristics
of Stoichiometric Uranium Dioxide;
P. E. Bohaboy, R. R. Asamoto, A. E. Conti
(May 1969).

GEAP-10062

Mechanical Properties of Irradiated Type-304
and Type-316 Stainless Steel in Fast Reactors;
F. A. Comprelli, H. J. Busboom, M. F. Gebhardt
(June 1969).

GEAP-10066

Mechanical Properties Evaluation of Austenitic Stainless Steels Irradiated in EBR-II;

T. Lauritzen, A. Withop, G. P. Ferguson
(July 1969).

GEAP-13553

An Evaluation of the Quality of Seamless Austenitic Stainless Steel Tubing;

T. A. Lauritzen
(December 1969).

GEAP-13515

The Melting Point of Plutonia-Urania Mixed Oxides Irradiated to High Burnup;

J. L. Krankota, C. N. Craig
(July 1969).

GEAP-13556

BICYCL - A Computer Code for Two-Dimensional Analysis of Equilibrium Fuel Cycles;

P. C. Vaughn
(December 1969).

GEAP-13543

Molten Fuel Movement in Transient Overpower Tests of Irradiated Oxide Fuel;

T. Hikido
(September 1969).

GEAP-13591

Pu-239 Unresolved Resonance Data for ENDF/B Version II;

J. T. Hitchcock and B. A. Hutchins
(April 1970).

GEAP-13549

Performance of Plutonium-Uranium Mixed Oxide Fuel Pins in a Fast Reactor (EBR-II) to 50,000 MWd/Te;

R. C. Nelson
(August 1969)

GEAP-13551

Effects of Nuclear Methods Approximations on LMFBR Analysis;

R. Protsik
(August 1969).

GEAP-13592

ENDRUN-1, A Computer Code to Generate A Generalized Multigroup Data File from ENDF/B;

B. A. Hutchins and L. N. Price
(April 1970).

CONTRIBUTORS

Task B—Defected Fuel Behavior Task Leader—P. E. Novak

P. E. Bohaboy
J. M. Bridges
W. L. Conant
R. J. Ernst
K. E. Gregoire
R. E. Harris
G. L. Howe
J. L. Linssen
E. A. Melaika
R. E. Murata
E. D. Ormiston
D. E. Plumlee
G. L. Stimmell
H. J. Stock
J. G. Telles
G. L. Tunnell
E. E. Walker

Task C-1—Safety Experimentation (Transient Overpower) Task Leader—J. H. Field

D. L. Brown
T. Hikido
J. L. Scofield
G. R. Thomas
R. C. Thompson
W. L. Conant
S. P. Darr
J. P. Gregg
D. E. Hafer
S. Lukezic
R. E. Smith
G. L. Stimmell
D. S. Vash
D. L. Zimmerman

Task C-2—Safety Experimentation (Loss of Flow) Task Leader—H. C. Pfefferlen

D. L. Brown
C. A. Calkins
W. L. Conant
M. D. Dermer
J. H. Field
K. G. Janberg
R. G. Stuart
R. C. Thompson
G. W. Tunnell

Task E—Fuel Materials Properties and Performance Evaluation Task Leader—B. F. Rubin

E. A. Aitken
P. E. Bohaboy
D. A. Cantley
W. L. Conant
R. N. Duncan
S. K. Evans
J. Haskins
C. L. Hilf
S. Jain
T. E. Lannin
J. E. Lewis
T. E. Ludlow
E. J. McAlpine
G. F. Melde
C. R. Mijares
P. E. Ray
H. S. Rosenbaum
R. E. Smith

Task F—Fast Flux Tests and Support Task Leader—C. N. Craig

R. R. Asamoto
C. F. Barrett
W. Bennett
G. C. Berg
N. I. Brown
D. A. Cantley
W. L. Conant
A. E. Conti
L. S. Dorfman
R. N. Duncan
R. C. Gebhardt
C. L. Hilf
D. P. Hines
G. R. Hull
S. K. Jain
T. H. Koppel
E. F. Kurtz
T. A. Lauritzen
E. J. McAlpine
W. H. McCarthy
T. R. McIntyre
C. R. Mijares
G. F. Melde
T. S. Oldberg
K. J. Perry

P. E. Ray
C. C. Ripley
R. E. Smith
M. L. Thompson
C. G. Valente
D. L. Zimmerman

Task G-6—Core Design Analysis
Task Leader—P. R. Pluta

K. R. Absher
B. D. Atcheson
A. T. Brown
J. M. Cole
W. J. Delaney
J. H. Field
W. R. Gee, Jr.
M. B. Goldstein
S. Kaplan
L. M. McWethy
W. G. Meinhardt
J. R. Punches
C. C. Ripley
F. A. Sciacca
D. B. Sherer
H. J. Snyder
M. I. Temme
P. M. Tschamper

Task G-10—Kinetic Analysis
Task Leader—J. H. Field

D. L. Brown
D. D. Freeman
J. T. Hitchcock
K. G. Janberg
P. M. Magee
R. G. Stuart
R. C. Thompson
T. F. Thompson

Task L—Cladding Development
Task Leader—H. J. Busboom

W. K. Appleby
J. S. Armijo

R. R. Asamoto
A. E. Conti
J. G. Douglass
J. D. Eveland
R. H. Eymann
G. P. Ferguson
D. H. Frisby
T. A. Lauritzen
A. Montes
R. P. Rhoads
G. F. Rieger
P. J. Ring
D. W. Sandusky
C. N. Spalaris
W. J. Wagner
D. L. Zimmerman

Task N—Core Physics Analysis
Task Leader—B. A. Hutchins

N. Barth
C. L. Cowan
S. C. Crick
P. Greebler
M. D. Kelley
R. S. Seymour
J. E. Turner

Task P—Fuel Element Design
Task Leader—D. P. Hines

C. F. Barrett
D. K. English
D. P. Hines
M. O. Marlowe
T. Oldberg
M. B. Reynolds
J. D. Stephen

Program Planning and Coordination

W. E. Baily
C. N. Spalaris
E. L. Zebroski

Assistant Director
Engineering Standards
U. S. Atomic Energy Commission
Washington, D.C. 20545

RDT Site Office
U. S. Atomic Energy Commission
Argonne National Laboratory
Building 2
Argonne, Illinois 60439

Assistant Director, Nuclear Safety
U. S. Atomic Energy Commission
Washington, D.C. 20545

RDT Site Office
U. S. Atomic Energy Commission
Atomics International
P. O. Box 1446
Canoga Park, California 91304

Assistant Director,
Plant Engineering
U. S. Atomic Energy Commission
Washington, D.C. 20545

RDT Site Office
U. S. Atomic Energy Commission
Argonne National Laboratory
P. O. Box 2108
Idaho Falls, Idaho 83401

Assistant Director
Program Analysis
U. S. Atomic Energy Commission
Washington, D. C. 20545

RDT Site Office
U. S. Atomic Energy Commission
General Electric Company
310 DeGuigne Drive
Sunnyvale, California 94086

Assistant Director
Project Management
U. S. Atomic Energy Commission
Washington, D.C. 20545

(2)

RDT Site Office
U. S. Atomic Energy Commission
Gulf General Atomic Incorporated
P. O. Box 2325
San Diego, California 92112

Assistant Director
Reactor Engineering
U. S. Atomic Energy Commission
Washington, D.C. 20545

(2)

Division of Naval Reactors
Chief, Nuclear Materials Branch
U. S. Atomic Energy Commission
Washington, D.C. 20545

Assistant Director
Reactor Technology
U. S. Atomic Energy Commission
Washington, D.C. 20545

Chief
Fuels and Materials Branch
U. S. Atomic Energy Commission
Washington, D.C. 20545

(3)

Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

(2)

Chief
Reactor Vessels Branch
U. S. Atomic Energy Commission
Washington, D.C. 20545

Director, Contracts Division
U. S. Atomic Energy Commission
San Francisco Operations Office
2111 Bancroft Way
Berkeley, California 94704

Director
Metallurgy Division
Argonne National Laboratory (2)
9700 South Cass Avenue
Argonne, Illinois 60439

Division Leader (CMB)
Chemistry and Metallurgy Division
Los Alamos Scientific Laboratory
P. O. Box 1663
Los Alamos, New Mexico 87544

Manager,
FFTF Project (6)
Pacific Northwest Laboratory
P. O. Box 999
Richland, Washington 99352

FFTF Fuels Department (2)
Pacific Northwest Laboratory
P. O. Box 999
Richland, Washington 99352

Manager
Chemistry and Metallurgy Division
Pacific Northwest Laboratory
P. O. Box 999
Richland, Washington 99352

Director
Metallurgy and Materials
Science Division
Brookhaven National Laboratory
Upton, New York 11973

RDT Site Office
U. S. Atomic Energy Commission
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830

Division Chief, M&S Division
NASA - Lewis Research Center
21000 Brookpark Road
Cleveland, Ohio 44135

Director
Division of Reactor Standards (2)
U. S. Atomic Energy Commission
Washington, D.C. 20545

General Manager
Westinghouse Electric Corporation
Bettis Atomic Power Laboratory
P. O. Box 70
West Mifflin, Pennsylvania 15122

Division of Technical Information
Extension (3)* (50)**
U. S. Atomic Energy Commission
P. O. Box 62
Oak Ridge, Tennessee 37831

Associate Manager
Materials Engineering Department
Battelle Memorial Institute
Columbus Laboratories
505 King Avenue
Columbus, Ohio 43201

Director
LMFBR Program Office (2)
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Manager, Nuclear Laboratories
Combustion Engineering Inc.
Nuclear Division
Prospect Hill Road
Windsor, Connecticut 06095

Director
Chemical Engineering Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Manager, Plutonium Chemistry
and Ceramics Fuels Development
Nuclear Materials and Equipment
Corporation
Plutonium Laboratory
Leechburg, Pennsylvania 15656

Head, Fuels and Materials
Atomic Power Development Associates
1911 First Street
Detroit, Michigan 48226

Director (3)
Nuclear Safety Program
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830

Director
Metals and Ceramics Division (2)
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830

K-2 Group Leader
Reactor Division
Los Alamos Scientific Laboratory
P. O. Box 1663
Los Alamos, New Mexico 87544

Director, Liquid Metal
Engineering Center
Atoms International
P. O. Box 1449
Canoga Park, California 91304

Westinghouse Electric Corporation
EBR-II Project
Argonne National Laboratory
P. O. Box 2528
Idaho Falls, Idaho 83401

Manager-Advanced Development
Activity
General Electric Company
Knolls Atomic Power Laboratory
P. O. Box 1072
Schenectady, New York 12301

Experiment Manager
EBR-II Project
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

Director, LMFBR Technology Program
Atoms International
P. O. Box 309
Canoga Park, California 91304

Manager
Liquid Metal Information Center
Atoms International
P. O. Box 1449
Canoga Park, California 91304

Director
Nuclear Development Center
The Babcock and Wilcox Company
Atomic Energy Division
Lynchburg, Virginia 24501

Irradiation Coordinator
EBR-II Project
Argonne National Laboratory
P. O. Box 2528
Idaho Falls, Idaho 83401

Laboratory Assistant Director
Gulf General Atomic Incorporated
P. O. Box 608
San Diego, California 92112

Manager, Research
United Nuclear Corporation
Research and Engineering Center
Grasslands Road
Elmsford, New York 10523